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

June 11, 2020

Ms. Cheryl A. Gayheart
Regulatory Affairs Director
Southern Nuclear Operating Co., Inc.
3535 Colonnade Parkway
Birmingham, AL 35243

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 304 AND 249, REGARDING LICENSE AMENDMENT REQUEST TO ADOPT NFPA-805 PERFORMANCE BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR ELECTRIC GENERATING PLANTS (2001 EDITION) (EPID L-2018-LLA-0107)

Dear Ms. Gayheart:

The Nuclear Regulatory Commission (NRC, the Commission) has issued the enclosed Amendment No. 304 to Renewed Facility Operating License (RFOL) No. DPR-57 and Amendment No. 249 to RFOL No. NPF-5 for the Edwin I. Hatch Nuclear Plant (Hatch), Unit Nos 1 and 2, respectively.

The amendments consist of changes to the FOLs and Technical Specifications (TSs) in response to your application dated April 4, 2018 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML18096A936), as supplemented by letters dated May 28, August 9, October 7, December 13, 2019, February 5, and March 13, 2020 (ADAMS Accession Nos. ML19151A444, ML19225D178, ML19280C812, ML19351D121, ML20036F588, and ML20073M564 respectively).

The amendments transition the Hatch Fire Protection Program from Title 10 of the *Code of Federal Regulations* (10 CFR), Sections 50.48(a) and (b), to 10 CFR 50.48(c), National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition.

The application, as supplemented, also identified several current exemptions to 10 CFR Part 50, Appendix R, granted by the NRC on November 16, 1981; April 18, 1984; and January 2, 1987 as described in Attachment O of the letter dated April 4, 2018, that would no longer be applicable once the amendment is implemented. These exemptions are rescinded upon implementation of the amendment.

A copy of the related Safety Evaluation is also enclosed.

A Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

John G. Lamb, Senior Project Manager
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-321 and 50-366

Enclosures:
1. Amendment No. 304 to DPR-57
2. Amendment No. 249 to NPF-5
3. Safety Evaluation

cc: Listserv

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 304
Renewed License No. DPR-57

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 1 (the facility) Renewed Facility Operating License No. DPR-57 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated April 4, 2018, as supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C(3) of Renewed Facility Operating License No. DPR-57 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 304, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and as-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 National Fire Protection Association (NFPA) 805 risk assessments, or methods that have been demonstrated to bound the risk impact.

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

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1 x10^{-7}/yr for CDF and less than

1 x 10^{-8} /yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

- 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 NRC SE dated June 11, 2020, to determine that certain fire protection program changes meet the minimal criterion. The

licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) 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 (b)(2) above.

(2) The licensee shall implement the modifications described in Attachment S2, Table S-2, "Plant Modifications Committed," of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after issuance of the NRC SE. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) The licensee shall implement the items as listed in Attachment S2, Table S-3, "Implementation Items," of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2020.06.11 10:51:36 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. DPR-57 and Technical Specifications

Date of Issuance: June 11, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 304

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE NO. DPR-57

DOCKET NO. 50-321

Replace the following pages of the License and the 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 Pages	Insert Pages
License	License
1	1
2	2
3	3
4	4
5	5
5a	-
6	6
7	7
8	8
-	9
-	10
-	11
TSs	TSs
5.0-6	5.0-6

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-321

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 1

RENEWED FACILITY OPERATING LICENSE DPR-57

Renewed License No. DPR-57

1. The U.S. Nuclear Regulatory Commission (the Commission), having previously made the findings set forth in License No. DPR-57 issued on August 6, 1974[1], has now reached the following findings:

   A. The application to renew License No. DPR-57, filed by Southern Nuclear Operating Company, Inc., on behalf of Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made.

   B. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have

---

1. Following the initial filling of the application for license, Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, became co-owners with Georgia Power Company (GPC) of the Edwin I. Hatch Nuclear Plant, Unit 1, and together are hereinafter referred to as the Owners.

been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the Edwin I. Hatch Nuclear Plant, Unit 1, and any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations.

C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

D. There is reasonable assurance that (1) the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

E. Southern Nuclear Operating Company, Inc.[2] (herein called Southern Nuclear), is technically qualified and, together, Southern Nuclear and the Owners are financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission.

F. The Owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations.

G. The renewal of this operating license will not be inimical to the common defense and security or the health and safety of the public.

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs, and considering available alternatives, the Commission concludes that the issuance of this Renewed Facility Operating License No. DPR-57 is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

I. The receipt, possession, and use of source, byproduct, and special nuclear material, as authorized by this renewed license, will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31.

---

2. Southern Nuclear Operating Company, Inc. succeeds Georgia Power Company as operator of the Edwin I. Hatch Nuclear Plant, Unit 1. Southern Nuclear is authorized by the Owners to exercise exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-57, issued on October 13, 1974, is superseded by Renewed Facility Operating License No. DPR-57, which is hereby issued to Southern Nuclear Operating Company, Inc., and the Owners, to read as follows:

   A. This renewed license applies to the Edwin I. Hatch Nuclear Plant, Unit No. 1, a direct-cycle, boiling-water reactor and associated equipment (the facility), owned by Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, and operated by Southern Nuclear. The facility is located 11 miles north of Baxley, in Appling County, Georgia, and is described in the Updated Final Safety Analysis Report, as supplemented and amended, and the Environmental Report, as supplemented and amended.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the following:

      (1) Southern Nuclear, pursuant to Section 104b of the Act and 10 CFR Part 50, to possess, manage, use, maintain, and operate the facility at the designated location in Appling County, Georgia, in accordance with the procedures and limitations set forth in this renewed license

      (2) Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, pursuant to Section 104b of the Act and 10 CFR Part 50, to possess, but not operate, the facility at the designated location in Appling County, Georgia, in accordance with the procedures and limitations set forth in this license

      (3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended

      (4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and fission detectors in amounts as required

      (5) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required, any byproduct, source or special nuclear material, without restriction to chemical or physical form,

for sample analysis or instrument calibration, or associated with radioactive apparatus or components

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

C. This renewed 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, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional conditions specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 304, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

The Surveillance Requirement (SR) contained in the Technical Specifications and listed below, is not required to be performed immediately upon implementation of Amendment No. 195. The SR listed below shall be successfully demonstrated before the time and condition specified:

SR 3.8.1.18 shall be successfully demonstrated at its next regularly scheduled performance.

(3) Fire Protection

Southern Nuclear Operating Company shall implement and maintain 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 April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 10, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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

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

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

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

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

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

- 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 NRC SE dated June 10, 2020, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) 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 (b)(2) above.

(2) The licensee shall implement the modifications described in Attachment S2, Table S-2, “Plant Modifications Committed,” of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after the issuance of the NRC SE. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

(3) The licensee shall implement the items as listed in Attachment S2, Table S-3, “Implementation Items,” of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

(4.a) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: “Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan,” with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 265, as supplemented by a change approved by License Amendment No. 274.

(4.b) Mitigation Strategy License Condition

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

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

(b) Operations to mitigate fuel damage considering the following:
1. Protection and use of personnel assets
2. Communications
3. Minimizing fire spread
4. Procedures for implementing integrated fire response strategy

5. Identification of readily-available pre- staged equipment
6. Training on integrated fire response strategy
7. Spent fuel pool mitigation measures

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

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

(5) FSAR Supplement

The licensee's Final Safety Analysis Report Supplement, dated September 5, 2001, shall be included in the next Updated Final Safety Evaluation Analysis Report update, required by 10 CFR 50.71(e).

(6) Safety Analysis Report

The licensee's Final Safety Analysis Report Supplement, dated September 5, 2001, submitted pursuant to 10 CFR 54.21(d), describes certain future inspection activities to be completed before the period of extended operations begins. The licensee shall complete those activities no later than August 6, 2014.

(7) Integrated Surveillance Program

The licensee shall implement a staff-approved reactor vessel integrated surveillance program for the extended period of operation which satisfies the requirements of 10 CFR Part 54. Such a program will be implemented through a staff-approved Boiling Water Reactor Vessel and Internals Project program or through a staff-approved plant-specific program. The plant-specific program, if needed, will be developed in a manner that is consistent with other aging management programs, will include consideration of the 10 program attributes utilized for other aging management programs, and will provide a technical justification for any program attribute not covered by the plant-specific surveillance material testing program. The plant-specific program, if needed, will include the following actions:

(a) Capsules will periodically be removed to determine the rate of embrittlement.

(b) Capsules will be removed at neutron fluence levels that provide relevant data for assessing the integrity of the Plant Hatch, Unit 1 reactor pressure vessel (in particular, for the determination of

reactor pressure vessel pressure-temperature limits through the period of extended operation).

(c) Capsules will contain material to monitor the impact of irradiation on the Plant Hatch Unit 1 reactor pressure vessel and will contain dosimetry to monitor neutron fluence.

Before the renewal term begins, the licensee will notify the NRC of its decision to implement the integrated surveillance program or a plant-specific program, and provide the appropriate revisions to the Updated Final Safety Analysis Report Supplement summary descriptions of the vessel surveillance material testing program.

(8) Design Bases Accident Radiological Consequences Analyses

Southern Nuclear is authorized to credit administering potassium iodide to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room until May 31, 2012. Should design basis changes be completed rendering the crediting of potassium iodide no longer necessary prior to May 31, 2012, Southern Nuclear will remove the crediting of potassium iodide from the design basis accident radiological consequences analyses (reference Unit 2 FSAR paragraph 15.3.3.4.2.2) in the next Updated Final Safety Analysis Report as required by 10 CFR 50.71(e).

(9) Alternative Source Term

1) Southern Nuclear Operating Company, Inc (SNC, the licensee) shall complete actions by April 30, 2010, as described in SNC's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be capable of being manually switched over from normally operating power supplies, to a Class 1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

2) SNC shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008 (NL 08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible

instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

3) SNC shall complete actions by May 31, 2010, as described in SNC's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

4) SNC shall complete actions by May 31, 2012 for HNP Unit 1, as described in SNC's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve alternate leakage treatment boundary valves, such that they can be closed in the event of a loss of offsite power without requiring local operation:

1N38-F101A, 1N38-F101B, 1N33-F012, 1N33-F013

5) SNC shall implement actions by May 31, 2010, as described in SNC's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

(10) <u>TSTF-448, Control Room Habitability</u>

Upon implementation of the Amendments adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.14.c.(i), the assessment of CRE habitability as required by Specification 5.5.14.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.14.d, shall be considered met. Following implementation:

a. The first performance of SR 3.7.4.4, in accordance with Specification 5.5.14.c.(i), shall be within the next 18 months.

b. The first performance of the periodic assessment of CRE habitability, Specification 5.5.14.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, of the next successful tracer gas test.

c. The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

D. Southern Nuclear shall not market or broker power or energy from Edwin I. Hatch Nuclear Plant, Unit 1.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, August 6, 2034.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan

Date of Issuance: January 15, 2002

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

d. DELETED

e. All programs and manuals specified in Specification 5.5.

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY

OGLETHORPE POWER CORPORATION

MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA

CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 249
Renewed License No. NPF-5

1. The Nuclear Regulatory Commission (NRC, the Commission) has found that:

   A. The application for amendment to the Edwin I. Hatch Nuclear Plant, Unit No. 2 (the facility) Renewed Facility Operating License No. NPF-5 filed by Southern Nuclear Operating Company, Inc. (the licensee), acting for itself, Georgia Power Company, Oglethorpe Power Corporation, Municipal Electric Authority of Georgia, and City of Dalton, Georgia (the owners), dated April 4, 2018, as supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as 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 set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraphs 2.C.(2) and 2.C(3)(a) of Renewed Facility Operating License No. NPF-5 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 249 are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3)(a) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in *the* licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and as-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.

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

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1×10^{-7}/yr for CDF and less than 1×10^{-8}/yr for LERF) The proposed change must also be consistent with

the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

(4) 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 component, system, procedure, or physical arrangement functionality 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 component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- 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 NRC SE dated June 11, 2020, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety

margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) 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 (b)(2) above.

(2) The licensee shall implement the modifications described in Attachment S2, Table S-2, "Plant Modifications Committed," of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after issuance of the NRC SE. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) The licensee shall implement the items as listed in Attachment S2, Table S-3, "Implementation Items," of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Michael T. Markley

Digitally signed by Michael T. Markley
Date: 2020.06.11 10:52:59 -04'00'

Michael T. Markley, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility Operating License No. NPF-5 and Technical Specifications

Date of Issuance: June 11, 2020

ATTACHMENT TO LICENSE AMENDMENT NO. 249

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NO. NPF-5

DOCKET NO. 50-366

Replace the following pages of the License and the 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 Pages	Insert Pages
License	License
1	1
2	2
3	3
4	4
5	5
5a	-
6	6
6a	-
6b	-
7	7
8	8
9	9
10	10
11	11
-	12
-	13
-	14
-	15
-	16
TSs	TSs
5.0-6	5.0-6

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

SOUTHERN NUCLEAR OPERATING COMPANY, INC.

GEORGIA POWER COMPANY
OGLETHORPE POWER CORPORATION
MUNICIPAL ELECTRIC AUTHORITY OF GEORGIA
CITY OF DALTON, GEORGIA

DOCKET NO. 50-366

EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

RENEWED FACILITY OPERATING LICENSE NPF-5

Renewed License No. NPF-5

1. The U.S. Nuclear Regulatory Commission (the Commission), having previously made the findings set forth in License No. NPF-5 issued on June 13, 1978, has now reached the following findings:

   A. The application to renew License No. NPF-5, filed by Southern Nuclear Operating Company, Inc., on behalf of Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made.

   B. Actions have been identified and have been or will be taken with respect to (1) managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under 10 CFR 54.21(a)(1), and (2) time-limited aging analyses that have been identified to require review under 10 CFR 54.21(c), such that there is reasonable assurance that the activities authorized by this renewed license will continue to be conducted in accordance with the current licensing basis, as defined in 10 CFR 54.3, for the Edwin I. Hatch Nuclear Plant, Unit 2, and any changes made to the plant's current licensing basis in order to comply with 10 CFR 54.29(a) are in accord with the Act and the Commission's regulations.

   C. The facility requires exemptions from certain requirements of (1) 10 CFR 50.55a(g)(2) and (2) Appendices G and H to 10 CFR Part 50. These exemptions are described in the safety evaluations supporting the granting of

these exemptions, prepared by the NRC's Office of Nuclear Reactor Regulation, which were enclosed with the letter transmitting the original license, dated June 13, 1978. These exemptions were granted. With these exemptions, the facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

D. There is reasonable assurance that (1) the activities authorized by this renewed license can be conducted without endangering the health and safety of the public, and (2) such activities will be conducted in compliance with the rules and regulations of the Commission.

E. Southern Nuclear Operating Company, Inc.[1] (herein called Southern Nuclear), is technically qualified and, together, Southern Nuclear and the Owners, are financially qualified to engage in the activities authorized by this renewed license in accordance with the rules and regulations of the Commission.

F. The Owners have satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations.

G. The renewal of this operating license will not be inimical to the common defense and security or to the health and safety of the public.

H. After weighing the environmental, economic, technical, and other benefits of the facility against environmental costs, and considering available alternatives, the Commission concludes that the issuance of this Renewed Facility Operating License No. NPF-5 subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

I. The receipt, possession, and use of source, byproduct, and special nuclear material, as authorized by this renewed license, will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23, and 70.31.

---

1. Southern Nuclear Operating Company, Inc. succeeds Georgia Power Company as operator of the Edwin I. Hatch Nuclear Plant, Unit 2. Southern Nuclear is authorized by the Owners to exercise exclusive responsibility and control over the physical construction, operation, and maintenance of the facility.

2. On the basis of the foregoing findings regarding this facility, Facility Operating License No. NPF-5, issued on June 13, 1978, is superseded by Renewed Facility Operating License No. NPF-5, which is hereby issued to Southern Nuclear Operating Company, Inc., and the Owners, to read as follows:

   A. This renewed license applies to the Edwin I. Hatch Nuclear Plant, Unit No. 2, a boiling-water reactor and associated equipment (the facility), owned by Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, and operated by Southern Nuclear. The facility is located in Appling County, Georgia, and is described in the Updated Final Safety Analysis Report, as supplemented and amended, and the Environmental Report, as supplemented and amended.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses the following:

      (1) Southern Nuclear, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, manage, use, maintain, and operate the facility at the designated location in Appling County, Georgia, in accordance with the procedures and limitations set forth in this renewed license

      (2) Georgia Power Company, the Oglethorpe Power Corporation, the Municipal Electric Authority of Georgia, and the City of Dalton, Georgia, pursuant to Section 103 of the Act and 10 CFR Part 50, to possess, but not operate, the facility at the designated location in Appling County, Georgia, in accordance with the procedures and limitations set for in this license

      (3) Southern Nuclear, pursuant to the Act and 10 CFR Part 70, to receive, possess, and use at any time, special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended

      (4) Southern Nuclear, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time, any byproduct, source, and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation, and radiation monitoring equipment calibration, and fission detectors in amounts as required

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

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

C. This renewed 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, Section 50.54 of Part 50, and Section 70.32 of Part 70; all applicable provisions of the Act and the rules, regulations, and orders of the Commission now or hereafter in effect; and the additional condition[2] specified or incorporated below:

(1) Maximum Power Level

Southern Nuclear is authorized to operate the facility at steady-state reactor core power levels not in excess of 2,804 megawatts thermal, in accordance with the conditions specified herein.

(2) Technical Specifications

The Technical Specifications (Appendix A) and the Environmental Protection Plan (Appendix B), as revised through Amendment No. 249, are hereby incorporated in the renewed license. Southern Nuclear shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(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 license supported by a favorable evaluation by the Commission.

(a) Fire Protection

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 10, 2020. 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

2. The original licensee authorized to possess, use, and operate the facility was Georgia Power Company (GPC). Consequently, certain historical references to GPC remain in certain license conditions.

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.

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

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

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

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

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

a) 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 component, system, procedure, or physical arrangement functionality 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 component, system, procedure, or physical arrangement functionality using a relevant technical requirement or standard. The four specific sections of NFPA 805, Chapter 3, are:

- 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.

b) 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 NRC SE dated June 10, 2020, 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.

(3) Transition License Conditions

a) Before achieving full compliance with 10 CFR 50.48(c), as specified by (b) and (c) 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)(b) above.

b) The licensee shall implement the modifications described in Attachment S2, Table S-2, "Plant Modifications Committed," of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after the issuance of the NRC SE. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

c) The licensee shall implement the items as listed in Attachment S2, Table S-3, "Implementation Items," of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

(b.1) Physical Protection

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The plan is entitled: "Southern Nuclear Operating Company Security Plan, Training and Qualification Plan, and Safeguards Contingency Plan," with revisions submitted through May 15, 2006.

Southern Nuclear shall fully implement and maintain in effect all provisions of the Commission-approved cyber security plan (CSP), including changes made pursuant to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The Southern Nuclear CSP was approved by License Amendment No. 209, as supplemented by a change approved by License Amendment No. 219.

(b.2) Mitigation Strategy License Condition

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

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

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

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

(b.3) The licensee shall implement and maintain all Actions required by Attachment 2 to NRC Order EA-06-137, issued June 20, 2006, except the last action that requires incorporation of the strategies into the site security plan, contingency plan, emergency plan and/or guard training and qualification plan, as appropriate.

(c) FSAR Supplement

The licensee's Final Safety Analysis Report Supplement dated September 5, 2001, shall be included in the next Updated Final Safety Analysis Report update, required by 10 CFR 50.71(e).

(d) Safety Analysis Report

The licensee's Final Safety Analysis Report Supplement dated September 5, 2001, submitted pursuant to 10 CFR 54.21(d), describes certain future inspection activities to be completed before the period of extended operations begins. The licensee shall complete those activities no later than June 13, 2018.

(e) Integrated Surveillance Program

The licensee shall implement a staff-approved reactor vessel integrated surveillance program for the extended period of

operation which satisfies the requirements of 10 CFR Part 54. Such a program will be implemented through a staff-approved Boiling Water Reactor Vessel Internals Project program or through a staff-approved plant-specific program. The plant-specific program, if needed, will be developed in a manner consistent with other aging management programs, will include consideration of the 10 program attributes utilized for other aging management programs, and will provide a technical justification for any program attribute not covered by the plant-specific surveillance material testing program. The plant-specific program, if needed, will include the following actions:

i. Capsules will periodically be removed to determine the rate of embrittlement.

ii. Capsules will be removed at neutron fluence levels that provide relevant data for assessing the integrity of the Plant Hatch Unit 2 reactor pressure vessel (in particular, for the determination of reactor pressure vessel pressure-temperature limits through the period of extended operation).

iii. Capsules will contain material to monitor the impact of irradiation on the Plant Hatch Unit 2 reactor pressure vessel and will contain dosimetry to monitor neutron fluence.

Before the renewal term begins, the licensee will notify the NRC of its decision to implement the integrated surveillance program or a plant-specific program, and provide the appropriate revisions to the Updated Final Safety Analysis Report Supplement summary descriptions of the vessel surveillance material testing program.

(f) Design Bases Accident Radiological Consequences Analyses

Southern Nuclear is authorized to credit administering potassium iodide to reduce the 30 day post-accident thyroid radiological dose to the operators in the main control room until May 31, 2011. Should design basis changes be completed rendering the crediting of potassium iodide no longer necessary prior to May 31, 2011, Southern Nuclear will remove the crediting of potassium iodide from the design basis accident radiological consequences analyses (reference Unit 2 FSAR paragraph 15.3.3.4.2.2) in the next Updated Final Safety Analysis Report update as required by 10 CFR 50.71(e).

(g) Alternative Source Term

i) Southern Nuclear Operating Company, Inc (SNC, the licensee) shall complete actions by April 30, 2010, as described in SNC's letters dated October 18, 2007, and March 13, 2008, to complete the design modifications to the HNP turbine building ventilation exhaust systems. Specifically, the HNP Units 1 and 2 turbine building exhaust fans shall be

capable of being manually switched over from normally operating power supplies, to a Class -1E circuit that will be isolated by an appropriately rated safety related, environmentally and seismically qualified circuit breaker. For further protection and isolation, the licensee shall also use fuses that will be located in a seismically qualified manual transfer switch housing. The aforementioned circuit breaker and fuses shall be adequately coordinated with the upstream load center breaker over the entire range. These devices shall be adequately rated to prevent adverse effects of a fault to the rest of the distribution system.

ii) SNC shall implement modifications by May 31, 2010, as described in Enclosure 1, section 2.7.3.2, of the LAR and section 5.7 of SNC's letter dated February 25, 2008, (NL 08-0175) to modify the design for the air supply to the turbine building exhaust ventilation dampers, such that operating air to the dampers will be supplied from a non-interruptible instrument air source to eliminate single failure point vulnerability to loss of system/instrument air.

iii) SNC shall complete actions by May 31, 2010, as described in SNC's letter dated February 25, 2008 (NL-08-0175) to install and implement the capability for Standby Liquid Control System hand switch jumpers for HNP Units 1 and 2.

iv) SNC shall complete actions by May 31, 2011, for HNP Unit 2, as described in SNC's letters dated February 25, 2008 (NL-08-0175) and July 2, 2008 (NL-08-1022), to modify the following Main Steam Isolation Valve alternate leakage treatment boundary valves, such that they can be closed in the event of a loss of offsite power without requiring local operation:

2N11-F004A, 2N11-F004B, 2N33-F003, 2N33-F004

v) SNC shall implement actions by May 31, 2010, as described in SNC's letter dated February 27, 2008, to assure that temperature switches which monitor charcoal bed temperature meet the environmental qualification requirements of 10 CFR 50.49.

(h) <u>TSTF-448 Control Room Habitability</u>

Upon implementation of the Amendments adopting TSTF-448, Revision 3, the determination of control room envelope (CRE) unfiltered air inleakage as required by SR 3.7.4.4, in accordance with TS 5.5.14.c.(i), the assessment of CRE habitability as required by Specification 5.5.14.c.(ii), and the measurement of CRE pressure as required by Specification 5.5.14.d, shall be considered met. following implementation:

i) The first performance of SR 3.7.4.4, in accordance with Specification 5.5.14.c.(i), shall be within the next 18 months.

ii) The first performance of the periodic assessment of CRE habitability, Specification 5.5.14.c.(ii), shall be within 3 years, plus the 9-month allowance of SR 3.0.2, of the next successful tracer gas test.

iii) The first performance of the periodic measurement of CRE pressure, Specification 5.5.14.d, shall be within 24 months, plus the 6 months allowed by SR 3.0.2, from the date of the most recent successful pressure measurement test.

D. This renewed license is subject to the following antitrust conditions:

(1) As used herein:

(a) “Entity” means any financially responsible person, private or public corporation, municipality, county, cooperative, association, joint stock association or business trust, owning, operating or proposing to own or operate equipment or facilities within the state of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) for

the generation, transmission, or distribution of electricity, provided that, except for municipalities, counties, or rural electric cooperatives, "entity" is restricted to those which are or will be public utilities under the laws of the State of Georgia or under the laws of the United States, and are or will be providing retail electric service under a contract or rate schedule on file with and subject to the regulation of the Public Service Commission of the State of Georgia or any regulatory agency of the United States, and provided further, that as to municipalities, counties, or rural electric cooperatives, "entity" is restricted to those which provide electricity to the public at retail within the State of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) or to responsible and legally qualified organizations of such municipalities, counties, and/or cooperatives in the State of Georgia (other than Chatham, Effingham, Fannin, Towns and Union Counties) to the extent they may bind their members.

(b) "Power Company" means Georgia Power Company, any successor, assignee of this license, or assignee of all or substantially all of Georgia Power Company's assets, and any affiliate or subsidiary of Georgia Power Company to the extent it engages in the ownership of any bulk power supply generation or transmission resource in the State of Georgia (but specifically not including (1) flood rights and other land rights acquired in the State of Georgia incidental to hydroelectric generation facilities located in another state and (2) facilities located west of the thread of the stream on that part of the Chattahoochee River serving as the boundary between the states of Georgia and Alabama).

(2) Power Company recognizes that it is often in the public interest for those engaging in bulk power supply and purchases to interconnect, coordinate for reliability and economy, and engage in bulk power supply transactions in order to increase interconnected system reliability and reduce the costs of electric power. Such arrangements must provide for Power Company's costs (including a reasonable return) in connection therewith and allow other participating entities full access to the benefits available from interconnected bulk power supply operations and must provide net benefits to Power Company. In entering into such arrangements neither Power Company nor any other participant should be required to violate the principles of sound engineering practice or forego a reasonably contemporaneous alternative arrangement with another, developed in good faith in arms length negotiations (but not including arrangements between Power Company and its affiliates or subsidiaries which impair entities' rights hereunder more than they would be impaired were such arrangements made in good faith between Power Company a non-affiliate or non-subsidiary) which affords it greater benefits. Any such arrangement must provide for adequate notice and joint planning procedures consistent with sound engineering practice, and must relieve Power Company from obligations undertaken by it in the event such procedures are not followed by any participating entity.

Power Company recognizes that each entity may acquire some or all of its bulk power supply from sources other than Power Company.

In the implementation of the obligations stated in the succeeding paragraphs, Power Company and entities shall act in accordance with the foregoing principles, and these principles are conditions to each of Power Company's obligations herein undertaken.

(3) Power Company shall interconnect with any entity which provides, or which has undertaken firm contractual obligations to provide, some or all of its bulk power supply from source other than Power Company on terms to be included in an interconnection agreement which shall provide for appropriate allocation of the costs of interconnection facilities; provided, however, that if an entity undertakes to negotiate such a firm contractual obligation, the Power Company shall, in good faith, negotiate with such entity concerning any proposed interconnection. Such interconnection agreement shall provide, without undue preference or discrimination, for the following, among other things, insofar as consistent with the operating necessities of Power Company's and any participating entity's systems:

(a) maintenance and coordination of reserves, including, where appropriate, the purchase and sale thereof,

(b) emergency support,

(c) maintenance support,

(d) economy energy exchanges,

(e) purchase and sale of firm and non-firm capacity and energy,

(f) economic dispatch of power resources within the State of Georgia, provided, however, that in no event shall such arrangements impose a higher percentage of reserve requirements on the participating entity than that maintained by Power Company for similar resources.

(4) Power Company shall sell full requirements power to any entity. Power Company shall sell partial requirements power to any entity. Such sales shall be made pursuant to rates on file with the Federal Power Commission, or any successor regulatory agency, and subject to reasonable terms and conditions.

(5) (a) Power Company shall transmit ("transmission service") bulk power over its system to any entity or entities with which it is interconnected, pursuant to rate schedules on file with the Federal Power Commission which will fully compensate Power Company for the use of its system, to the extent that such arrangements can be accommodated from a functional engineering standpoint and to the extent that Power Company has surplus line capacity or reasonably available funds to finance new construction for this purpose. To the extent the entity or entities are able, they shall reciprocally provide transmission service to Power Company. Transmission service will be provided under this subparagraph for the delivery of power to an entity for its or its members' consumption and retail distribution or for casual resale to another entity for (1) its consumption or (2) its retail distribution. Nothing contained herein shall require the Power Company to transmit bulk power so as to have the effect of making the Tennessee Valley Authority ("TVA") or its distributors, directly or indirectly, a source of power supply outside the area determined by the TVA Board of Directors by resolution of May 16, 1966 to be the area for which the TVA or its distributors were the primary source of power supply on July 1, 1957, the date specified in the Revenue Bond Act of 1959, 16 USC 831 n-4.

(b) Power Company shall transmit over its system from any entity or entities with which it is interconnected, pursuant to rate schedules on file with the Federal Power Commission which will fully compensate Power Company for the use of its system, bulk power which results from any such entity having excess capacity available from self-owned generating resources in the State of Georgia, to the extent such excess necessarily results from economic unit sizing or from failure to forecast load accurately or from such generating resources becoming operational earlier than the planned in-service date, to the extent that such arrangements can be accommodated from a functional engineering standpoint, and to the extent Power Company has surplus line capacity available.

(6) Upon request, Power Company shall provide service to any entity purchasing partial requirements service, full requirements service or transmission service from Power Company at a delivery voltage appropriate for loads served by such entity, commensurate with Power Company's available transmission facilities. Sales of such service shall be made pursuant to rates on file with the Federal Power Commission or any successor regulatory agency, and subject to reasonable terms and conditions.

(7) Upon reasonable notice, Power Company shall grant any entity the opportunity to purchase an appropriate share in the ownership of, or, at the option of the entity, to purchase an appropriate share of unit power from each of the following nuclear generating units at Power Company's costs, to the extent the same are constructed and operated: Hatch 2, Vogtle 1, Vogtle 2, and any other nuclear generating unit constructed by Power Company in the State of Georgia which, in the application filed with USAEC or its successor agency, is scheduled for commercial operation prior to January 1, 1989.

An entity's request for a share must have regard for the economic size of such nuclear unit(s), for the entity's load size, growth and characteristics, and for demands upon Power Company's system from other entities and Power Company's retail customers, all in accordance with sound engineering practice. Executory agreements to accomplish the foregoing shall contain provisions reasonably specified by Power Company requiring the entity to consummate and pay for such purchase by an early date or dates certain. For purposes of this provision, "unit power" shall mean capacity and associated energy from a specified generating unit.

(8) Southern Nuclear shall not market or broker power or energy from Edwin I. Hatch Nuclear Plant, Unit 2. Georgia Power Company shall continue to be responsible for compliance with the obligations imposed on it in its antitrust license conditions. Georgia Power Company is responsible and accountable for the actions of Southern Nuclear, to the extent that Southern Nuclear's actions may, in any way, contravene the existing antitrust license conditions.

(9) To effect the foregoing conditions, the following steps shall be taken:

(a) Power Company shall file with the appropriate regulatory authorities and thereafter maintain in force as needed an appropriate transmission tariff available to any entity;

(b) Power Company shall file with the appropriate regulatory authorities and thereafter maintain in force as needed an appropriate partial requirements tariff available to any entity; Power Company shall have its liability limited to the partial requirements service actually contracted for and the entity shall be made responsible for the security of the bulk power supply resources acquired by the entity from sources other than the Power Company;

(c) Power Company shall amend the general terms and conditions of its current Federal Power Commission tariff and thereafter maintain in force as needed provisions to enable any entity to receive bulk power at transmission voltage at appropriate rates;

(d) Power Company shall not have the unilateral right to defeat the intended access by each entity to alternative sources of bulk power supply provided by the conditions to this license; but Power Company shall retain the right to seek regulatory approval of changes in its tariffs to the end that it be adequately compensated for services it provides, specifically including, but not limited to, the provisions of Section 205 of the Federal Power Act;

(e) Power Company shall use its best efforts to amend any outstanding contract to which it is a party that contains provisions which are inconsistent with the conditions of this license;

(f) Power Company affirms that no consents are or will become necessary from Power Company's parent, affiliates or subsidiaries to enable Power Company to carry out its obligations hereunder or to enable the entities to enjoy their rights hereunder;

(g) All provisions of these conditions shall be subject to and implemented in accordance with the laws of the United States and of the State of Georgia, as applicable, and with rules, regulations, and orders of agencies of both, as applicable.

3. This renewed license is effective as of the date of issuance and shall expire at midnight, June 13, 2038.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

Samuel J. Collins, Director
Office of Nuclear Reactor Regulation

Attachments:
Appendix A – Technical Specifications
Appendix B – Environmental Protection Plan

Date of Issuance: January 15, 2002

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;

c. Quality assurance for effluent and environmental monitoring;

d. DELETED

e. All programs and manuals specified in Specification 5.5.

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 NOS. 304 AND 249 TO RENEWED

FACILITY OPERATING LICENSE NOS. DPR-57 AND NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-321 AND 50-366

Table of Contents

1.0 INTRODUCTION ...- 1 -

1.1 Background ...- 1 -

1.2 Requested Licensing Action ...- 3 -

2.0 REGULATORY EVALUATION ...- 5 -

2.1 Other 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 ...- 21 -

2.5 Rescission of Exemptions ...- 21 -

2.6 Self-Approval Process for FPP Changes (Post-Transition) ...- 23 -

2.7 Modifications and Implementation Items ...- 25 -

3.0 TECHNICAL EVALUATION ...- 26 -

3.1 NFPA 805 Fundamental FPP and Minimum Design Requirements ...- 27 -

3.1.1 Compliance with NFPA 805, Chapter 3, Requirements ...- 28 -

3.1.1.1 Compliance Strategy - Complies ... - 29 -

3.1.1.2 Compliance Strategy – Complies with Clarification ... - 30 -

3.1.1.3 Compliance Strategy - Complies with Use of EEEEs ... - 30 -

3.1.1.4 Compliance Strategy - Complies by Previous NRC Approval ... - 31 -

3.1.1.5 Compliance Strategy – Submit for NRC Approval ... - 31 -

3.1.1.6 Compliance Strategy – Complies, With Required Action ... - 33 -

3.1.1.7 Compliance Strategy – Multiple Strategies ... - 34 -

3.1.1.8 Chapter 3 Sections Not Reviewed ... - 34 -

3.1.1.9 Compliance with Chapter 3 Requirements Conclusion ... - 34 -

3.1.2 Identification of the Power Block ...- 35 -

3.1.3 Plant-specific- Treatments or Technologies ...- 35 -

3.1.4 Approval Requests for Performance-Based Methods for NFPA 805, Chapter 3,Elements ...- 35 -

3.1.4.1 Approval Request 1: NFPA 805, Section 3.2.3(1) – Performance-Based Fire Protection Inspection, Testing, and Maintenance Frequencies ... - 36 -

3.1.4.2 Approval Request 2: NFPA 805, Sections 3.3.4 – Thermal Insulation Materials ... - 38 -

3.1.4.3 Approval Request 3: NFPA 805, Section 3.3.5.1 – Wiring Above Suspended Ceilings ... - 40 -

3.1.4.4 Approval Request 4: NFPA 805, Section 3.3.5.2 – Metal Conduits ... - 41 -

3.1.4.5 Approval Request 5: NFPA 805, Section 3.5.2 and 3.5.10 – Use of Check Valves in the Fire Water Tank Discharge Piping ... - 43 -

3.1.4.6 Approval Request 6: NFPA 805, Section 3.5.3 – Installation of Fire Pumps ... - 45 -

3.1.4.7 Approval Request 7: NFPA 805, Section 3.5.5 – Fire Pump Separation ... - 47 -

3.1.4.8 Approval Request 8: NFPA 805, Section 3.5.16 – Fire Protection Water Supply System ... - 48 -

3.1.4.9 Approval Request 9: NFPA 805, Section 3.6.1 – Installation of Standpipe and Hose Systems ... - 50 -

3.2 Nuclear Safety Capability Assessment Methods ... - 52 -

3.2.1 Compliance with NFPA 805 Nuclear Safety Capability Assessment Methods ... - 53 -

3.2.1.1 Attribute Alignment – Aligns with NEI 00-01 Sections ... - 54 -

3.2.1.2 Attribute Alignment − Aligns with Intent ... - 56 -

3.2.1.3 Attribute Alignment − Not in Alignment, but Prior NRC Approval ... - 58 -

3.2.1.4 Attribute Alignment − Not in Alignment, but No Adverse Consequences ... - 58 -

3.2.1.5 Attribute Alignment − Not in Alignment ... - 58 -

3.2.1.6 NFPA 805 − Nuclear Safety Capability Assessment Methods Conclusion ... - 58 -

3.2.2 Maintaining Fuel in a Safe and Stable Condition ... - 59 -

3.2.3 Applicability of Feed and Bleed ... - 61 -

3.2.4 Assessment of Multiple Spurious Operations ... - 61 -

3.2.5 Establishing Recovery Actions ... - 63 -

3.2.6 Plant-specific- Treatments or Technologies ... - 66 -

3.2.7 Conclusion for Section 3.2 ... - 66 -

3.3 Fire Modeling ... - 67 -

3.4 Fire Risk Assessments ... - 67 -

3.4.1 Maintaining Defense-in-Depth ... - 69 -

3.4.2 Safety Margins ... - 70 -

3.4.3 Quality of the Fire Probabilistic Risk Assessment ... - 71 -

3.4.3.1 Internal Events PRA Model ... - 71 -
3.4.3.2 Fire Probabilistic Risk Assessment Model ... - 73 -
3.4.3.3 Fire Modeling in Support of the Development of the Fire Risk Evaluations ... - 79 -
3.4.3.4 Fire Probabilistic Risk Assessment Model Quality During Self-Approval ... - 88 -
3.4.3.5 Conclusions Regarding Fire Probabilistic Risk Assessment Quality ... - 88 -
3.4.4 Fire Risk Evaluations ... - 89 -
3.4.5 Additional Risk Presented by Recovery Actions ... - 91 -
3.4.6 Cumulative Risk and Combined Changes ... - 92 -
3.4.7 Uncertainty and Sensitivity Analyses ... - 93 -
3.4.8 Conclusion for Section 3.4 ... - 94 -
3.5 Nuclear Safety Capability Assessment Results ... - 95 -
3.5.1 Nuclear Safety Capability Assessment Results by Fire Area ... - 95 -
3.5.1.1 Fire Detection and Suppression Systems Required to Meet the NSPC ... - 96 -
3.5.1.2 Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria ... - 96 -
3.5.1.3 Licensing Actions ... - 97 -
3.5.1.4 Existing Engineering Equivalency Evaluations ... - 97 -
3.5.1.5 Variances from Deterministic Requirements ... - 98 -
3.5.1.6 Recovery Actions ... - 98 -
3.5.1.7 Plant Fire Barriers and Separations ... - 99 -
3.5.1.8 Electrical Raceway Fire Barrier Systems ... - 99 -
3.5.1.9 Conclusion for Section 3.5.1 ... - 100 -
3.5.2 Clarification of Prior NRC Approvals ... - 101 -
3.5.3 Fire Protection During Non-Power Operational Modes ... - 101 -
3.5.3.1 NPO Strategy and Plant Operating States ... - 101 -
3.5.3.2 NPO Analysis Process ... - 101 -
3.5.3.3 NPO KSFs and SSC Used to Achieve Performance ... - 101 -
3.5.3.4 NPO Pinch Point Resolutions and Program Implementation ... - 103 -
3.5.4 Conclusion for Section 3.5 ... - 104 -
3.6 Radioactive Release Performance Criteria ... - 105 -
3.7 NFPA 805 Monitoring Program ... - 107 -
3.8 Post-Implementation Plant Change Evaluation Process ... - 108 -
3.9 Program Documentation, Configuration Control, and Quality Assurance ... - 111 -
3.9.1 Documentation ... - 111 -

3.9.2 Configuration Control ....- 111 -
3.9.3 Quality ....- 112 -
3.9.3.1 Independent Review .... - 112 -
3.9.3.2 Verification and Validation .... - 113 -
3.9.3.3 Limitations of Use .... - 114 -
3.9.3.4 Qualification of Users .... - 114 -
3.9.3.5 Uncertainty Analysis .... - 114 -
4.0 FIRE PROTECTION LICENSE CONDITION ....- 115 -
5.0 SUMMARY ....- 118 -
6.0 STATE CONSULTATION ....- 119 -
7.0 ENVIRONMENTAL CONSIDERATION ....- 120 -
8.0 CONCLUSION ....- 120 -
9.0 REFERENCES ....- 121 -

ATTACHMENTS

Attachment A: Abbreviations and Acronyms .... - A1 -

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 304 AND 249 TO RENEWED

FACILITY OPERATING LICENSE NOS. DPR-57 AND NPF-5

SOUTHERN NUCLEAR OPERATING COMPANY

EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-321 AND 50-366

1.0 INTRODUCTION

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC or the Commission) started developing fire protection requirements in the 1970s. In 1976, the NRC published comprehensive fire protection guidelines in the form of 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.

In 1980, to resolve issues identified in those reports, the NRC amended its regulations for fire protection in operating nuclear power plants (NPPs) and published its Final Rule, Fire Protection Program for Operating Nuclear Power Plants, in the *Federal Register* (FR) on November 19, 1980 (45 FR 76602), adding Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.48, "Fire Protection," and Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," to 10 CFR Part 50.

Section 50.48(a)(1) of 10 CFR requires each holder of an operating license and holders of a combined operating license issued under 10 CFR Part 52 to have a fire protection plan that satisfies General Design Criterion (GDC) 3 of Appendix A to 10 CFR Part 50 and states that the fire protection plan must describe the overall fire protection program (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 paragraph (a)(1), including administrative controls and personnel requirements for fire prevention and manual suppression activities; automatic and manual fire detection and suppression systems; and the means to limit fire damage to structures, systems, and components (SSC) to ensure the capability to safely shut down the plant. Section 50.48(a)(3) of 10 CFR requires that the licensee retain the fire

protection plan and each change to the plan as a record until the Commission terminates the license, and that the licensee retain each superseded revision of the procedures for 3 years.

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 but allows PB or deterministic approaches to be used to meet performance criteria.

NRC Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1 (Reference 4), states, in part, that:

> On March 26, 1998, the NRC staff sent to the Commission SECY-98-058, "Development of a Risk-informed, Performance-based Regulation for Fire Protection at Nuclear Power Plants" (Reference 5), in which it proposed to work with the NFPA and the industry to develop a risk-informed, performance-based consensus standard for nuclear power plant fire protection. This consensus standard could be endorsed in a future rulemaking as an alternative set of fire protection requirements to the existing regulations in 10 CFR 50.48. In SECY-00-0009, "Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking," dated January 13, 2000 (Reference 6), the NRC staff requested and received Commission approval to proceed with rulemaking to permit operating reactor licensees to adopt NFPA 805 as an alternative to existing fire protection requirements. On February 9, 2001, the NFPA Standards Council approved the 2001 Edition of NFPA 805 as an American National Standard for performance-based fire protection for light-water nuclear power plants.

A licensee that elects to adopt NFPA 805 must meet the performance goals, objectives, and criteria that are itemized in Chapter 1 of NFPA 805 through the implementation of PB or deterministic approaches. The goals include ensuring that reactivity control, inventory and pressure control, decay heat removal (DHR), vital auxiliaries, and process monitoring are achieved and maintained. The licensee then must establish plant fire protection requirements using the methodology in Chapter 2 of NFPA 805 such that the minimum FPP elements and design criteria contained in Chapter 3 of NFPA 805 are satisfied.

Next, the licensee identifies fire areas and fire hazards through a plant-wide analysis, and then applies either a PB or a deterministic approach to meet the performance criteria. As part of a PB approach, the licensee will use engineering evaluations, probabilistic safety assessments (PSAs)[1], and fire modeling (FM) calculations to show that the criteria are met. Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the performance criteria. It also specifies that at least one success path to

---

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.

achieve the nuclear safety performance criteria (NSPC) shall be maintained free of fire damage by a single fire.

RG 1.205 states that:

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

Throughout this SE, where the NRC staff states that the licensee's FPP element is in compliance 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).

RG 1.205 also states that:

> In parallel with the Commission's efforts to issue a rule incorporating the risk-informed, performance-based fire protection provisions of NFPA 805, NEI [Nuclear Energy Institute] published implementing guidance for the specific provisions of NFPA 805 and 10 CFR 50.48(c) in NEI 04-02 ["Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)," Revision 2 (Reference 7)].

RG 1.205 provides the NRC staff's regulatory position on NEI 04-02, Revision 2, and offers additional information and guidance to supplement the NEI document and assist licensees in meeting the NRC's regulations in 10 CFR 50.48(c) related to adopting a RI/PB FPP. RG 1.205 endorses the guidance of NEI 04-02, Revision 2, subject to certain exceptions, as providing methods acceptable to the NRC staff for adopting an FPP consistent with the 2001 Edition of NFPA 805 and 10 CFR 50.48(c).

Accordingly, Southern Nuclear Operating Company (SNC, the licensee) requested license amendments to allow it to establish and maintain the Edwin I. Hatch Nuclear Plant, (Hatch), Unit Nos. 1 and 2, FPP in accordance with 10 CFR 50.48(c) and change the Renewed Facility Operating Licenses.

1.2 Requested Licensing Action

By letter dated April 4, 2018 (Reference 8), as supplemented by letters dated May 28, 2019 (Reference 9), August 9, 2019 (Reference 10), October 7, 2019 (Reference 11), December 13, 2019 (Reference 12), February 5, 2020 (Reference 13), and March 13, 2020 (Reference 14), the licensee submitted an application for license amendments to transition the Hatch FPP from 10 CFR 50.48(a) and (b), to 10 CFR 50.48(c), NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants," 2001 Edition. The supplemental letters were in response to the NRC staff's requests for additional information (RAIs) dated March 29, 2019 (Reference 15) and August 8, 2019 (Reference 16). The licensee's supplemental letters dated May 28, 2019, August 9, 2019, October 7, 2019, December 13, 2019, February 5, 2020, and March 13, 2020, provided additional information that clarified the application, but did not expand the overall scope of the application as originally

noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 26, 2019 (84 FR 11340).

The April 4, 2018, 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 license amendment request (LAR)[2] attachments and reference to the complete versions and revisions[3] are as follows:

Attachment A:	NEI 04-02 Table B-1 Transition of Fundamental Fire Protection Program & Design Elements (Reference 8), (Reference 9), (Reference 10), (Reference 12)
Attachment B:	NEI 04-02 Table B-2 - Nuclear Safety Capability Assessment - Methodology Review (Reference 8)
Attachment C:	NEI 04-02 Table B-3 - Fire Area Transition (Reference 8), (Reference 9), (Reference 10), (Reference 12)
Attachment D:	NEI 04-02 Non-Power Operational Modes Transition (Reference 8)
Attachment E:	NEI 04-02 Radioactive Release Transition (Reference 8)
Attachment F:	Fire-Induced Multiple Spurious Operations Resolution (Reference 8)
Attachment G:	Recovery Actions Transition (Reference 8), (Reference 12)
Attachment H:	NFPA 805 Frequently Asked Question Summary Table (Reference 8), (Reference 12)
Attachment I:	Definition of Power Block (Reference 8)
Attachment J:	Fire Modeling V&V (Reference 8), (Reference 9), (Reference 10), (Reference 12)
Attachment K:	Existing Licensing Action Transition (Reference 8)
Attachment L:	NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii)) (Reference 8), (Reference 12)
Attachment M:	License Condition Changes (Reference 8), (Reference 12)
Attachment N:	Technical Specification Changes (Reference 8)
Attachment O:	Orders and Exemptions (Reference 8)
Attachment P:	RI-PB Alternatives to NFPA 805 10 CFR 50.48(c)(4) (Reference 8)
Attachment Q:	No Significant Hazards Evaluations (Reference 8)
Attachment R:	Environmental Considerations Evaluation (Reference 8)
Attachment S:	Modifications and Implementation Items (Reference 8), (Reference 9), (Reference 10), (Reference 12)
Attachment T:	Clarification of Prior NRC Approvals (Reference 8)
Attachment U:	Internal Events PRA Quality (Reference 8)
Attachment V:	Fire PRA Quality (Reference 8)
Attachment W:	Fire PRA Insights (Reference 8), (Reference 12)

The existing deterministic fire protection licensing basis for Hatch, which implements the fire protection requirements of 10 CFR 50.48 and 10 CFR Part 50, Appendix R, was approved by the NRC on May 11, 1973, May 9, 1974, June 13, 1978, October 4, 1978, and February 11, 1983 and is referenced in the Hatch Updated Final Safety Analysis Report (UFSAR), and described in the updated Fire Hazards Analysis and Fire Protection Program dated

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 identified here and elsewhere in the SE where applicable.

July 22, 1986. The proposed amendments 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 Hatch 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 Hatch 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 Hatch. 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 modifications in LAR Table S-2, as supplemented, and implementation items in LAR Table S-3,[4] as supplemented.

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 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 (AEC) published for public comment in the Federal Register (32 FR 10213), a revised and expanded set of 70 draft GDC. On February 20, 1971, the AEC 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.

The construction permits of Hatch, Unit 1, and Hatch, Unit 2, were issued on September 30, 1969, and on December 27, 1972, respectively. Consequently, Hatch, Unit 2, is licensed in

---

4 LAR Attachment S, Tables S-1, S-2, and S-3 are referred to as Table S-1, Table S-2, and Table S-3 in this SE. The licensee supplemented its initial LAR Attachment S with an Attachment S2 to its December 13, 2019 submittal that contained the final versions of Tables S-1, S-2, and S-3.

conformance with 10 CFR Part 50, Appendix A, “General Design Criteria.”
Hatch, Unit 1, is licensed in conformance with the 1967 version of 10 CFR Part 50, Appendix A, “General Design Criteria for Nuclear Power Plant Construction Permits” (ADAMS Accession No. ML043310029). Hatch, Unit 1, Final Safety Analysis Report (FSAR), Appendix F, “Conformance To Atomic Energy Commission Criteria,” describes the relevant licensing bases for Hatch, Unit 1. The operating license for Hatch, Unit 1, was issued in 1974, and the operating license for Hatch, Unit 2 was issued in 1978.

Section 3.0, “Design of Structures, Components, Equipment, and Systems,” of the NRC safety evaluation report related to the operation of Hatch Unit 1, describes the NRC staff’s evaluation of the facility’s conformance with the GDC for the original facility operating license. Hatch Unit 1 was designed and constructed in accordance with the GDC issued for comment in July of 1967. The NRC safety evaluation report concluded that there was reasonable assurance that the plant met the intent of the GDC published in the FR on May 21, 1971.

Section 3.0, “Design of Structures, Components, Equipment, and Systems,” of the NRC safety evaluation report (NUREG-0411) related to the operation of Hatch Unit 2, describes the NRC staff’s evaluation of the facility’s conformance with the GDC for the original facility operating license. Hatch Unit 2 was designed and constructed in accordance with the amended GDC dated July 7, 1971 and the NRC safety evaluation report concluded that the plant design conformed to the amended criteria.

The regulations in 10 CFR 50.48 establish requirements for nuclear power plant (NPP) 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 FPP and Design Elements,” must submit a 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 Hatch included several requests under 10 CFR 50.48(c)(2)(vii), which are discussed in LAR Attachment L, as supplemented, but the LAR for Hatch 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 4). 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 Other 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 NPPs. 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 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.

The regulations in 10 CFR Part 20, “Standards for Protection Against Radiation,” establishes radiation protection limits including the definition of “as low as reasonably achievable,” also known as “ALARA.”

2.2 Applicable Guidance

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

RG 1.205, Revision 1, issued December 2009 (Reference 4), provides guidance for use in complying with the requirements that the NRC has promulgated for RI/PB FPPs that comply with 10 CFR 50.48 and the referenced 2001 Edition of the NFPA standard. It endorses portions of NEI 04-02, Revision 2 (Reference 7), where it has been found to provide methods acceptable to the NRC for implementing NFPA 805 and complying with 10 CFR 50.48(c). The regulatory positions in Section C of RG 1.205 include clarification of the guidance provided in NEI 04-02, as well as NRC exceptions to the guidance. RG 1.205 sets forth regulatory positions, emphasizes certain issues, clarifies the requirements of 10 CFR 50.48(c) and NFPA 805, clarifies the guidance in NEI 04-02, and modifies the NEI 04-02 guidance where required. Should a conflict occur between NEI 04-02 and this RG, the regulatory positions in RG 1.205 govern. This RG also indicates that Chapter 3 of NEI 00-01, “Guidance for Post-Fire NFPA 805 Safe Shutdown Circuit Analysis,” Revision 2, issued May 2009, (Reference 17), 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).

NEI 04-02, “Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)” (Reference 7), provides guidance for implementing the requirements of 10 CFR 50.48(c) and represents methods for implementing in whole or in part an RI/PB FPP. This implementing guidance for NFPA 805 has two primary purposes: (1) provide direction and clarification for adopting NFPA 805 as an acceptable approach to fire protection, consistent with 10 CFR 50.48(c), and (2) provide additional, supplemental technical guidance and methods for using NFPA 805 and its appendices to demonstrate compliance with

fire protection requirements. Although there is a significant amount of detail in NFPA 805 and its appendices, clarification and additional guidance for select issues help ensure consistency and effective utilization of the standard. The NEI 04-02 guidance focuses attention on the RI/PB fire protection goals, objectives, and performance criteria contained in the 02 guidance and focuses attention on the RI/PB fire protection goals, objectives, and performance criteria contained in NFPA 805 and the RI/PB tools considered acceptable for demonstrating compliance. Revision 2 of NEI 04-02 incorporates guidance from RG 1.205 and approved Frequently Asked Questions (FAQs).

NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 2 (Reference 17), provides a deterministic methodology for performing 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 (MSO). The RI method is intended for application by licensees to determine the risk significance of identified circuit failure issues related to MSO.

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 18), describes a method the NRC staff considers acceptable for implementing specific parts of the NRC regulations. The guidance does not preclude other approaches for requesting licensing basis changes. Rather, RG 1.174 is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. As such, the RG provides general guidance concerning one approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's licensing basis and for assessing the impact of such proposed changes on the risk associated with plant design and operation.

RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009 (Reference 19), provides guidance to licensees for use in determining the technical adequacy of the base probabilistic risk assessment (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.

However, 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.

American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-Sa-2009, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 20), provides guidance PRAs used to support RI decisions for commercial light water reactor NPPs and prescribes a method for applying these requirements for specific applications. The

standard gives guidance for a Level 1 PRA of internal and external hazards for all plant operating modes. In addition, the standard provides guidance for a limited Level 2 PRA sufficient to evaluate large early release frequency (LERF). The only hazards excluded explicitly from the scope are accidents resulting from purposeful human-induced security threats (e.g., sabotage). The standard applies to PRAs used to support applications of RI decision-making related to design, licensing, procurement, construction, operation, and maintenance.

RG 1.189, "Fire Protection for Nuclear Power Plants," Revision 2, issued October 2009 (Reference 21), provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP. The NRC staff developed the RG to provide a comprehensive fire protection guidance document and to identify the scope and depth of fire protection that the NRC staff would consider acceptable for NPPs.

NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (Reference 22), 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 23), provides the NRC staff with guidance for evaluating the technical adequacy of a licensee's PRA results when used to request RI changes to the licensing basis.

NUREG-0800, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, issued June 2007 (Reference 24), 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 Fire PRA (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 25), and 2 (Reference 26), and Supplement 1 (Reference 27), 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.

Memorandum from Richard P. Correia, RES, to Joseph G. Giitter, Office of Nuclear Reactor Regulation (NRR) titled, "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," dated June 14, 2013 (Reference 28), notes that based on new experimental information documented in NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," issued April 2008 (Reference 29), and NUREG/CR- 7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," issued April 2012 (Reference 30), the reduction in hot short probabilities for circuits provided with control power transformers identified in NUREG/CR-6850 cannot be repeated in experiments, and therefore, may be too high and should be reduced.

NUREG-1805, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 31), provides quantitative methods known as "Fire Dynamics Tools (FDTs)" to assist regional fire protection inspectors in performing fire hazard analysis. The FDTs are intended to assist fire protection inspectors in performing RI evaluations of credible fires that may cause critical damage to essential safe shutdown (SSD) equipment, as required by the new reactor oversight process defined in the NRC's inspection manual.

NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1 through 7 (Reference 32), and Supplement 1 (Reference 33), provides technical documentation regarding the predictive capabilities of a specific set of fire models for the analysis of fire hazards in NPP scenarios. This report is the result of a collaborative program with EPRI and the National Institute of Standards and Technology (NIST). The selected models are:

1. FDTs developed by NRC (Volume 3),

2. Fire-Induced Vulnerability Evaluation Methodology-Rev. 1 developed by EPRI (Volume 4),

3. The zone model Consolidated Model of Fire and Smoke Transport (CFAST) developed by NIST (Volume 5),

4. The zone model MAGIC developed by Électricité de France (Volume 6), and

5. The computational fluid dynamics model fire dynamics simulator developed by NIST (Volume 7).

In addition to the fire model volumes, Volume 1 is the comprehensive main report and Volume 2 is a description of the experiments and associated experimental uncertainty used in developing this report. Supplement 1 evaluated the latest versions of the five fire models at the time, which included additional test data for validation of the models.

NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," Volume 1 (Reference 34), describes Phase 1 of the CHRISTIFIRE testing program conducted by NIST. The overall goal of this multi-year program is to quantify the burning characteristics of grouped electrical cables installed in cable trays. This first phase of the program focuses on horizontal tray configurations. CHRISTIFIRE addresses the burning behavior of a cable in a fire beyond the point of electrical failure. The data obtained from this project can be used for the development of fire models to calculate the heat release rate (HRR) and flame spread of a cable fire.

NUREG-1855, Volume 1, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making” (Reference 35), provides guidance on how to treat uncertainties associated with PRA in RI decision making. The objectives of this guidance include fostering an understanding of the uncertainties associated with PRA and their impact on the results of PRA, and providing a pragmatic approach to addressing these uncertainties in the context of the decision making.

NUREG-1921, “EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report” (Reference 36), which presents the state-of-the-art in fire HRA practice. This report was developed jointly between RES and EPRI to develop the methodology and supporting guidelines for estimating human error probabilities (HEPs) for human failure events (HFEs) following the fire-induced initiating events of an FPRA. The report builds on existing HRA methods and is intended primarily for practitioners conducting a fire HRA to support an FPRA.

NUREG-1934, “Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)” (Reference 37), describes the implications of the verification and validation (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 NPP fire hazard analyses. The report also provides information to assist fire model users in applying this technology in the NPP environment.

Generic Letter (GL) 2006-03, “Potentially Nonconforming Hemyc and MT Fire Barrier Configurations” (Reference 38), which requested that licensees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements in light of the information provided in this GL and, if appropriate, take additional actions.

NFPA 101, “Life Safety Code” (Reference 39), provides the minimum requirements for egress; features of fire protection, sprinkler systems, alarms, emergency lighting, smoke barriers; and special hazard protection.

NFPA 30, “Flammable and Combustible Liquids Code” (Reference 40), provides requirements for the safe storage, handling, and use of flammable and combustible liquids.

NFPA 51B, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work” (Reference 41), provides requirements for preventing injury, loss of life, and loss of property from fire or explosion as a result of hot work projects such as welding, heat treating, grinding, and similar applications producing or using sparks, flames, or heat.

NFPA 72, “National Fire Alarm and Signaling Code” (Reference 42), 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 and emergency communications systems, and their components.

NFPA 76, “Standard for the Fire Protection of Telecommunications Facilities” (Reference 43), provides requirements for fire protection of telecommunications facilities providing telephone, data, internet transmission, wireless, and video services, as well as life safety for the occupants, plus protection of equipment and service continuity.

NFPA 241, “Standard for Safeguarding Construction, Alteration, and Demolition Operations” (Reference 44), provides requirements for preventing or minimizing fire damage to structures, including those in underground locations, during construction, alteration, or demolition.

NFPA 262, “Standard Method of Test for Flame Travel and Smoke of Wires and Cables for Use in Air-Handling Spaces” (Reference 45), provides a test procedure to evaluate the potential for smoke and fire spread along cables and wires housed in a plenum or other air transport spaces.

2.3 NFPA 805 Frequently Asked Questions

In the LAR, the licensee proposed to use a number of documents commonly known as NFPA 805 FAQs. The following table provides the set of FAQs the licensee used that the NRC staff referenced in the preparation of this SE, as well as the SE sections to which each FAQ is referenced.

Table 2.3-1: NFPA 805 Frequently Asked Questions

FAQ #	FAQ Title and Summary	Reference	SE Section
07-0030	“Establishing Recovery Actions” • This FAQ provides an acceptable process for determining the recovery actions (RAs) for NFPA 805, Chapter 4 compliance. The process includes: ▪ Differentiation between RAs and activities in the main control room (MCR) or at primary control station(s) (PCS). ▪ Determination of which RAs are required by the NFPA 805 FPP. ▪ Evaluate the additional risk presented by the use of RAs. ▪ Evaluate the feasibility of the identified RAs. ▪ Evaluate the reliability of the identified RAs.	(Reference 46)	3.2.5 3.4.5 3.5.1.6
07-0038	“Lessons Learned on Multiple Spurious Operations (MSOs)” • This FAQ reflects an acceptable process for the treatment of MSOs during transition to NFPA 805: ▪ Step 1 – Identify potential MSO combinations of concern. ▪ Step 2 – Expert panel assesses plant-specific- vulnerabilities and reviews MSOs of concern. ▪ Step 3 – Update the 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 47)	3.2.4 3.2.7

FAQ #	FAQ Title and Summary	Reference	SE Section
07-0039	"Incorporation of Pilot Plant Lessons Learned – Table B-2-" • This FAQ provides additional detail for the comparison of the licensee's safe shutdown (SSD) strategy to the endorsed industry guidance, NEI 00-01 "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 1 (Reference 48). In short, the process has the licensees: ▪ Assemble industry and plant-specific- documentation; ▪ Determine which sections of the guidance are applicable; ▪ Compare the existing SSD methodology to the applicable guidance; and ▪ Document any discrepancies.	(Reference 49)	3.2.1
07-0040	"Non-Power Operations (NPO) Clarifications" • This FAQ clarifies an acceptable NFPA 805 NPO program. The process includes: ▪ Selecting NPOs equipment and cabling. ▪ Evaluation of NPOs Higher Risk Evolutions (HREs). ▪ Analyzing NPO Key Safety Functions (KSFs). ▪ Identifying plant areas to protect or "pinch points" during NPOs HREs and actions to be taken if KSFs are lost.	(Reference 50)	3.5.3 3.5.4
08-0054	"Compliance with Chapter 4 of NFPA 805" • This FAQ provides an acceptable process to demonstrate Chapter 4 compliance for transition: ▪ Step 1 – Assemble documentation ▪ Step 2 – Document Fulfillment of NSPC ▪ Step 3 – Variance From Deterministic Requirements (VFDR) Identification, Characterization, and Resolution Considerations ▪ Step 4 – Performance-Based Evaluations ▪ Step 5 – Final VFDR Evaluation ▪ Step 6 – Document Required Fire Protection Systems and Features	(Reference 51)	3.4.4 3.5.1.4
09-0056	"Radioactive Release Transition" • This FAQ provides an acceptable level of detail and content for the radioactive release Section of the LAR. It includes: ▪ Justification of the compartmentation, if the radioactive release review is not performed on a fire area basis. ▪ Pre-fire plan and fire brigade training review results. ▪ Results from the review of engineering controls for gaseous and liquid effluents.	(Reference 52)	3.5

FAQ #	FAQ Title and Summary	Reference	SE Section
10-0059	"Monitoring Program" • This FAQ provides clarification regarding the implementation of an NFPA 805 monitoring program for transition. It includes: ▪ Monitoring program analysis units; ▪ Screening of low safety significant SSC; ▪ Action level thresholds; and ▪ The use of existing monitoring programs.	(Reference 53)	3.7
12-0064	"Hot Work/Transient Fire Frequency Influence Factors" • This FAQ clarifies and updates the treatment of hot work and transient fire frequency influence factors. The updated treatment involves the use of sensitivity studies when the updated influence factors are used.	(Reference 54)	3.4.3.2
13-0004	"Clarifications on Treatment of Sensitive Electronics" • 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 components.	(Reference 55)	3.4.3.2
13-0005	"Cable Fires Special Cases: Self-Ignited and Caused by Welding and Cutting" • This FAQ provides additional guidance for detailed FPRA/FM concerning self-ignited cable fires and cable fires caused by welding and cutting.	(Reference 56)	3.1.4.3
14-0008	"Main Control Board Treatment" • This FAQ clarifies main control board (MCB) definition and gives guidance on application of the frequencies in Appendix L to NUREG/CR-6850.	(Reference 57)	3.4.3.2

2.4 Orders, License Conditions, and Technical Specifications

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

2.4.1 Orders

The NRC staff reviewed LAR Section 5.2.3, "Orders and Exemptions," and LAR Attachment O, regarding NRC-issued orders pertinent to Hatch that are being revised or superseded by the NFPA 805 transition process. The LAR stated that the licensee conducted a review of its docketed correspondence to determine if there were any orders or exemptions that needed to be superseded or revised. The LAR also stated that the licensee conducted a review to ensure that compliance with the physical protection requirements, security orders, and adherence to those commitments applicable to Hatch are maintained. The licensee discussed the affected orders and exemptions in LAR Attachment O.

The licensee requested that 18 exemptions be rescinded for Hatch. The licensee also determined that no orders need to be superseded or revised to implement an FPP that complies with 10 CFR 50.48(c).

Based on its review, the NRC staff accepts the licensee's determination that 18 exemptions should be rescinded for Hatch, and that no orders need to be superseded or revised to implement NFPA 805 at Hatch. (See SE Section 2.5 for the NRC staff's detailed evaluation of the exemptions being rescinded.)

The licensee also performed a specific review of the license amendments that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 (subsequently incorporated into 10 CFR 50.54(hh)(2)) to ensure that any changes being made in order to comply with 10 CFR 50.48(c) do not invalidate existing commitments applicable to Hatch. The licensee's review of the order confirmed that changes to the FPP during transition to NFPA 805 will not affect the measures required by Section B.5.b of Commission Order EA-02-026 (10 CFR 50.54(hh)(2)). The licensee will continue to have strategies that address large fires and explosions including a firefighting response strategy, operations to mitigate fuel damage, and actions to minimize release upon transition to NFPA 805. The NRC staff concludes that the licensee's determination regarding Commission Order EA-02-026 (10 CFR 50.54(hh)(2)) is acceptable.

2.4.2 License Conditions

In Section 5.2.1, "License Condition Changes," of the LAR dated April 4, 2018, it states that the current Hatch fire protection license condition 2.C(3) is to be replaced by the following proposed license condition in LAR Attachment M:

> (3) <u>Fire Protection</u>
>
> Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in the licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.
>
> (a) <u>Risk-Informed Changes that May Be Made Without Prior NRC Approval</u>
>
> > A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and as-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.

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

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

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

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

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

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

- 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)
- 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 June 11, 2020, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) 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 (b)(2) above.

(2) The licensee shall implement the modifications described in Attachment S2, Table S-2, "Plant Modifications Committed," of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after issuance of the NRC Safety Evaluation (SE). The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) The licensee shall implement the items as listed in Attachment S2, Table S-3, "Implementation Items," of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

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 modifications and implementation items. SE Section 4.0 provides the NRC staff's review of the proposed license condition.

The new Fire Protection license condition will affect the remaining RFOL page numbers; these are editorial changes and do not affect the technical content of the license conditions.

2.4.3 Technical Specifications

Section 5.2.2, of the LAR, "Technical Specifications," and Attachment N of the LAR, state that the licensee conducted a review of the Hatch 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 Guide 1.33, Revision 2, Appendix A, February 1978;
> b. The emergency operating procedures required to implement the requirements of NUREG-0737 and NUREG-0737, Supplement 1;
> c. Quality assurance for effluent and environmental monitoring;
> ~~d. Fire Protection Program implementation; and~~ **DELETED**
> e. All programs and manuals 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 Hatch 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 Hatch 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 Hatch fire protection license conditions 2.C(3). Section 2.2 of the LAR, "NRC Acceptance of the Fire Protection Licensing Basis," identifies exemptions from 10 CFR Part 50, Appendix R, for Hatch, and provides a summary of each exemption (identified as Licensing Actions 1–21 in LAR Attachment K). These exemptions were granted by the NRC by letters dated November 16, 1981, April 18, 1984, January 2, 1987, and March 24, 1987.

In Attachment O of the LAR, the licensee stated that the current Hatch 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 Hatch 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 Attachment K of the LAR.

For the following exemptions, in Attachment K of the LAR, the licensee indicates that the underlying condition has been evaluated using RI/PB methods and found to be acceptable with no further actions, because the philosophy of DID and sufficient safety margins are maintained; or has been evaluated and found to be adequate for the hazard or functionally equivalent; or has been evaluated and found to be not required by NFPA 805.

- Licensing Action 1: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.3, for lack of an area-wide automatic fire suppression system in the control room (CR).

- Licensing Action 2: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2. for lack of automatic fire suppression systems and 3-hour fire rated barriers in the Unit 1 4160 volt transformer room and the Unit 1 west 600 volt switchgear room.

- Licensing Action 3: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of 3-hour fire rated barriers in the Unit 1 and Unit 2 switchgear rooms, Unit 2 transformer room, and control building working floor.

- Licensing Action 4: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of a 3-hour fire rated barrier or 1-hour barrier and area-wide automatic fire detection and suppression in the Unit 1 and Unit 2 reactor buildings.

- Licensing Action 5: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of an area-wide automatic fire suppression system in the Unit 2 control building health physics area.

- Licensing Action 6: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of 3-hour fire rated barriers in the Unit 2 control building switchgear hallway.

- Licensing Action 7: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of a 3-hour fire rated barrier in the station battery rooms.

- Licensing Action 8: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of a 3-hour fire rated barrier in the Unit 2 turbine building condenser bay.

- Licensing Action 9: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of 3-hour fire rated barriers in the turbine buildings, east cableway, and west cableway.

- Licensing Action 10: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of an area-wide suppression system in diesel building switchgear room 2G.

- Licensing Action 11: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of 3-hour fire rated barriers or 1-hour barriers and area-wide automatic fire detection and suppression in the control building common corridor.

- Licensing Action 12: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of an area-wide automatic fire suppression system in the intake structure.

- Licensing Action 13: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2, for lack of a 3-hour fire rated barrier in the control building east corridor.

- Licensing Action 14: Deviations from the requirements of NFPA Standards Nos. 13, 14, and 15 with respect to hanger selection and spacing.

- Licensing Action 15: Deviations from the criteria of NFPA Standards Nos. 13, 14, and 15 with respect to component selection.

- Licensing Action 16: Deviations from the criteria of NFPA Standard Nos. 13, 15, and 72E with respect to sprinkler head/nozzle and detector placement.

- Licensing Action 17: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.J, for lack of emergency lighting in access and egress routes.

- Licensing Action 18: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2.c, for lack of 1-hour fire rated barriers in the Unit 1 and Unit 2 Reactor Buildings.

- Licensing Action 19: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.1.a, for residual heat removal (RHR) and reactor core isolation cooling (RCIC) system pump repairs.

- Licensing Action 20: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2.a, and III.G.2.b for lack of 1-hour fire rated barriers in the Unit 1 and 2 reactor buildings.

- Licensing Action 21: Exemption from the requirements of 10 CFR 50 Appendix R, Section III.G.2.b, for lack of 20 feet of redundant cable separation in the intake structure.

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

2.6 Self-Approval Process for FPP Changes (Post-Transition)

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 (PCEs).

Section 2.2.9 of NFPA-805, "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.

Section 2.4.4 of NFPA 805, "Plant Change Evaluation," states, in part, that:

> A plant change evaluation [PCE] 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 Hatch FPP if the requirements of NFPA 805 regarding engineering analyses, fire risk evaluations (FREs), and PCEs are met. The licensee intends to use an FPRA to evaluate the risk of proposed future plant changes at Hatch. Risk assessments for PCEs 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 core damage frequency (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 PCE 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. Section 2.4 of NFPA 805, 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 plant modifications and implementation items listed in LAR Table S-2, as supplemented, and Table S-3, as supplemented, are completed, the risk analysis will not be consistent with the as-built, as-operated and as-maintained plant. Therefore, for changes made during the transition period, the proposed license condition will require the licensee to use its PCE 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, 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 December 13, 2019, (Reference 12), the licensee provided the final version of LAR Attachment S (Attachment S2), as supplemented, and a revised schedule. Table S-1, as supplemented, identifies no completed plant modifications, Table S-2, as supplemented, identifies 14 incomplete modifications that are necessary to bring Hatch into compliance with NFPA 805. Table S-3, as supplemented, provides the list of implementation items, which are actions that the licensee will complete during implementation of the license amendments to transition to NFPA 805 (e.g., procedure changes or development of NFPA 805 programs). The licensee proposed to complete the plant modifications in LAR Table S-2, as supplemented, by the startup of the second refueling outage for each unit after issuance of the SE, and proposed to complete the implementation items in LAR Table S-3, as supplemented, within 365 days after NRC approval with the exception of implementation item IMP-19 which the licensee proposed to complete within 180 days after all modifications for the respective unit are operable. The licensee also stated that appropriate compensatory measures will be maintained until the plant modifications are completed.

The NRC staff confirmed that the modifications identified in LAR Table S-2, as supplemented, are consistent with the modifications listed in LAR Attachment C, as supplemented. The LAR did not identify any completed modifications credited for the transition to NFPA 805. The NRC staff also confirmed that the implementation items identified in LAR Table S-3, as supplemented, 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 modifications 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 modifications in LAR Table S-2, as supplemented, and the implementation items in LAR Table S-3, as supplemented, were credited by the licensee in its analysis supporting the LAR and must be completed for Hatch 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 (Reference 3), in accordance with 10 CFR 50.48(c). The NRC staff used the guidance provided in NUREG 0800, Section 9.5.1.2, (Reference 22), 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 FPP 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 NSPC in NFPA 805, Section 1.5.1.

Section 3.3 provides the NRC staff's evaluation of the 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 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 PCE process.

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

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 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 Hatch 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 approach described in NEI 04-02, Revision 2 (Reference 7), as endorsed by the NRC in RG 1.205, Revision 1, (Reference 4), to assess the proposed 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 FPP and provided specific compliance statements for each Chapter 3 attribute that contained applicable requirements. As discussed below, some subsections of NFPA 805, Chapter 3, do not contain requirements or are otherwise not applicable, and others are provided with multiple compliance statements to fully document compliance with the element.

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 in LAR Attachment A, "NEI 04-02 Table B-1, Transition of Fundamental FPP and Design Elements" (also called the B-1 Table), as "Complies."

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

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

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

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

6. The existing FPP element does not comply with the requirement but will be in direct compliance with the completion of a required action: noted in LAR Attachment A, Table B-1- as "Complies, with Required Action." These outstanding actions are identified in LAR Table S-2, as supplemented, or Table S-3, as supplemented, as modifications or implementation items.

Compliance approach 6, “Complies, with Required Action,” is a change from the NEI 04-02 based approach in that it is a new category not included in NEI 04-02. The intent of this method for achieving compliance is to identify FPP elements that will comply after completion of an action by the licensee. The required actions are identified in LAR Table S-2, as supplemented, or Table S-3, as supplemented, as modifications or implementation items and would be required by the proposed license condition.

The NRC staff concludes that compliance approach 6 is acceptable because the actions for achieving compliance will incorporate the provisions of NFPA 805 in the FPP and would be required by the proposed license condition.

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 changes. 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 Section 4.2.2 of the LAR, “Existing Engineering Equivalency Evaluation Transition”, 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 Hatch 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 Section 4.2.3 of the LAR, “Licensing Action Transition”, that the bases for licensing actions approved previously 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.

Attachment A to the LAR, 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 Strategy - Complies

For the majority of the NFPA 805, Chapter 3, requirements, as modified by 10 CFR 50.48(c)(2), the licensee determined that the RI/PB FPP complies directly with the fundamental FPP element using the existing FPP element. In these instances, based on the validity of the licensee’s statements, the NRC staff concludes that the licensee’s statements of compliance are acceptable. A more detailed review of the licensee’s compliance with Section 3.3.5.2 of NFPA 805 is provided below

.

Section 3.3.5.2 of NFPA 805

The licensee identified the compliance strategy for NFPA 805, Section 3.3.5.2, "Electrical Raceway Construction Limits" in LAR Attachment A, Table B-1, as “Complies.” In FPE RAI 02, (Reference 15), the NRC staff requested that the licensee describe whether there were any non-embedded, non-metallic conduits installed at the plant. In addition, the NRC staff requested that the licensee discuss whether approval was being requested for future installations or if future installations would be installed in accordance with the requirements of NFPA 805. In its response to FPE RAI 02 (Reference 9), the licensee stated that its design criteria do not allow for non-embedded, non-metallic conduits to be installed within the power block areas of the plant. In addition, the licensee also stated that it revised its approval request to address existing and future installation of flexible metallic and polyvinyl chloride (PVC) coated flexible metallic conduits. The NRC staff concludes that the licensee’s response to the RAI is acceptable because the licensee demonstrated that non-embedded, non-metallic conduits are not installed in the plant, and that existing and future installations of flexible metallic and PVC coated flexible metallic conduits would be installed in accordance with NFPA 805.

3.1.1.2 Compliance Strategy – Complies with Clarification

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

3.1.1.3 Compliance Strategy - Complies with Use of EEEEs

In several NFPA 805, Chapter 3, requirements, the licensee demonstrated compliance with the fundamental FPP element through the use of EEEEs. The NRC staff reviewed the licensee’s statement of continued validity for the EEEEs, identified implementation items and the statement on the quality and appropriateness of the evaluations, and concluded that the licensee’s statements of compliance in these instances are acceptable.

A more detailed review of the licensee’s compliance with Sections 3.3.7.1, 3.3.8, and 3.4.1(a)(1) of NFPA 805 is provided below.

Sections 3.3.7.1, 3.3.8, and 3.4.1(a)(1) of NFPA 805

Section 3.3.7.1 of NFPA 805, requires that storage of flammable gas 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 and also that NFPA 50A, “Standard for Gaseous Hydrogen Systems at Consumer Sites”, be followed for hydrogen storage.

Section 3.3.8 of NFPA 805, requires that bulk storage of flammable and combustible liquids shall not be permitted inside structures containing systems, equipment, or components important to nuclear safety, and as a minimum, storage and use shall comply with NFPA 30, “Flammable and Combustible Liquids Code.”

Section 3.4.1(a)1 of NFPA 805, requires that on-site fire-fighting capability consist of a fully staffed, trained, and equipped fire-fighting force that is available at all times to control and extinguish all fires on site, and that this force shall have a minimum complement of five persons

on duty and shall conform with NFPA 600, Standard on Industrial Fire Brigades (interior structural firefighting).

In Attachment A, Table B-1 of the LAR, the licensee described the compliance strategy for NFPA 805 Sections 3.3.7.1, 3.3.8, and 3.4.1(a)(1) but did not describe whether or how non-compliances were resolved.

In its response to FPE RAI 04 (Reference 9), the licensee stated that it addressed instances of identified non-compliances one of three ways: (1) maintenance type items were entered into the site's Corrective Action Program (CAP) and prioritized based on significance; (2) non-maintenance-type deviations were added to LAR Table S-2, as supplemented, or Table S-3, as supplemented; or (3) in some instances, by writing an engineering evaluation justifying the non-compliance as acceptable. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that it addressed identified non-compliances.

3.1.1.4 Compliance Strategy - Complies by Previous NRC Approval

Section 3.1 of NFPA 805, 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 NRC SE dated October 4, 1978 (Reference 58).

In each instance, the licensee evaluated the basis for the original NRC approval and determined that in all cases, the bases remain valid.

The NRC staff reviewed the information provided by the licensee and concludes that the licensee has demonstrated previous NRC approval using suitable documentation. The NRC staff also concludes that 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.

The licensee identified no licensing actions that required clarification of NFPA 805, Chapter 3, requirements.

3.1.1.5 Compliance Strategy – Submit for NRC Approval

The licensee requested approval for the use of PB methods to demonstrate compliance with fundamental FPP elements. In accordance with 10 CFR 50.48(c)(2)(vii), the licensee requested specific approvals be included in the license amendments approving the transition to NFPA 805. The NFPA 805 sections identified in LAR Attachment A, Table B-1 as complying by this method are as follows:

- Section 3.2.3(1), which requires establishing procedures for implementation of the FPP, including inspection, testing, and maintenance of fire protection systems and features credited by the FPP. The licensee requested NRC approval for the use of a PB method to establish inspection, testing, and maintenance frequencies for fire protection systems and features, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.1)

- Section 3.3.4, which concerns the use of thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials. The licensee requested NRC approval for the use of a PB method to justify the installation of some of these materials, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.2)

- Section 3.3.5.1, which concerns the installation of wiring above suspended ceilings. The licensee requested NRC approval for the use of a PB method to justify the installation of certain cables above suspended ceilings, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.3)

- Section 3.3.5.2, which concerns the installation of metal tray and metal conduit for electrical raceways. The licensee requested NRC approval for the use of a PB method to justify the installation of embedded PVC conduits and PVC coated flexible metallic conduits in lengths up to 6 feet between equipment and rigid conduits, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.4)

- Section 3.5.2 and 3.5.10, which concern the use of check valves in the fire protection water storage tanks discharge piping. The licensee requested NRC approval for the use of a PB method to justify the lack of check valves in the fire protection water storage tanks discharge piping, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.5)

- Section 3.5.3, which concerns the installation of fire pumps. The licensee requested NRC approval for the use of a PB method to justify the installation of its fire pumps, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.6)

- Section 3.5.5, which concerns the 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. The licensee requested NRC approval of a PB method to justify the lack of fire rated barriers between the fire pumps and other areas of the plant, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.7.)

- Section 3.5.16, which concerns the use of the fire protection water supply system for fire protection use only. The licensee requested NRC approval of a PB method to justify the use of the fire protection water supply for non-fire protection purposes, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.8)

- Section 3.6.1, which concerns the installation of standpipe and hose systems in buildings in the power block. The licensee requested NRC approval of a PB method to justify the installation of Class II standpipe and hose systems in lieu of Class III standpipe and hose systems, thereby meeting the requirements of NFPA 805. (See SE Section 3.1.4.9)

As discussed in SE Section 3.1.4 below, the NRC staff concludes that the use of PB methods to demonstrate compliance with these fundamental FPP elements is acceptable because the licensee demonstrated that the PB methods meet the requirements of 10 CFR 50.48(c)(2)(vii).

3.1.1.6 Compliance Strategy – Complies, With Required Action

For several NFPA 805, Chapter 3, requirements, the licensee stated that compliance with the fundamental FPP element will be achieved through the completion of a required action. The NRC staff concludes that the licensee's statement of compliance is acceptable because the included actions for achieving compliance will incorporate the provisions of NFPA 805 in the FPP and would be required by the proposed license condition.

In Table S-3 of the LAR, as supplemented, the licensee identifies each implementation item as applicable to both Unit Nos. 1 and 2.

A more detailed review of the licensee's compliance with Section 3.3.5.1 of NFPA 805 is provided below.

Section 3.3.5.1 of NFPA 805

The compliance strategy for NFPA 805 Section 3.3.5.1, in LAR Attachment A, Table B-1, is identified as "Complies with Required Action," with the actions being revising plant documentation and submitting for NRC approval. In LAR Attachment L, (page L-11) as supplemented, in Approval Request 3, the licensee indicated that fire zones contain wiring above suspended ceilings that is not in compliance with NFPA 805, Section 3.3.5.1 and that one of the bases for their approval request is that, " ... there are small quantities of low voltage video, communication, and data cables, which are not susceptible to self-ignition." However, the licensee provided no justification for its statements that these types of cables are not susceptible to self-ignition.

In its response to FPE RAI 01 (Reference 9), the licensee stated that the technical basis is provided in NUREG-1805, Section 7.3 and Appendix A (Reference 31), as follows:

> Section 7.3
> It is common practice to consider only self-ignited cable fires to occur in power cable trays since they carry enough electrical energy for ignition. Control and instrumentation cables typically do not carry enough electrical energy for self-ignition (Page 7-2).
>
> Appendix A
> Instrument circuits generally use low voltages (50 volts or less). Control circuits are commonly in the 120-250 volt range. Power circuits encountered within an NPP generally range from 120 to 4, 160 volts, with offsite power circuits ranging to 15 kV or higher (Page A-5).

Based on the information provided by the licensee, the NRC staff concludes that the licensee's response to the RAI is acceptable because low voltage cables do not carry enough electrical energy for self-ignition which is consistent with the guidance in NUREG-1805.

3.1.1.7 Compliance Strategy – Multiple Strategies

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

In each of these cases, the NRC staff concludes that the individual compliance statements are acceptable, the combination of compliance strategies is acceptable, and holistic compliance with the fundamental FPP element is assured.

3.1.1.8 Chapter 3 Sections Not Reviewed

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

- Sections that do not contain any technical requirements (e.g., NFPA 805, Sections 3.4.5 and 3.11).

- Sections that are <u>not applicable</u> because of the following:

    - The licensee stated that it does not have systems of this type installed (e.g., 3.8.1(3) which concerns manual fire alarm stations, 3.9.1(3) and (4), which concern water mist fire protection systems and foam-water sprinkler and foam-water spray systems), 3.10.1(3) which concerns clean agent fire extinguishing systems, and 3.10.4 which concerns the use of primary and backup gaseous fire suppression systems.

    - Hatch, Units 1 and 2 are boiling-water reactors (BWRs) and do not have reactor coolant pumps. The requirements are structured with an applicability statement (e.g., Section 3.3.12, which applies to reactor coolant pumps in non-inerted containments; or Sections 3.4.1(a)(2) and 3.4.1(a)(3), which apply to the fire brigade standards, which depend on the type of brigade specified in the FPP); or Section 3.5.1(a), which applies to the method to determine the water supply).

3.1.1.9 Compliance with Chapter 3 Requirements Conclusion

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

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

- Provided appropriate documentation of its state of compliance with NFPA 805, Chapter 3, requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:
  - With the requirement directly, or with the requirement directly after the completion of an implementation item or modification; or
  - With the intent of the requirement (or element) given adequate justification; or
  - By previous NRC staff approval of an alternative to the requirement; or
  - Through the use of an engineering equivalency evaluation; or
  - Through the use of a combination of the above methods; or
  - Through the use of a PB method that the NRC staff has approved in accordance with 10 CFR 50.48(c)(2)(vii).

3.1.2 Identification of the Power Block

NFPA 805 defines the power block as: "Structures that have equipment required for nuclear plant operations." The NRC staff reviewed the structures identified in LAR Attachment I, as comprising the "power block." The plant structures listed are established as part of the power block for the purpose of denoting the structures and equipment included in the RI/PB FPP that have additional requirements in accordance with 10 CFR 50.48(c) and NFPA 805. As stated in LAR Section 4.1.3, the power block includes structures that contain equipment required for nuclear plant operations such as Containment, Auxiliary Building, Service Building, Control Building, Fuel Building, Radioactive Waste, Water Treatment, Turbine Building, and intake structures or structures that are identified in the facility's pre-transition licensing basis.

The NRC staff concludes that the licensee appropriately evaluated the structures and equipment and adequately documented a list of those structures that fall under the definition of "power block" in NFPA 805.

3.1.3 Plant-specific Treatments or Technologies

The GL 2006-03 (Reference 38) requested that licensees evaluate their facilities to confirm compliance with existing applicable regulatory requirements in light of the results of NRC testing that determined both Hemyc™ and MT™ fire barriers failed to provide the protective function intended for compliance with existing regulations for the configurations tested using the NRC's thermal acceptance criteria. In a letter dated June 9, 2006 (Reference 59), the licensee stated that it does not have Hemyc™ nor MT™ installed on site. Since neither Hemyc™ nor MT™ Electrical Raceway Fire Barrier System (ERFBS) are used, the NRC staff concludes that the generic issue, GL 2006-03, related to the use of these ERFBSs is not applicable.

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 for use of the PB methods permitted elsewhere in the standard as a means of demonstrating compliance with

the prescriptive NFPA 805, Chapter 3, fundamental FPP elements and minimum design requirements. Paragraph 50.48(c)(2)(vii) of 10 CFR requires that an acceptable PB approach accomplish the following:

(A) Satisfies the performance goals, objectives, and 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 safe shutdown (SSD) capability).

In Attachment L of the LAR, as supplemented, the licensee requested NRC staff review and approval of PB methods to demonstrate an equivalent level of fire protection for the requirement of the elements identified in SE Section 3.1.1.5. The NRC staff's evaluation of these proposed methods is provided below.

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 SSC 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: NFPA 805, Section 3.2.3(1) – Performance-Based Fire Protection Inspection, Testing, and Maintenance Frequencies

In Attachment L of the LAR (page L-2), as supplemented, in Approval Request 1, the licensee requested approval of a PB method to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805.

The licensee stated that:

> NFPA 805, Section 2.6, Monitoring, requires 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 FPP in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid."
>
> NFPA 805, Section 2.6.1, Availability, Reliability, and Performance Levels, requires that "Acceptable levels of availability, reliability, and performance shall be established."
>
> NFPA 805, Section 2.6.2 requires that "Methods to monitor availability, reliability, and performance." shall be established. The methods shall consider the plant operating experience and industry operating experience."

3.1.4.1.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee stated that the PB inspection, testing, and maintenance frequencies will be established as described in EPRI Technical Report TR-1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide," (Reference 60).

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability.

3.1.4.1.2 NRC Staff Evaluation of the PB Alternative to NFPA 805, Section 3.2.3(1), "Procedures: Inspection, Testing, and Maintenance for Fire Protection Systems and Features Credited by the Fire Protection Program"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the use of EPRI TR-1006756 to establish inspection, testing, and maintenance frequencies for fire protection systems and features: (1) is consistent with the requirements of NFPA 805, Section 2.6; (2) is a method to ensure that the required NFPA 805 availability, reliability, and performance goals are maintained; (3) ensures that performance monitoring is performed in conjunction with the NFPA 805 monitoring program that considers site specific operating experience; (4) ensures that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analysis including those credited to meet the radioactive release performance criteria; and, (5) does not impact nuclear safety and does not compromise post-fire safe shutdown capability as analyzed previously.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the use of EPRI TR-1006756 will ensure that the availability and reliability of the fire protection systems and features credited for DID are maintained to the levels assumed in the NFPA 805 engineering analysis. In addition, the level of fire protection that will be provided so that a fire will not prevent essential safety functions from being performed is not changed because the use of EPRI TR-1006756 does not impact the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the licensee demonstrated that the use of EPRI TR-1006756 will ensure that the criteria in NFPA 805 Section 2.6 are met. The NRC staff determined that the proposed change does not impact any safety analysis

acceptance criteria used in the licensing basis because the licensee demonstrated that the use of EPRI TR-1006756 will not compromise automatic or manual fire suppression functions and post-fire safe shutdown capability.

3.1.4.1.3 NRC Staff Conclusion of the PB Alternative to NFPA 805, Section 3.2.3(1), “Procedures: Inspection, Testing, and Maintenance for Fire Protection Systems and Features Credited by the Fire Protection Program”

Based on the above, the NRC approves the proposed PB alternative of using EPRI Technical Report TR-1006756 to establish the inspection, testing, and maintenance frequencies for fire protection systems and features as the compliance method for NFPA 805, Section 3.2.3(1).

3.1.4.2 Approval Request 2: NFPA 805, Sections 3.3.4 – Thermal Insulation Materials

In Attachment L of the LAR (page L-7), as supplemented, in Approval Request 2, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection for NFPA 805, Section 3.3.4 regarding the requirement that thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials be noncombustible or limited combustible.

The licensee stated that:

> NFPA 805, Section 1.6.36 defines Limited Combustible as:
>
> > Material that, in the form in which it is used, has a potential heat value not exceeding 3500 Btu/lb (8141 kJ/kg) and either has a structural base of noncombustible material with a surfacing not exceeding a thickness of 1/8in. (3.2 mm) that has a flame spread rating not greater than 50, or has another material having neither a flame spread rating greater than 25 nor evidence of continued progressive combustion, even on surfaces exposed by cutting through the material on any plane.

The licensee stated that:

> NFPA 805, Section 1.6.41 defines Noncombustible as:
>
> > A material that, in the form in which it is used and under the conditions anticipated, will not ignite, burn, support combustion, or release flammable vapors when subjected to fire or heat.

3.1.4.2.1 Licensee’s Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for the use of flexible closed-cell thermal insultation in power block buildings that has a potential heat value that exceeds the NFPA 805 limit of 3500 Btu/lb, and for the use of 2-inch thick expanded polystyrene bead boards covered with aluminum stucco used as blowout panels for Unit 2.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.2.2 NRC Staff Evaluation of the PB Alternative to NFPA 805, Section 3.3.4, "Insulation Materials"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the insulation materials: (1) are included in the FM methodology to estimate additional HRRs, potential changes in conditions, and the possibility of new targets within a fire scenario; (2) do not adversely affect installed fire protection systems and features due to their limited application; (3) meet the intent of the revised limited combustible material definition because the materials have a flame spread rating of 25 or less; (4) will not result in any additional fire damage or flame spread because there are no significant ignition sources or fire hazards near the materials; and, (5) do not impact nuclear safety and do not compromise post-fire safe shutdown capability as analyzed previously.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the insulation materials are not considered a method for preventing fires from starting, or detecting, controlling, or extinguishing fires. In addition, the level of fire protection that will be provided so that a fire will not prevent essential safety functions from being performed is not changed because the limited quantities and concentrations of insulation materials do not impact the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the licensee demonstrated that the insulation materials meet the flame spread and smoke developed criteria in American Society for Testing and Materials, Standard E84, "Standard Test Method for Surface Burning Characteristics of Building Materials," (ASTM E84) (Reference 61), and will not support progressive continued combustion. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the insulation materials will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability.

3.1.4.2.3 NRC Staff Conclusion of the PB Alternative to NFPA 805, Section 3.3.4, "Insulation Materials"

Based on the above, the NRC approves the proposed PB alternative for the use of thermal insulation materials that do not meet the heat value content criteria of NFPA 805, Section 3.3.4, "Insulation Materials."

3.1.4.3 Approval Request 3: NFPA 805, Section 3.3.5.1 – Wiring Above Suspended Ceilings

In Attachment L of the LAR (page L-11), as supplemented, in Approval Request 3, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.5.1 requirement that states wiring above suspended ceilings shall be kept to a minimum, and that where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers.

The licensee stated that:

> NFPA 805, Section 3.3.5.1 states:
>
> > Wiring above suspended ceilings shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers.

3.1.4.3.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for the presence of unenclosed cables above the suspended ceilings in certain areas of the control building, radio chemistry lab and radiation protection area, CR area, and the shift supervisor's area.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.3.2 NRC Staff Evaluation of the PB Method to NFPA 805, Section 3.3.5.1, "Electrical: Wiring Above Suspended Ceilings"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the unenclosed cables above the suspended ceilings in the identified areas are: (1) kept to a minimum to support communications and other similar uses; (2) in areas where the majority of the power, control and instrumentation cables are enclosed; (3) predominantly Institute of Electrical and Electronics Engineers Standard 383, "Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations," (IEEE-383) (Reference 62) qualified; (4) in areas where there are no other significant combustibles; (5) in areas where fires involving cable trays without covers are

not expected to propagate beyond the cable tray of origin; (6) in an area equipped with smoke detection both above and below the ceiling (CR areas, fire zones 0024B and 0024C); (7) in an area where smoke from a fire above the ceiling can be manually exhausted to maintain a tenable environment in the MCR; (8) in an area protected by partial water spray suppression and spot-type smoke detection (HP Counting Room fire zone 0014G and Hot Lab fire zone 0014H), (9) considered in the plant FM in support of the fire risk model; and (10) subject to the requirements of NFPA 805, Section 3.3.5.1 for future replacements or new installations (See Table S-3, as supplemented, Implementation Item IMP-3).

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the unenclosed cables above the suspended ceilings are not considered a method for preventing fires from starting, or detecting, controlling, or extinguishing fires. In addition, the level of fire protection that will be provided so that a fire will not prevent essential safety functions from being performed is not changed because the limited quantities and concentrations of unenclosed cables above suspended ceilings do not impact the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the licensee demonstrated that the unenclosed cables above suspended ceilings have insulation that is predominantly IEEE-383 qualified which does not support progressive continued combustion. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the unenclosed cables above suspended ceilings will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability.

3.1.4.3.3 NRC Staff Conclusion of the PB Method to NFPA 805, Section 3.3.5.1, “Electrical: Wiring Above Suspended Ceilings”

Based on the above, the NRC approves the proposed PB alternative for the use of unenclosed cables above suspended ceilings that do not meet the criteria of NFPA 805, Section 3.3.5.1, “Electrical: Wiring Above Suspended Ceilings.”

3.1.4.4 Approval Request 4: NFPA 805, Section 3.3.5.2 – Metal Conduits

In Attachment L of the LAR (page L-16), as supplemented, in Approval Request 4, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.3.5.2, requirement that only metal tray and metal conduits shall be used for electrical raceways, that thin wall metallic tubing shall not be used for power, instrumentation, or control cables, and that flexible metallic conduits shall only be used in short lengths to connect components.

The licensee stated that:

> NFPA 805, Section 3.3.5.2 states:
>
> > Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.

3.1.4.4.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for the use of nonmetallic conduit in embedded applications, for existing installations of flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet, and for the future use of flexible metallic and PVC coated flexible metallic conduits in lengths up to 6-feet (See LAR Table S-3, Implementation Item IMP-20).

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.4.2 NRC Staff Evaluation of the PB Method to NFPA 805, Section 3.3.5.2, "Electrical: Metal Tray and Metal Conduits"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet in existing installations: (1) have equivalent physical and electrical protection as uncoated flexible metallic conduit, and because the characteristics of the metallic body of the conduit are not affected by the coating; (2) have a very thin PVC coating which is not expected to provide any credible influence on fire propagation and the amount of PVC introduced to a given fire area is considered negligible; (3) are installed in areas where existing controls such as fire-rated barriers, ERFBS, and spatial separation, etc. ensures redundant cabling and circuitry would not be affected by a fire; and (4) are installed such that the conduits are not in danger of being damaged by equipment or personnel.

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the non-metallic conduits in embedded applications: (1) are embedded in concrete which provides physical barrier protection and separation; (2) are not subject to flame or heat impingement from an external source which would result in structural failure, contribution to the fire load, and/or damage to circuits; (3) are allowed by NFPA 70, "National Electric Code" (Reference 63); (4) would not damage external targets if a failure were to occur that resulted in a fire; and (5) are installed such that they are not in danger of being damaged by equipment or personnel.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet and the non-metallic conduits in embedded applications are not considered a method for

preventing fires from starting, or detecting, controlling, or extinguishing fires. In addition, the level of fire protection that will be provided so that a fire will not prevent essential safety functions from being performed is not changed because the flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet and the non-metallic conduits in embedded applications do not impact the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because for the flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet the licensee demonstrated that the metallic body of the conduit is not affected by the coating, and for the non-metallic conduits in embedded applications, the licensee demonstrated that they are allowed by NFPA 70. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet and the non-metallic conduits in embedded applications will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability.

3.1.4.4.3 NRC Staff Conclusion of the PB Method to NFPA 805, Section 3.3.5.2, "Electrical: Metal Tray and Metal Conduits"

Based on the above, the NRC approves the proposed PB alternative for the use of flexible metallic and PVC coated flexible metallic conduits in lengths greater than 3-feet and the use of non-metallic conduits in embedded applications that do not meet the criteria of NFPA 805, Section 3.3.5.2, "Electrical: Metal Tray and Metal Conduits."

3.1.4.5 Approval Request 5: NFPA 805, Section 3.5.2 and 3.5.10 – Use of Check Valves in the Fire Water Tank Discharge Piping

In Attachment L of the LAR (page L-20), as supplemented, in Approval Request 5, the licensee requested approval of PB methods to demonstrate an equivalent level of fire protection for the:

NFPA 805, Section 3.5.2 states:

> The tanks shall be interconnected such that fire pumps can take suction from either or both. A failure in one tank or its piping shall not allow both tanks to drain. The tanks shall be designed in accordance with NFPA 22, Standard for Water Tanks for Private Fire Protection (Reference 64).

NFPA 805, Section 3.5.10 states:

> An underground yard fire main loop, designed and installed in accordance with NFPA 24, Standard for the Installation of Private Fire Service Mains and Their Appurtenances (Reference 65), shall be installed to furnish anticipated water requirements.

The licensee stated that:

> NFPA 22, 1976 Edition, Section 8-2.11 states:
>
> > An approved check valve shall be placed horizontally in the discharge pipe and shall be located in a pit under the tank when the tank is on an independent tower. When the tank is located over a building, the check valve shall ordinarily be placed in a pit, preferably outside the building. When yard room is not available, the check valve may be located on the ground floor or in the basement of a building provided that it is adequately protected against breakage.

3.1.4.5.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for the lack of check valves in the fire protection water storage tank discharge piping.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.5.2 NRC Staff Evaluation of the PB Method to NFPA 805 Sections 3.5.2 and 3.5.10, "Water Supply"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because the lack of check valves in the fire protection water storage tank discharge piping has limited impact on the ability of the fire water system to provide the needed water supply because: (1) there are alarms in the CR that activate when the water tank level is lowered 12 inches from the required tank level which is less than 10,000 gallons of the 300,000 gallon tank capacity; (2) prompt action is taken on a low level alarm through an abnormal operating procedure (AOP) to investigate the cause and take corrective actions; and, (3) control isolation valves are available for isolation of each tank and the associated discharge piping in the event of a leak.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the lack of check valves in the fire protection water storage tank discharge piping is not considered a method for preventing fires from starting. In addition, the level of fire protection that will be provided so that a fire will not prevent essential safety functions from being performed is not changed because of the lack of control valves in the fire water storage tank

discharge piping; the installed control valves, low-level water alarms, and AOPs ensure that the fire protection water storage tanks are not drained in the event of a leak and therefore, there is no impact on the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the lack of check valves on the fire protection water storage tank discharge piping does not impact the ability to provide the needed water supply during a fire event. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the lack of check valves in the fire protection water storage tank discharge piping will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability because of the installed control valves, low-level water alarms, and AOP that ensures the tanks are not drained in the event of a leak.

3.1.4.5.3 NRC Staff Conclusion of the PB Method to NFPA 805 Sections 3.5.2 and 3.5.10, "Water Supply"

Based on the above, the NRC approves the proposed performance-based alternative for the lack of check valves in the fire protection water storage tank discharge piping that does not meet the criteria of NFPA 805, Section 3.5.2 and 3.5.10.

3.1.4.6 Approval Request 6: NFPA 805, Section 3.5.3 – Installation of Fire Pumps

In Attachment L of the LAR (page L-23), as supplemented, in Approval Request 6, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection to the NFPA 805, Section 3.5.3, requirement that fire pumps be designed and installed in accordance with NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection (Reference 66), and shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source.

The licensee stated that:

> NFPA 805 Section 3.5.3 states:
>
> > Fire pumps, designed and installed in accordance with NFPA 20, Standard for the Installation of Stationary Pumps for Fire Protection, shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source.

3.1.4.6.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for six different deviations affecting nine code sections that include configuration of overcurrent devices, use of Class 1 E emergency power supply circuit breakers, inadequate interrupting rating for the fire pump circuit, use of non-listed components within the controller, portions of the controller function located out of sight of the motor, and lack of a handle or lever for the current motor-circuit switching mechanism of the controller.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.6.2 NRC Staff Evaluation of the PB Method to NFPA 805 Section 3.5.3 “Fire Pumps”

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because in regards to the fire pump code deviations (1) the licensee has demonstrated through operating experience the adequacy of the existing circuit breaker and electric motor-driven fire pump configuration; (2) the 1 E bus is supplied by offsite power as well as an emergency generator; (3) the licensee has evaluated the non-Underwriters Laboratories (UL) listed components and determined that they are equivalent to listed components and adequate for their use in their current configuration; (4) the breaker at the 1 E bus can be operated mechanically in the event of a loss of control power; (5) the electric fire pump is inspected, tested, and maintained at periodic intervals to ensure reliable operation; (6) there are two redundant diesel fire pumps with both automatic and manual start capability that remain available in the unlikely event that the electric fire pump is not available; (7) a 70 gpm, 125 psi, pressure maintaining jockey pump is provided to keep the system filled and pressurized during low flow draw-offs and leakages; (8) under normal operating conditions, three fire pumps with independent power supplies and controls are available to serve the maximum sprinkler system and hose stream hydraulic demands, and, (9) site personnel are trained to operate the controller function and are knowledgeable of its location.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the fire pump code deviations are not considered a method for preventing fires from starting. In addition, the level of fire protection that will be provided, so that a fire will not prevent essential safety functions from being performed, is not changed because of the redundancy provided by the two diesel driven fire pumps and therefore, there is no impact on the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the fire pump code deviations do not impact the ability to provide the needed water supply during a fire event. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the fire pump code deviations will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability primarily because of the redundancy provided by the diesel fire pumps.

3.1.4.6.3 NRC Staff Conclusion of the PB Method to NFPA 805 Section 3.5.3 “Fire Pumps”

Based on the above, the NRC approves the proposed PB alternatives for the fire pump code deviations that do not meet the criteria of NFPA 805, Section 3.5.3.

3.1.4.7 Approval Request 7: NFPA 805, Section 3.5.5 – Fire Pump Separation

In Attachment L of the LAR (page L-28), as supplemented, in Approval Request 7, the licensee requested approval 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.

The licensee stated that:

> NFPA 805 Section 3.5.5 states:
>
>> Each pump and its driver and controls shall be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers.

3.1.4.7.1 Licensee’s Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested the use of a PB method for the lack of fire rated barriers between the fire pumps and other areas of the plant.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.7.2 NRC Staff Evaluation of the PB Method to NFPA 805 Section 3.5.5 “Fire Pump Separation”

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because in regards to the lack of fire rated barriers between the fire pumps and other areas of the plant: (1) the fire pumps are properly separated from one another with fire rated barriers, and as such, the loss of one pump would not impact the ability to provide 100% of the required fire water demand; (2) there are no significant intervening combustibles between the fire pump house and nearby structures; (3) the closest structures to the fire pump house are the fire protection water tanks which are located approximately 10 feet away and these tanks do not present a hazard to the fire pumps or their functionality; (4) both

the diesel driven pumps and the electric fire pump are provided with automatic sprinkler protection; (5) with the exception of the fire water storage tanks, the fire pump house is located greater than 50 feet from the nearest significant structure; and, (6) there are no credible fire scenarios that would impact all three fire pumps.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the lack of fire rated barriers between the fire pumps, and other areas of the plant, are not considered a method for preventing fires from starting. In addition, the level of fire protection that will be provided, so that a fire will not prevent essential safety functions from being performed, is not changed because of the fire rated barriers between the fire pumps themselves, the automatic sprinkler protection provided in the pump house, and the lack of credible fire scenarios. Therefore, there is no impact on the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the lack of fire rated barriers between the fire pumps and other areas of the plant do not impact the ability to provide the needed water supply during a fire event. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the lack of fire rated barriers between the fire pumps and other areas of the plant will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability because of the fire rated barriers between the fire pumps themselves, the automatic sprinkler protection provided in the pump house, and the lack of credible fire scenarios.

3.1.4.7.3 NRC Staff Conclusion of the PB Method to NFPA 805 Section 3.5.5 "Fire Pump Separation"

Based on the above, the NRC approves the proposed PB alternatives for the lack of fire rated barriers between the fire pumps and other areas of the plant that do not meet the criteria of NFPA 805, Section 3.5.3.

3.1.4.8 Approval Request 8: NFPA 805, Section 3.5.16 – Fire Protection Water Supply System

In Attachment L of the LAR (page L-31), as supplemented, in Approval Request 8, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection for the NFPA 805, Section 3.5.16, requirement that the fire protection water supply system be dedicated for fire protection use only.

The licensee stated that:

> NFPA 805 Section 3.5.16 states:
>
> > The fire protection (FP) water supply system shall be dedicated for fire protection use only.
> >
> > Exception No. 1: Fire protection water supply systems shall be permitted to be used to provide backup to nuclear safety systems, provided the fire protection water supply systems are designed and maintained to deliver

> the combined fire and nuclear safety flow demands for the duration specified by the applicable analysis.
>
> Exception No. 2: Fire protection water storage can be provided by plant systems serving other functions, provided the storage has a dedicated capacity capable of providing the maximum fire protection demand for the specified duration as determined in this section.

3.1.4.8.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested approval of a PB method for the use of fire protection water supply system for several non-fire protection purposes including decay heat removal, loss of plant service water, standby liquid control, spent fuel pool and reactor vessel cooling, alternate reactor pressure vessel water level control, primary containment flooding, and cleaning main condenser tubes.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.8.2 NRC Staff Evaluation of the PB method to NFPA 805, Section 3.5.16, "Dedicated Use of Fire Protection Water Supply"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because in regards to the use of the fire protection water supply for non-fire protection purposes (1) DHR is primarily used during refueling outages when the decay heat load is high so system use is limited and the maximum water flow to the DHR cooling tower is 260 gpm; (2) for loss of plant service water (PSW) the flow to the mechanical vacuum pump would be approximately 125 gpm and the condensate pump only requires approximately 6 gpm of water for cooling to the pump bearings; (3) for standby liquid control (SBLC) the maximum amount of fire protection water required to refill the SBLC tank, based on the overflow capacity of the tank, is 5150 gallons and would only be used if the normal supply from the demineralized water transfer system is unavailable; (4) the fire protection water supply is only used if the normal and alternative sources of water are unavailable; (5) for cleaning main condenser tubes, the flow rate is only 26 gpm and is only conducted during outages when the condenser is available for maintenance which is typically only 5 days; (6) concurrent flow demands from emergency use and fire suppression is highly unlikely, but if concurrent flow demands were to occur, the fire protection water system would recover quickly and additional fire pumps could be started to ensure appropriate flow and pressure; (7) the fire protection water

storage tanks levels are controlled by automatic makeup pumps and each 700 gpm pump is capable of refilling either tank within 8 hours; (8) routine surveillance checks using a local tank level indicator verify that the tank level is kept above the minimum level; and (9) restoration procedures are included in the applicable Emergency Operating, Abnormal Operating, and Infrequent Operations procedures which realign the system to normal operating conditions after non-fire protection use.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the use of fire protection water for non-fire protection purposes is not considered a method for preventing fires from starting. In addition, the level of fire protection that will be provided, so that a fire will not prevent essential safety functions from being performed, is not changed because of the use of fire protection water for non-fire protection purposes is limited to certain situations where the normal and alternative water supplies are not available, and concurrent demands are unlikely but can be alleviated by starting additional fire pumps and stopping the non-fire protection use. Therefore, there is no impact on the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the non-fire protection use of the fire protection water supply does not impact the ability to provide the needed water supply during a fire event. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the use of the fire protection water system for non-fire protection purposes will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability because of the ability to stop the non-fire protection use, restore the system to its rated capacity, and use the redundant diesel fire pumps.

3.1.4.8.3 NRC Staff Evaluation of the PB method to NFPA 805, Section 3.5.16, “Dedicated Use of Fire Protection Water Supply”

Based on the above, the NRC approves the proposed PB alternatives for use of the fire protection water system for non-fire protection purposes which do not meet the criteria of NFPA 805, Section 3.5.5.

3.1.4.9 Approval Request 9: NFPA 805, Section 3.6.1 – Installation of Standpipe and Hose Systems

In Attachment L of the LAR (Page L-36), as supplemented, in Approval Request 9, the licensee requested approval of a PB method to demonstrate an equivalent level of fire protection to the NFPA 805, Section 3.6.1, requirement that Class III standpipe and hose systems be installed in accordance with NFPA 14, “Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems,” (Reference 67).

The licensee stated that:

> NFPA 805 Section 3.6.1 states:
>
> > For all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14, Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems.

3.1.4.9.1 Licensee's Request

In Attachment L of the LAR, the licensee provided the basis for the request, including a DID and safety margin evaluation. The licensee requested approval of a PB method for the use of reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections as required for Class II service.

The licensee concluded that the alternative approach outlined in the LAR achieves the following criteria:

- Satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release;
- Maintains safety margins; and
- Maintains fire protection defense-in-depth (fire prevention, fire detection, fire suppression, mitigation, and post-fire safe shutdown capability).

3.1.4.9.2 NRC Staff Evaluation of the PB Method to NFPA 805 Section 3.6.1, "Standpipe and Hose Stations"

The NRC staff reviewed the information provided by the licensee in its LAR, which included discussions of the impact of the proposed change on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, DID, and safety margins as required by 10 CFR 50.48(c)(2)(vii).

The NRC staff determined that the proposed change has no impact on the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release because in regards to the use of the reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections (1) the evaluated Class II standpipe and hose systems have sufficient hose to reach all portions of the fire zone in which they are located; (2) there is either full or partial overlapping redundant hose coverage from adjacent fire zones; (3) areas that have only partial overlapping redundant coverage are protected by automatic wet-pipe sprinkler systems; (4) the fire brigade is trained to use standpipe and hose systems located in adjacent fire areas/fire zones; and (5) there is adequate available flow and pressure from the fire water supply mains feeding the evaluated standpipe and hose systems.

The NRC staff determined that the proposed change has no impact on any of the DID echelons because the reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections are not considered a method for preventing fires from starting. In addition, the level of fire protection that will be provided, so that a fire will not prevent essential safety functions from being performed, is not changed because the reactor building hose stations below the 130' elevation that only include 1.5 in. hose connections have sufficient hose to reach all portions of the fire zone in which they are located and there is full or partial overlapping

redundant hose coverage. Therefore, there is no impact on the availability and reliability of fire protection systems and features.

The NRC staff also determined that the proposed change continues to maintain sufficient safety margins. The NRC staff determined that the proposed change does not impact any codes and standards, or their alternatives accepted for use by the NRC because the reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections do not impact the ability to provide the needed water supply during a fire event. The NRC staff determined that the proposed change does not impact any safety analysis acceptance criteria used in the licensing basis because the licensee demonstrated that the reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections will not compromise automatic or manual fire suppression functions and post fire safe shutdown capability because of the sufficient hose to reach all portions of the fire zone and because of the full or partial overlapping coverage.

3.1.4.9.3 NRC Staff Evaluation of the PB Method to NFPA 805 Section 3.6.1, "Standpipe and Hose Stations"

Based on the above, the NRC approves the proposed PB alternative for the reactor building hose stations below the 130 ft. elevation that only include 1.5 in. hose connections which do not meet the criteria of NFPA 805, Section 3.6.1.

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, are 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 its LAR, Section 4.2.1, "Nuclear Safety Capability Assessment Methodology", 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 RAs.
4. Evaluating MSO.

Section 3.2.1 of the SE 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 Section 3.5 of the SE provides the NRC staff's review of the fourth step. Section 3.2.2 of the SE provides the NRC staff's review regarding the establishment of safe and stable conditions for the plant. Section 3.2.3 of the SE discusses the applicability of feed-and-bleed.

Section 3.2.4 of the SE provides the NRC staff's review of the licensee's assessment of MSOs. Section 3.2.5 of the SE 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 of the SE) 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, through the endorsement of NEI 04-02, Revision 2, and Chapter 3 of NEI 00- 01, Revision 2 (Reference 17), 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, as supplemented.

As noted in NEI 04-02, the nuclear safety goals, objectives, and performance criteria of NFPA 805 (see Section 2.0 of the SE) 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 and 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 49), 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 Section 4.2.1.1 of the LAR, the licensee stated it evaluated its NSCA methodology against the guidance provided in Chapter 3 of NEI 00-01, Revision 2, as discussed in NEI 04-02, Appendix B-2. The licensee stated that it correlated each specific section of NFPA 805 Section 2.4.2 to the corresponding section of Chapter 3 of NEI 00-01 Revision 2 and that based upon the content of the NEI 00-01 methodology statements, it determined the applicability of the section to Hatch.

The licensee further stated that it compared the plant-specific methodology to applicable sections of NEI 00-01 and determined which alignment statement and associated basis to assign including: Aligns; Aligns with intent; Not in Alignment; Not in Alignment, but Prior NRC Approval; and, Not in Alignment, but no adverse consequences. The licensee further stated that for those sections that do not align, it made an assessment to determine if the failure to maintain strict alignment with the guidance in NEI 00-01 could have adverse consequences, and that since NEI 00-01 is a guidance document, portions of its text could be interpreted as 'good practice' or intended as an example of an efficient means of performing the analyses. The licensee documented the results of this review in LAR Attachment B, as updated by RAI responses.

3.2.1.1 Attribute Alignment – 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 attributes, the licensee determined that the SSA aligns directly with the attribute. In these instances, based on the validity of the licensee’s statements, the NRC staff concludes that the licensee’s statements of alignment are acceptable. A more detailed review of the licensee’s compliance with Sections 3.4.1.7 and 3.5.2.4 of NEI 00-01 is provided below.

<u>NEI 00-01, Section 3.4.1.7</u>

Section 3.4.1.7 of NEI 00-01, states, “For the components on the required safe shutdown path classified as required hot shutdown components as defined in Appendix H, Appendix R

compliance requires that one train of systems necessary to achieve and maintain hot shutdown conditions from either the CR or emergency control station(s) is free of fire damage (III.G.1.a)."

In Table S-2 of the LAR, as supplemented, Item 1, the licensee indicated that a modification will replace or install fusible links for various sliding fire doors to meet the requirements of its code of record, and to resolve the potential water intrusion into SSD Switchgear 1R23S004 due to fire suppression activities.

In its response to SSD RAI 02 (Reference 9), the licensee clarified that the Table S-2, as supplemented, Item 1 modification will also replace the existing 180^0 angle nozzle located outside of fire door 1L48C31 with a 65^0 angle nozzle to eliminate the direct water spray onto Switchgear 1R23S004 during sprinkler discharge. In addition, since Switchgear 1R23S004 is set on a 3-inch pad, any minimal amount of water that passes through the fire door prior to the actuation of the fusible link would not affect Switchgear 1R23S004. Therefore, the licensee determined that the timely actuation of the fusible link is not relied upon to protect Switchgear 1R23S004 from water damage. The licensee revised the proposed Table S-2, as supplemented, Item 1 modification description to include the replacement of fire nozzles as appropriate.

Based on the above, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that the proposed modification will preclude water intrusion into SSD Switchgear 1R23S004 due to fire suppression activities.

NEI 00-01, Section 3.5.2.4

Section 3.5.2.4 of NEI 00-01, states, "The concern for the common power source associated circuits of concern is the loss of a safe shutdown power source due to inadequate breaker/fuse coordination. In the case of a fire-induced cable failure on a non-safe shutdown load circuit supplied from the safe shutdown power source, a lack of coordination between the upstream supply breaker/fuse feeding the safe shutdown power source and the load breaker/fuse supplying the non-safe shutdown faulted circuit can result in loss of the safe shutdown bus. This would result in the loss of power to the safe shutdown equipment supplied from that power source preventing the safe shutdown equipment from performing its required safe shutdown function. The licensee's analysis indicated that some protective devices may not be coordinated or coordination may be undetermined but will be addressed through procedures.

In its response to SSD RAI 01 (Reference 9), the licensee confirmed that it completed electrical coordination studies for credited power supplies per LAR, Attachment B, Sections 3.3.1.1.4, 3.3.1.1.5, 3.3.2, 3.5.2.4, and 3.5.2.5. The licensee stated that it resolved all breaker coordination issues except for those associated with plant modifications listed in LAR Table S-2, as supplemented, Items 6 and 7. In addition, the licensee stated that it will address the multiple high impedance faults (MHIF) issue using the guidance in NUREG/CR-7150, Volume 3, (see LAR Table S-3, as supplemented, Implementation Item IMP-9). Based on the information provided by the licensee, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that an electrical coordination study has been completed, that all breaker coordination issues have been resolved with the exception of two related to plant modifications, and that the multiple high impedance fault issue will be addressed in accordance with NRC recommended guidance through an implementation item which would be required to be completed by the proposed license condition.

3.2.1.2 Attribute Alignment − Aligns with Intent

For certain of the NEI 00-01, Chapter 3 attributes, the licensee determined that the SSA aligns with the intent of the attribute and provided additional clarification when describing its means of alignment. A more detailed review of the licensee's compliance with Sections 3.1.1.7, 3.1.1.9, 3.1.2.4, 3.1.3.4, 3.2.1.1, 3.2.2.3, and 3.4.1.6 of NEI 00-01 is provided below.

The NRC staff has reviewed each of the above attributes and determined that the licensee has provided justifications for their methods of identifying and selecting SSD equipment, and for achieving and maintaining post-fire safe and stable conditions. The NRC staff's review of the licensee's evaluation for each of these sections is discussed below.

Section 3.1.1.7 of NEI 00-01, states in part:

> For the case of redundant shutdown, offsite power may be credited if demonstrated to be free of fire damage. Offsite power should be assumed to remain available for those cases where its availability may adversely impact safety (i.e., reliance cannot be placed on fire causing a loss of offsite power if the consequences of offsite power availability are more severe than its presumed loss). No credit should be taken for a fire causing a loss of offsite power. For areas where train separation cannot be achieved and alternative shutdown capability is necessary, shutdown must be demonstrated both where offsite power is available and where offsite power is not available for 72 hours.

Section 3.1.1.9 of NEI 00-01, states in part:

> The post-fire safe shutdown analysis assumes a 72-hour coping period starting with a reactor scram/trip. Fire-induced impacts that provide no adverse consequences to hot shutdown within this 72-hour period need not be included in the post-fire safe shutdown analysis. At least one train can be repaired or made operable within 72 hours using onsite capability to achieve cold shutdown.

Section 3.1.2.4 of NEI 00-01, states in part:

> [BWR] Systems selected for the decay heat removal function(s) should be capable of:
>
> - Removing sufficient decay heat from primary containment, to prevent containment over-pressurization and failure.
> - Satisfying the net positive suction head (NPSH) requirements of any safe shutdown systems taking suction from the containment (suppression pool).
> - Removing sufficient decay heat from the reactor to achieve cold shutdown. (This is not a hot shutdown requirement.) This does not restrict the use of other systems.

Section 3.4.1.6 of NEI 00-01, states in part:

> Where appropriate to achieve and maintain cold shutdown within 72 hours, use repairs to equipment required in support *of* post-fire shutdown.

Attachment B of the LAR states that the licensee's analysis explicitly accounts for the criteria listed in Section 3.1.1.7 of NEI 00-01. The Hatch NSCA models offsite power, as well as onsite power from the emergency diesel generators (EDGs), and offsite power is only credited in fire areas where it can be demonstrated to be free of fire damage. Furthermore, LAR Attachment B also states that the NSCA demonstrates that the plant can be placed in a safe and stable condition in accordance with NFPA 805, and that the potential fire effects on systems and components required to maintain cold shutdown is addressed in the NPO analysis. The NRC staff's review determined that the licensee aligns with the intent of NEI 00-01, Section 3.1.1.7, Section 3.1.1.9, Section 3.1.2.4 and Section 3.4.1.6 because the licensee analyzed the availability of offsite power for each fire area and has demonstrated that the plant can be placed in a safe and stable condition and can achieve cold shutdown, as necessary, in the event of a fire.

Section 3.1.3.4 of NEI 00-01, states in part:

> Assign a path designation to each combination of systems. The path will serve to document the combination of systems relied upon for safe shutdown in each fire area.

Attachment B of the LAR states that Hatch's analysis identifies the overall process utilized to model the combinations of plant systems that satisfy each of the NSPC, and each performance goal may have multiple success paths representing a different combination of systems/functions.

The NRC staff's review determined that the licensee analysis aligns with the intent of NEI 00-01, Section 3.1.3.4. The licensee has identified a SSD success path for each fire area.

Section 3.2.1.1 of NEI 00-01, states in part:

> Safe shutdown equipment can be divided into two categories. Equipment may be categorized as (1) primary components or (2) secondary components.

Attachment B of the LAR states that the Hatch NSCA made no explicit distinction between primary and secondary components. In analyzing the SSD success path, secondary components are included with the primary component, and any required or associated circuits are also assigned to the primary component. The NRC staff's review determined that the licensee aligns with the intent of NEI 00-01, Section 3.2.1.1 because the licensee's SSA methodology includes all primary and supporting components required for the SSD success path in each fire area.

Section 3.2.2.3 of NEI 00-01, states in part:

> Prepare a table listing the equipment identified for each system and the shutdown path that it supports.
>
> Identify instrument tubing that may cause subsequent effects on instrument readings or signals as a result of fire. Determine and consider the fire area location of the instrument tubing when evaluating the effects of fire damage to circuits and equipment in the fire area.

Attachment B of the LAR states that specific SSD paths are not identified; rather, each component required for SSD is tied to at least one basic event fault tree in the ARCPlus software. These fault trees depict the various component and system interrelationships that must be met to achieve SSD. In addition, an SSD Equipment List (SSEL) has been developed, and the required information to support the SSA using the ARCPlus software has been included in the SSEL. LAR Attachment B also states that instrument sense lines, which support the achievement of the NSPC of NFPA 805 have also been analyzed and documented. The NRC staff's review determined that the licensee analysis aligns with the intent of NEI 00-01, Section 3.2.2.3 because the licensee's SSA methodology includes all components, including instrument sense lines required for the SSD success path in each fire area.

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

The licensee determined that there were no requirements that are not aligned with the NEI 00-01 attributes but had prior NRC approvals.

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

The licensee determined that there were no requirements that are not aligned with the NEI 00-01 attributes, but with no adverse consequences.

3.2.1.5 Attribute Alignment − Not in Alignment

The licensee determined that there were no requirements that are not aligned with the NEI 00-01 attributes requiring further assessments.

3.2.1.6 NFPA 805 − Nuclear Safety Capability Assessment Methods Conclusion

The NRC staff reviewed the documentation provided by the licensee describing the process used to perform the NSCA required by NFPA 805, Section 2.4.2. The licensee performed this evaluation by comparing the SSA against the NFPA 805 NSCA methodology requirements using NEI 00-01, Revision 2. The licensee documented the results of its review in LAR Attachment B, Table B-2 and B-3, in accordance with NEI 04-02, Revision 2 (Reference 7).

Based on its review of the information provided in the licensee's submittal, as supplemented, the NRC staff concludes that the method used by the licensee to perform the NSCA with respect to the selection of systems and equipment, selection of cables, and identification of the location of nuclear safety equipment and cables as required by NFPA 805, Section 2.4.2, is acceptable because it either:

- Met the NRC-endorsed guidance directly,
- Met the intent of the endorsed guidance and the licensee provided adequate justification,
- Had a previous NRC staff approval of an alternative to the guidance, or
- Could demonstrate that not meeting the guidance had no adverse effect.

3.2.2 Maintaining Fuel in a Safe and Stable Condition

The nuclear safety goals, objectives, and performance criteria of NFPA 805 allow more flexibility than the previous deterministic FPPs based on Appendix R to 10 CFR 50 and NUREG-0800, Section 9.5.1.1 (Reference 68), since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition rather than achieve and maintain cold shutdown in 72 hours. In LAR Section 4.2.1.2, the licensee stated that the NFPA 805 licensing basis is to achieve and maintain the fuel in a safe and stable condition assuming a fire occurs during Mode 1 (Power Operation), Mode 2 (Startup), or Mode 3 (Hot Shutdown), when the motor control center (MCC) breakers for the RHR shutdown cooling suction valves are de-energized. The licensee further stated that the analyses demonstrate that each fire area can maintain safe and stable hot shutdown conditions for a 24-hour coping period.

The licensee stated that to establish and maintain safe and stable conditions, Reactor Pressure Vessel (RPV) inventory makeup is provided by high pressure coolant injection (HPCI), RCIC, RHR or core spray (CS) systems, and that DHR is provided by RHR and RHR Service Water (RHRSW) with heat transfer to the ultimate heat sink. The licensee also stated that RPV pressure control is accomplished with the safety relief valves (SRVs). The licensee further stated that required alternating current (AC) buses are powered from Offsite Power (OSP) or onsite EDGs and required direct current (DC) buses are maintained from battery chargers with battery backup.

The licensee stated that minimum shift operating staff are capable of performing RAs to establish and maintain safe and stable plant conditions and that procedures provide for the establishment of the Emergency Response Organization (ERO), which can provide normal operations staff with support to maintain Hot Shutdown conditions for an extended period of time.

The licensee stated the following methods to maintain safe and stable conditions and related support actions:

- Reactivity Control is established and maintained by a reactor scram and insertion of control rods. Subcritical conditions are achieved for all reactor operating modes and maintaining subcritical conditions after a reactor scram is a passive function. There are no additional actions required for reactivity control to sustain safe and stable conditions beyond 24 hours.

- Hatch has design features and procedures to ensure that an adequate source of inventory is provided for RPV level control in sustained Mode 3 (Hot Shutdown) conditions. Makeup water will be provided from the Condensate Storage Tank (CST) and subsequently transitioned to the suppression pool. Adequate CST water capacity is available to maintain RPV level until conditions are met to transfer the RCIC, HPCI, RHR or CS pump suction to the suppression pool. Inventory required for RPV makeup when using low pressure injection is recirculated from the suppression pool. There are no additional actions required for makeup inventory to sustain safe and stable conditions beyond 24 hours.

- RPV Inventory and Pressure Control is maintained using high pressure systems, HPCI and RCIC, or low pressure injection with RHR in low pressure coolant injection (LPCI) mode or the CS system. For low pressure injection, the SRVs are credited for pressure reduction. Adequate nitrogen sources are available to the SRVs to establish safe and

stable conditions. Additional actions may be required to provide nitrogen from backup onsite or offsite supplies to sustain safe and stable conditions beyond 24 hours. Actions to replenish nitrogen supplies are considered routine, are covered by existing procedures and can be anticipated in ample time.

- Core decay heat in Mode 3 (Hot Shutdown) will be rejected to the torus through the SRV tailpipes. Long term cooling is maintained using the RHR system in Alternate Shutdown Cooling mode or Suppression Pool Cooling mode with at least 2 SRVs available to provide core flow. The RHRSW system rejects decay heat to the ultimate heat sink. There are no additional actions required for DHR to sustain safe and stable conditions beyond 24 hours.

- Exhaust steam from the RCIC/HPCI turbine is discharged to the barometric condensers and then returned to the suppression pool. Although this increases the suppression pool temperature, the peak pool temperature and pressure are within the design limits of the containment, and adequate NPSH for credited pumps is assured. There are no additional actions required for Suppression Pool cooling to sustain safe and stable conditions beyond 24 hours.

- Hatch is equipped with five diesel generators (DGs) supplying standby power to three 4.16kV essential buses for each unit. DG 1 B is shared between Unit 1 and 2; supplying power to either Bus 1 For 2F. The Fuel Oil Supply Storage and Transfer Subsystem provides a supply of fuel oil to the diesel engines. The storage capacity of the tanks is sufficient to operate four DGs for 7 days at 3250 KW power operation. Day tanks contain capacity for 2 hours of full-load operations. In the event that fuel oil supply approaches exhaustion after seven days, actions to replenish fuel oil are considered routine, are covered by existing procedures and can be anticipated in ample time.

- Battery chargers are credited with maintaining DC Station batteries at rated voltage. Should AC charging sources be lost, RAs may be required to recover sufficient chargers. Each Station Service battery is sized adequately to support the required loads for two hours without recharging. With battery chargers aligned to required DC buses, there are no additional actions required to sustain safe and stable conditions beyond 24 hours.

- A separate 125-V diesel building battery is furnished for each DG and its associated 4-kV bus. Each of the diesel building batteries is sized adequately to support the required loads for two hours without recharging. With battery chargers aligned, there are no additional actions required to sustain safe and stable conditions beyond 24 hours.

- The PSW system features two strainers per unit to remove suspended matter from the water leaving the PSW pumps. The operation of the strainers includes an automatic backwash feature. Additionally, plant procedures ensure that operators monitor differential pressure across the strainers. Additional action may be required to manually backwash PSW strainers to sustain safe and stable conditions beyond 24 hours.

- The RHRSW system features four strainers per unit to remove suspended matter from the water leaving the RHRSW pumps. The design of the RHRSW strainers provides for manual backwash. Plant procedures ensure that operators monitor differential pressure across the strainers during RHRSW operation. Additional action may be required to

manually backwash RHRSW strainers to sustain safe and stable conditions beyond 24 hours.

- The NSCA includes the required process monitoring instruments for the operation of each credited system. RAs may be required to mitigate the loss of required instrumentation to establish safe and stable conditions. There are no additional actions required for process monitoring to sustain safe and stable conditions beyond 24 hours.

Based on the information provided by the licensee as discussed above, that includes the licensee's methods and support actions to maintain the fuel in a safe and stable condition, the NRC staff concludes that there is reasonable assurance that the fuel can be maintained in a safe and stable condition, post-fire, for an extended period of time.

3.2.3 Applicability of Feed and Bleed

Hatch, Units 1 and 2, are BWRs; therefore, feed and bleed does not apply.

Section 50.48(c)(2)(iii) of 10 CFR limits the use of feed and bleed and states:

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

The NRC staff reviewed LAR Table 5-3, "10 CFR 50.48(c) – Applicability/Compliance Reference," and LAR Attachment C, as supplemented, to evaluate whether the licensee meets the feed and bleed requirements. In Table 5-3 of the LAR, the licensee stated that feed and bleed is not utilized as the sole fire-protected SSD methodology. The NRC staff confirmed this by reviewing the designated safe SSD listed in LAR Attachment C, as supplemented, for each fire area. This review confirmed that all fire area analyses include the SSD equipment necessary to provide DHR without relying on feed and bleed. In addition, all fire areas either met the deterministic requirements of NFPA 805, Section 4.2.3, or the PB evaluation performed in accordance with NFPA 805, Section 4.2.4, demonstrated that the integrated assessment of risk, DID, and safety margins for the fire area was acceptable.

Based on its review of the information provided in LAR Table 5-3, as well as the fire area analyses documented in LAR Attachment C, as supplemented, the NRC staff concludes that the licensee meets the requirements of 10 CFR 50.48(c)(2)(iii), because feed and bleed does not apply to Hatch, Units 1 and 2, because they are BWRs.

3.2.4 Assessment of Multiple Spurious Operations

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

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

circuits, and shorts to ground, to identify circuits that are required to support the proper operation of components required to achieve the nuclear safety performance criteria, including spurious operation and signals.

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

The NRC staff reviewed Section 4.2.1.4 of the LAR, "Evaluation of Multiple Spurious Operations," and LAR Attachment F, to determine whether the licensee has adequately addressed MSO concerns.

As part of the NFPA 805 transition project, the licensee stated that it reviewed and evaluated the susceptibility to fire-induced MSOs. The licensee stated that it conducted the process in accordance with NEI 04-02 and RG 1.205, as supplemented by FAQ 07-0038 (Reference 47).

In Attachment F of the LAR, the licensee stated that the review method used insights from the FPRA developed in support of transition to NFPA 805 and consisted of the following:

- Step 1 - Identify potential MSOs of concern.

- Step 2 - Conduct an expert panel to assess plant-specific vulnerabilities (e.g., per NEI 00-01, Revision 1, Section F.4.2).

- Step 3 - Update the FPRA model and NSCA to include the MSOs of concern.

- Step 4 - Evaluate for NFPA 805 compliance.

- Step 5 - Document the results.

For Step 1, the licensee stated that it conducted the initial MSO identification review using extensive review of plant systems and drawings to determine potential pathways and that this initial review was then supplemented by review of generic industry lists and data sources used as input to the overall assessment of MSOs. The licensee stated that the plant MSO identification process resulted in a list of potential MSO pathways for consideration by the MSO expert panel, and it used the following as input to the overall assessment of MSOs:

- Post-fire SSA
- Generic list of MSOs from Boiling Water Reactors Owners Group (BWROG)
- Self-assessment results
- PRA insights
- Operating Experience

For Step 2, the licensee stated that the 2009 Expert Panel consisted of a two-day meeting with representatives from SNC (and contractors) with experience in fire protection, post-fire SSD, circuit analysis, system engineering, plant operations and PRA. The panel conducted document reviews and held discussions on potential fire-induced spurious operations that could potentially impact plant safety.

The licensee stated that the 2012 supplemental Expert Panel consisted of a one-day phone conference meeting with representatives from SNC (and contractors) with experience in fire protection, post-fire SSD, circuit analysis, system engineering, PRA, and plant operations. The panel conducted document reviews, and held discussions focused on the new and clarified BWR MSO scenarios.

The licensee stated that it conducted training in the form of an introductory overview and slide presentation and that topics discussed included the purpose and scope of the SSA, PRA overview, overview training on the MSO issue including background on Fire-Induced MSOs, and format and status of the SSA and FPRA.

The licensee stated that it originally convened the MSO expert panel in October 2009 using the guidance in NEI 00-01, Revision 2, and that in October 2011, Revision 3 to NEI 00-01 was issued, and that this revision contained updates to the BWR generic MSO list, including nine new scenarios and many clarifications to existing scenarios. The licensee stated that it updated the Hatch MSO analysis to address the most current generic list and also to incorporate additional operating experience for MSO identification and resolution, and that to ensure the integrity of the MSO Expert Panel process, it convened a supplemental panel on October 17, 2012, to review the new and clarified scenarios, as well as additional implementing considerations from the original 2009 panel.

For Step 3, the licensee stated that the NSCA and FPRA analyses were updated in the Transition Process to reflect the treatment of applicable MSO scenarios and that this included the identification of equipment, identification of cables, and the routing of cables by plant locations. The licensee further stated that MSOs were examined for applicability to the FPRA and any differences or adjustments to the expert panel conclusions are discussed in its analyses.

For Step 4 the licensee stated that the MSO combination components of concern were also evaluated as part of the Hatch NSCA and that for cases where the pre-transition MSO combination components did not meet the deterministic compliance, the MSO combination components were added to the scope of the fire risk evaluations.

For Step 5, the licensee stated that results are documented in its analyses.

The NRC staff reviewed the licensee’s expert panel process for identifying circuits susceptible to MSOs, as described above, and concludes that the licensee adopted a systematic and comprehensive process for identifying MSOs to be analyzed using available industry guidance. Furthermore, the NRC staff concludes that the process used provides reasonable assurance that the FREs appropriately identify and include risk-significant MSO combinations. The NRC staff concludes that the licensee’s approach for assessing the potential for MSO combinations is acceptable.

3.2.5 Establishing Recovery Actions

Section 1.6.52 of NFPA 805, “Recovery Action,” defines a RA as:

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

Section 4.2.3.1 of NFPA 805, 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 4.2.4.

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

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

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

As described in Section 4.2.1.3 of the LAR, the licensee's process is based on the methodology of FAQ 07-0030 (Reference 46), and consisted of the following steps:

- Step 1: Clearly define the PCSs and determine which pre-transition operator manual actions (OMAs) are taken at PCSs (activities that occur in the MCR are not considered pre-transition OMAs). Activities that take place at PCSs or in the MCR are not RAs by definition.
- Step 2: Determine the population of RAs that is required to resolve variances from deterministic requirements (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.

Any OMAs meeting the definition of an RA are required to comply with the NFPA 805 requirements outlined above. Some of these OMAs may not be required to demonstrate the "availability of a success path" in accordance with Section 4.2.3.1 of NFPA 805 but may still be required to be retained in the RI/PB FPP because of DID considerations described in NFPA 805, Section 1.2.

In Attachment G of the LAR, as supplemented, the licensee identified which components were considered PCSs. The licensee further stated that it evaluated the set of RAs necessary to demonstrate the availability of a success path for NSPC per NEI 04-02, FAQ 07-0030, RG 1.205 and RG 1.174 and found that no credited RAs have an adverse impact on the FPRA. The licensee further stated that the feasibility criteria used in the licensee's assessment process were based on the criteria listed in NFPA 805, Appendix B.5.2, and NEI 04-02, Rev. 2. The criteria include the following:

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

- When RAs are necessary in the fire area under consideration, the analysis should demonstrate that the area is tenable for the actions to be performed and that fire or fire suppressant damage will not prevent the RA from being performed.

- The lighting should be evaluated to ensure sufficient lighting is available to perform the intended action.

- Walkthrough of operations guidance (modified, as necessary, based on the analysis) should be conducted to determine if adequate manpower is available to perform the potential RAs within the time constraints (before an unrecoverable condition is reached).

- The communications system should be evaluated to determine the availability of communication where required for coordination of RAs.

- Evaluations for all actions, which require traversing through the fire area or an action in the area of the fire, should be performed to determine acceptability.

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

- There should be a sufficient number of essential personnel to perform all of the required actions in the times required based on the minimum shift staffing. The use of essential personnel to perform actions should not interfere with any collateral industrial fire brigade or CR duties.

- Any tools, equipment, or keys required for the action should be available and accessible. This includes consideration of self-contained breathing apparatus and personal protective equipment, if required.

- Training - should be provided on the post-fire procedures and implementation of the RAs.

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

The licensee stated each of the feasibility criteria in FAQ 07-0030 were assessed for the RAs identified in Table G-1. This licensee further stated that all the risk benefit and DID RAs provided by the Fire Risk Evaluation calculation meet the feasibility criteria stipulated in

NFPA 805, Appendix B.5.2(e) and NEI 04-02, Revision 2. The licensee recommends the following to formally document the post-transition RA feasibility:

- Revise fire response procedures to ensure guidance is provided to operators for all credited RAs and actions for DID, with consideration for the Time Critical and Time Sensitive Action program, as applicable.

In evaluating the reliability of RAs, the licensee stated that the RAs that were modeled specifically in the FPRA were addressed using FPRA methods. The licensee's HRA addressed the reliability of these RAs and evaluated RAs depending on whether they correspond or not to MCR abandonment situations. Since actions taken at PCSs are not RAs, no independent reliability evaluation is required. Results of the reliability assessments in the HRA demonstrate that all credited NFPA 805 RAs are reliable.

Based on the above, the NRC staff concludes that the licensee followed the endorsed guidance of NEI 04-02 and RG 1.205 to identify and evaluate RAs in accordance with NFPA 805. Therefore, the NRC staff concludes that there is reasonable assurance of meeting the regulatory requirements of 10 CFR 50.48(c). The NRC staff concludes that the feasibility criteria applied to RAs are acceptable based on conformance with the endorsed guidance contained in NEI 04-02, subject to successful completion of LAR Table S-3, as supplemented, Implementation Item IMP-11.

3.2.6 Plant-specific Treatments or Technologies

The licensee credited Promat H Board raceway enclosures as ERFBS to meet the separation criteria for deterministic and risk reduction for PB compliance approaches. The licensee stated in LAR Attachment A that it performed a technical evaluation and determined that the Promat ERFBS complies, with no additional clarification, with the requirements of NFPA 805, Section 3.11.5. Based on its review, the NRC staff concludes that the use of the plant-specific ERFBS technology at Hatch is acceptable because the licensee has performed the assessment to ensure the ERFBS met the requirements of NFPA 805, Chapter 4.

3.2.7 Conclusion for Section 3.2

The NRC staff reviewed the licensee's LAR for conformity with the requirements contained in NFPA 805, Section 2.4.2, regarding the process used to perform the NSCA. The NRC staff concludes that the declared safe and stable condition proposed is acceptable and that the licensee's process is adequate to appropriately identify and locate the systems, equipment, and cables required to provide reasonable assurance of achieving and maintaining the fuel in a safe and stable condition, as well as to meet the NFPA 805 NSPC.

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

Subject to completion of LAR Table S-3, as supplemented, Implementation Item IMP-11, the NRC staff concludes that the process used by the licensee to review, categorize, and address

RAs during the transition is consistent with RG 1.205 and the NRC-endorsed guidance contained in NEI 04-02. Therefore, the information provided by the licensee provides reasonable assurance that the regulatory requirements of 10 CFR 50.48(c) and NFPA 805 for NSCA methods are met.

3.3 Fire Modeling

The NFPA 805 (Reference 3) 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 FRE are presented as two different approaches for PB compliance, the FRE approach generally involves some degree of FM to support engineering analyses and fire scenario development. NFPA 805, Section 1.6.18, defines a fire model as a “mathematical prediction of fire growth, environmental conditions, and potential effects on SSC, based on the conservation equations or empirical data.”

Section 4.5.2 of the LAR, “Performance-Based Approaches,” describes how the licensee used FM as part of the transition to NFPA 805 at Hatch, and LAR Section 4.7.3, “Compliance with Quality Requirements in Section 2.7.3 of NFPA 805,” describes how the licensee performed FM calculations in compliance with the NFPA 805 PB evaluation quality requirements for fire protection systems and features to determine whether the FM used to support transition to NFPA 805 is acceptable.

In Section 4.5.2.1 of the LAR, the licensee stated that it used the FM approach for the transition. The licensee used the FRE PB method (i.e., FPRA) with input from FM analyses. Therefore, the NRC staff reviewed the technical adequacy of the FREs, including the supporting FM analyses, as documented in Section 3.4.4 of the SE, to evaluate compliance with the NSPC.

The licensee did not propose any FM methods to support PB evaluations in accordance with NFPA 805, Section 4.2.4.1 as the sole means for demonstrating compliance with the NSPC. Therefore, the NRC staff concludes that there are no plant-specific FM methods acceptable for use to support compliance with NFPA 805, Section 4.2.4.1 for supporting the transition to NFPA 805.

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)

Section 2.4.4 of NFPA 805, provides risk acceptance criteria, DID requirements, and safety margins requirements for PCEs. 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.

Section 3.4.1 of the SE 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 PCE ensure that the philosophy of DID is maintained relative to fire protection and nuclear safety.

Section 3.4.2 of the SE 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 PCE ensure that sufficient safety margins are maintained.

Section 3.4.3 of the SE 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.
Section 3.4.4 of the SE 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.

Section 3.4.5 of the SE 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.

Section 3.4.6 of the SE 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.

Section 3.4.7 of the SE 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.

Section 3.4.8 of the SE 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 (Reference 8), as supplemented to determine whether the principles of DID will be maintained for the proposed transition to NFPA 805 at Hatch. Section 2.4.4.2 of NFPA 805 requires that the PCE 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 Section 2.0 of the SE), 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 SSC 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 Section 4.5.2.2 of the LAR, "Fire Risk Approach" and provided additional details in its response to PRA request for additional information (RAI) 17 (Reference 9). The licensee followed guidance contained in NEI 04-02, on consideration of DID as part of the change process. The licensee's method for evaluating DID addressed each of the three elements in NFPA 805, Section 1.2 referred to as echelons 1, 2, and 3, respectively. In its response to PRA RAI 17, the licensee described the evaluation of each of the three DID echelons and the criteria used in those evaluations to identify fire protection features and actions that needed to be credited for DID. The licensee's evaluation determined whether there was an overreliance on an echelon of DID and whether changes were needed to assure that an echelon of DID had been satisfactorily achieved. Many of the identified fire protection features are required to be in place in order to demonstrate compliance with the fundamental FPP and design elements of NFPA 805 Chapter 3, (e.g., the combustible material control and hot work control programs). The licensee also indicated that DID attributes come from FPRA and the NSCA considerations. Regarding the three echelons (i.e., Echelon 1 – Prevent fire from starting, Echelon-2 – Rapidly detect, control, and extinguish fire to limit fire damage, and Echelon 3 – Provide fire protection for SSC so that safety functions are performed), the licensee explained that the criteria for DID included quantitative criteria for defining the term "potentially risk-significant." For Echelon 2, the licensee stated that if VFDRs were never affected in "potentially risk significant" fire scenarios then it considered manual suppression adequate without crediting other systems. For Echelon 3, the licensee stated that if VFDRs were never affected in "potentially risk significant"

fire scenarios then it considered internal fire separation adequate without crediting RAs. The licensee stated that for DID, it defined “potentially risk-significant” to be:

- A scenario with a CDF > $1x10^{-7}$ per year and/or LERF > $1x10^{-8}$ per year, or
- A scenario with a CDF between $1x10^{-6}$ and $1x10^{-8}$ per year or LERF between $1x10^{-7}$ and $1x10^{-9}$ per year, and where DID Echelon 1 and 2 attributes are causing a significant reduction in risk, or
- A scenario with a conditional core damage probability (CCDP) > $8x10^{-2}$ to 1.0 and/or $7x10^{-2}$ to 1.0.

In its response to PRA RAI 17.b (Reference 9), the licensee stated that at the conclusion of its review no additional attributes were required to be credited as DID in LAR Attachment C due to an imbalance between echelons beyond DID attributes credited for the NSCA, licensing actions or engineering evaluations. As such, the NRC staff concludes that the information the licensee developed as part of the FRE and provided in the LAR, is considered the licensee’s internal record of the systems required to meet the NSPC and DID requirements of NFPA 805.

Based on the information provided by the licensee, the NRC staff concludes 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

Section 2.4.4.3 of NFPA 805, requires that the PCE ensure that sufficient safety margins are maintained. Consistent with RG 1.174 (Reference 18), the guidance in Section 5.3.5.3, “Safety Margins,” of NEI 04-02 (Reference 7), 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 it based the FRE methodology on the requirements in NFPA 805 and guidance in RG 1.205 and NEI 04-02. Section 4.5.2.2 of the LAR describes how the licensee addressed safety margins as part of the FRE process. The licensee performed an FRE 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.

In its response to PRA RAI 17 (Reference 9), supported by information in the LAR including LAR Attachment H, “NFPA 805 Frequently Asked Question Summary Table,” and LAR Attachment J, “Fire Modeling V&V,” the licensee described the methodology it used to evaluate safety margins in the FREs to include the following evaluations and determinations:

- The licensee reviewed FM for the FPRA for adequate safety margin and, in general, utilized industry and NRC guidance, including NUREG/CR-6850 (Reference 25), (Reference 26), and (Reference 27), NEI 04-02 (Reference 7), and associated FAQs

resolutions as described in Section 3.4 of the LAR, “NFPA 805 Frequently Asked Questions,” and specifically identified throughout the LAR. Verification and validation, performed in support of FM, utilized accepted codes and standards and the state of knowledge and uncertainties were considered in applying the FM tools.

- The licensee evaluated plant system performance given the specific demands associated with the postulated fire events. The methods, input parameters, and acceptance criteria utilized in the RI/PB analysis were reviewed against the plant design-basis events. This evaluation determined the safety margin established in the plant design basis events was preserved. Codes and standards that were used to determine the plant system performance were those acceptable to NRC.

- The licensee developed the FPRA logic model, including supporting FM, in accordance with NUREG/CR-6850 and ASME/ANS RA-Sa-2009 (Reference 20).

The results of the licensee’s safety margin assessment by fire area are provided in LAR Attachment C, Table C-1, as supplemented.

The NRC staff concludes that the safety margin criteria used by the licensee is acceptable because it is based on the requirements in NFPA 805 and guidance in RG 1.205 and NEI 04-02, which are consistent with the safety margin guidance in RG 1.174. Based on the information provided by the licensee, 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 PRA quality review is to determine whether the plant-specific PRA used in evaluating the proposed LAR is of sufficient scope, level of detail, and technical adequacy for the application. The NRC staff evaluated the PRA quality information provided by the licensee in its NFPA 805 submittal, as supplemented, including industry peer review results and self-assessments performed by the licensee. The NRC staff reviewed LAR Section 4.5.1, “Fire PRA Development and Assessment,” LAR Section 4.7, “Program Documentation, Configuration Control, and Quality Assurance,” LAR Attachment C, “NEI 04-02 Table B-3 – Fire Area Transition,” LAR Attachment U, “Internal Events PRA Quality,” LAR Attachment V, “Fire PRA Quality,” and LAR Attachment W, “Fire PRA Insights,” as well as 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 (IEPRA) model to capture the effects of fire.

In LAR Section 4.8.2, “Plant Modifications and Items to be Completed During the Implementation Phase,” the licensee stated that no plant changes (beyond those identified and scheduled to be implemented as part of the transition to a FPP based on NFPA 805) are outstanding with respect to their inclusion in the FPRA model.

3.4.3.1 Internal Events PRA Model

The licensee evaluated the technical adequacy of its IEPRA model, which is used to support development of the FPRA model, by having a full-scope IEPRA peer review performed in

November 2009 and a fact and observation (F&O) closure review performed on all open finding-level F&Os in April 2017. The full-scope IEPRA peer review was conducted using the process described in NEI 05-04, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," Revision 2 (Reference 69), against requirements in PRA standard ASME/ANS RA-Sa-2009 and RG 1.200, Revision 2 (Reference 19). RG 1.200, Revision 2, endorses PRA standard ASME/ANS RA-Sa-2009 with clarifications. The F&O closure review process detailed in "NEI 05-04/07-12/12-06 [sic][5], Appendix X: Close-Out of Facts and Observations (F&Os)," (Reference 70), to the guidance in NEI 05-04, NEI 07-12, "Fire Probabilistic Risk Assessment Peer Review Process Guidelines," Revision 1 (Reference 71), and NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," (Reference 72), is the process used to "Close Out of Facts and Observations." The NRC staff accepted, with conditions, a final version of Appendix X in the letter dated May 3, 2017 (Reference 73). The NRC staff reviewed the IEPRA F&O closure report from the April 2017 review and concludes that it aligned with the guidance in NRCs May 3, 2017 letter.

For PRA standard Supporting Requirements (SR), there are three degrees of "satisfaction" referred to as capability categories (i.e., CC-I, CC-II, and CC-III), with CC-I being the minimum, CC-II considered widely acceptable, and CC-III indicating the maximum achievable scope/level of detail, plant specificity, and realism. For many SRs, the capability categories (CCs) are combined (e.g., the requirement for meeting CC-I may be combined with CC-II, or the requirement may be the same across all CCs so that the requirement is simply met or not met. As part of the F&O closure review process each SR associated with an open finding is evaluated to CC-II. Also, as part of the F&O closure review process, the PRA model changes needed to resolve a finding are evaluated to determine whether the PRA model change constitutes a PRA upgrade. The bases for these determinations are documented in the F&O closure review report using the definition for a "PRA upgrade" in the PRA standard ASME/ANS RA-Sa-2009.

In Attachment U of the LAR, the licensee stated that the independent assessment (IA) team determined that the resolutions to open F&Os were maintenance updates (as opposed to PRA upgrades) and therefore, no focused-scope peer reviews were identified as being needed as a result of the F&O closure review.

Attachment U of the LAR, Table U-1, provides the licensee's dispositions to four finding-level F&Os that remained open at the end of the F&O closure review process. For each F&O, the licensee provided a basis for why the uncompleted resolution of the findings do not impact the NFPA 805 application. NRC staff evaluated each disposition presented in LAR Attachment U for significant impacts to the application. In general, the licensee provided sufficient information for NRC staff to conclude that final resolution of the open F&Os would not impact the NFPA 805 application.

The NRC staff has reviewed the open finding-level F&Os provided in the LAR along with the licensee's dispositions and determined that the resolution of the findings supports the determination that the quantitative results are adequate or have no significant impact on the FPRA. The NRC staff reviewed the licensee's IEPRA against the applicable SRs in ASME/ANS RA-Sa-2009 and concludes that the licensee's IEPRA 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 IEPRA is adequate and can be used to support the FPRA.

---

5 The title to Appendix X includes NEI 12-06 which is incorrect, it is supposed to be NEI 12-13.

3.4.3.2 Fire Probabilistic Risk Assessment Model

The licensee evaluated the technical adequacy of the FPRA model by having a full-scope peer review performed in May 2016, an F&O closure review performed on all open finding level F&Os in October through November 2017, and a focused-scope peer review performed concurrently with the F&O closure review. The full-scope FPRA peer review was performed using the process described in NEI 07-12 against the requirements in PRA standard ASME/ANS-RA-Sa-2009 and RG 1.200, Revision 2. The F&O closure review process detailed in "NEI 05-04/07-12/12-06 [sic], Appendix X: Close-Out of Facts and Observations," to the guidance in NEI 05-04, NEI 07-12, and NEI 12-13, is the process used to "Close Out of Facts and Observations." The NRC staff accepted, with conditions, a final version of Appendix X in the letter dated May 3, 2017 (Reference 73). In its response to PRA RAI 01 (Reference 9), the licensee clarified that it used guidance in its plant procedures providing explicit criteria consistent with criteria and examples from Appendix 1-A of the ASME/ANS RA-Sa-2009 PRA standard to characterize each FPRA F&O resolution as a maintenance update or PRA upgrade. The licensees stated that it provided this assessment to the F&O closure IA team. The licensee stated that the IA team determined that the resolution of one F&O (i.e., 201-16) constituted a PRA upgrade, and therefore, it performed a concurrent focused-scope peer review to review the new method used to calculate time-to-cable damage caused by fire associated with the resolution of F&O 201-16. The licensee stated that no additional F&Os were issued as a result of the focused-scope peer review.

In Attachment V of the LAR, "Fire PRA Quality," the licensee stated that the IA team reviewed and closed all F&Os. This included closure of F&Os for 13 SRs that were considered not to be met and three SRs determined to be met at only CC-I by the full-scope peer review in April 2016.

The NRC staff identified that there were updates made to the IEPRA to resolve F&Os that appeared to be relevant to the FPRA's underlying plant response model. The NRC staff requested confirmation that the applicable IEPRA model updates that were performed to resolve finding-level F&Os ahead of the internal event F&O closure review were also performed for the FPRA model used to determine the fire estimates for the NFPA 805 LAR. In its response to PRA RAI 02 (Reference 9), the licensee stated that applicable IEPRA model updates that were performed to resolve finding-level F&Os for the IEPRA F&O closure review were also performed for the FPRA. The licensee further stated that an exception to this were four F&Os that were not fully closed as part of the F&O closure process. However, as stated in the previous SE Section 3.4.3.1, the NRC staff reviewed the dispositions to each of these four F&Os and concluded that final resolution of open F&Os would not impact the NFPA 805 application.

Although the licensee stated in LAR Attachment V that the full-scope peer review identified "0 unreviewed analysis methods (UAMs)," the NRC staff reviewed the licensee's analysis for use of unacceptable methods and as a result requested further information on certain approaches. A discussion of issues from that review of methods is provided in the remainder of this SE section.

The NRC staff requested that the licensee identify methods used in the PRA that deviate from guidance in NUREG/CR-6850 or other accepted guidance. In its response to PRA RAI 04 (Reference 9), the licensee stated that "[t]here were no methods used in the development of the Hatch FPRA that deviated from the guidance provided in NUREG/CR-6850 or other NRC acceptable guidance." Based on the guidance in RG 1.200 to review key assumptions, the NRC staff reviewed the licensee's analysis using generic FPRA topics, and as a result

requested further information on certain approaches which are discussed in the remainder of this SE section. In its response to PRA RAI 03 (Reference 12) and (Reference 13), the licensee provided an update of the FPRA results in an aggregate analysis in which the licensee incorporated changes to the PRA model to resolve RAIs. Based on the above, the NRC staff notes that the licensee used NRC-approved guidance and, therefore, concludes that the licensee used acceptable methods.

The NRC staff identified that the licensee's detailed FM for certain fire areas credited a lower than 98% HRR for transient fires which is less than the guidance in NUREG/CR-6850 for transient fire HRRs; therefore, the NRC staff requested that the licensee justify the reduced HRR using factors laid out in the letter dated June 21, 2012 memo from Joseph Giitter, U.S. NRC, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and EPRI 1022993, Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires," (Reference 74). In its response to PRA RAI 06 (Reference 9), the licensee indicated that it credited a 98 percent HRR of 65 kW in fire zone 0501 (intake structure), fire zones 1104 and 2104 (the East Cableway of Unit 1 and 2, respectively), and in fire zone 0024A cable spreading room (CSR). The licensee explained that each of these four fire zones are classified as level A transient combustible control areas and per administrative procedures unattended combustible materials are not allowed in these locations except for short periods of time up to one hour. The licensee noted that all management of transient combustibles in these controlled areas requires special permitting from the work control center and that permits control the type, use, and removal of transient material as well as compensatory measures that may be needed while maintenance occurs. The licensee also stated that three of the fire zones (i.e., fire zones 0501, 1104 and 2104) require a continuous fire watch if combustible materials are stored unattended for a short period, and that fire zone 0024A in the CSR does not require a continuous fire watch but has low occupancy and traffic flow. Additionally, the licensee stated that all four areas have either signage or large red arrows painted on the floor indicating that no storage is allowed within the fire zone. Based on the above, the NRC staff concludes that the licensee's use of a reduced HRR in four fire zones is acceptable because the reduced HRR is commensurate with the high level of transient combustible administrative controls applied to these fire zones, and because there have been no examples that have occurred since the controls were implemented that could have resulted in a fire exceeding 65 kW.

The NRC staff noted that the licensee's treatment of sensitive electronics did not appear to be consistent with the NRC guidance in FAQ 13-0004, "Close-out of Fire Probabilistic Risk Assessment Frequently Asked Question 13-0004 on Clarifications Regarding Treatment of Sensitive Electronics," (Reference 55). The NRC staff requested that the licensee provide a description of its treatment of sensitive electronics and explain whether the treatment is consistent with the guidance in FAQ 13-0004, including any caveats about configurations that can invalidate the approach (i.e., sensitive electronic mounted on the surface of cabinets and the presence of louver or vents). In its response to PRA RAI 07 (Reference 9), the licensee explained that consistent with the guidance in FAQ 13-0004, it treated externally mounted sensitive electronics identified during FPRA walkdowns in the FPRA using the lower damage threshold heat flux of 3kW/m2. The licensee noted that it did not use the guidance in FAQ 13-0004 to treat internally mounted sensitive electronics in the FPRA, rather it used a screening approach and a sensitivity study instead. The licensee explained that it first screened electrical cabinets with FPRA cables based on functions not supported by sensitive electronics such as switchgear, MCCs, and distribution cabinets; then it screened cabinets based on: (1) locations modeled using bounding room scenarios with no development of non-suppression probabilities, (2) large open locations such as in the Turbine building where the temperature increase from a fire associated with another source is not a concern, and (3) cabinets that are

part of scenarios in which the fire risk is already bounded by the fire initiating event (e.g., loss of a control panel is bounded by assumed reactor trip). The licensee explained that after the screening process, it evaluated a few remaining cabinets in a sensitivity study and these were shown to result in an increase in CDF of less than 1×10^{-8} . The NRC staff requested confirmation that cabinets screened based on their function do not house any sensitive electronics and that cabinets screened because their failures are already included as contributing to plant trips do not also impact plant shutdown. In its response to PRA RAI 07.01 (Reference 11) and PRA RAI 03.b.01 (Reference 12) and (Reference 13), the licensee explained that it modified its approach and did not screen internally mounted sensitive electronics based on function (e.g., switchgear or MCC function) in the FPRA used in the aggregate analysis. The licensee indicated that for this set of internally mounted sensitive electronics, it applied the lower damage threshold heat flux of $3 kW/m^2$ as was done for the externally mounted sensitive electronics. Also, in its response to PRA RAI 07.01.b (Reference 11), the licensee explained that it only screened sensitive electronics whose fire-induced failure lead to a plant trip if they do not also affect SSD of the plant. Based on the above, the NRC staff concludes that the licensee's treatment of sensitive electronics is acceptable because, consistent with the guidance in NUREG/CR-6850 and FAQ 13-0004, the licensee treated externally mounted and certain internally mounted sensitive electronics using the lower damage threshold heat flux of $3 kW/m^2$ and the remainder of the internally mounted sensitive electronics were determined to have a minimal impact on the application.

The NRC staff noted that a transient combustible control violation identified in a Condition Report (CR) appeared to have the potential to affect the influence factors assigned to the CSR. The NRC staff requested that the licensee review CRs for all violations in Units 1 and 2 (associated with hot work and transient fires) and evaluate the impact of these violations on the influence factors used in the FPRAs. In its response to PRA RAI 08 (Reference 9), the licensee stated that it identified transient combustible and hot work violations using a key word search on the terms: "transient," "combustible," "housekeeping," "hot work," and "improper," for the January 2017 to March 2019 time interval (which corresponds to the time since the licensee implemented the new transient combustible controls program). The licensee presented a table of these violations showing that nearly all the violations occurred where some storage of combustible is allowed and in areas where a medium or high maintenance and/or storage influencing factor is assigned. The licensee discussed the four non-outage events that occurred in areas with low maintenance and storage influencing factors assigned to them. In the CSR, which is assigned a very low maintenance and storage influencing factor, a 1.5 ft. wood two-by-four was found. The licensee explained that the CSR is designated a Level A transient combustible area requiring a permit, so that combustibles are not left unattended except for short periods of time up to an hour. The licensee stated that the violation did not represent a long term or temporary-storage of material but rather an inadvertent situation that was detected. The licensee also discussed three other events that occurred in fire areas assigned a low storage or/and maintenance influencing factor. The NRC staff requested (1) justification for why the discovery of transient combustible material in the CSR does not constitute a violation that would require applying a higher maintenance influencing factor to the CSR, (2) confirmation that for three areas where violations have occurred no combustible/flammable materials are stored by practice, (i.e., fire zone 2205B – SE Corner Pump room, fire zone 1105 – East Cableway Foyer, and fire zone 2103 – SE Stairwell), and (3) confirmation that a performance monitoring program is in place that demonstrates the administrative control program is meeting expectations and objectives. FAQ 12-0064, "Close-out of NFPA 805 Frequently Asked Question 12-0064: Hot Work/Transient Fire Frequency Influence Factors," (Reference 54) use of a very low influencing factor requires that no violations have occurred in reasonable period, use of a low storage rating is to be used in an area where no combustible/flammable material is

stored by practice, and when applying a low storage or maintenance where violations have occurred, a performance monitoring program is required. In its response to PRA RAI 08.01 (Reference 11), and PRA RAI 03 (Reference 12), the licensee stated that the cited influencing factors were modified to be consistent with the guidance in FAQ 12-0064 for the FPRA used in the aggregate analysis. The licensee further stated that it assigned a medium storage influencing factor to the CSR and the other three cited fire zones because of the violations that have occurred. Based on the above, the NRC staff concludes that the licensee's use of transient influencing factors is consistent with NRC guidance in FAQ 12-0064 including consideration of the violations reported in CRs.

The NRC staff requested that the licensee explain the minimum joint Human Error Probability (HEP) values used in the FPRA. Also, consistent with guidance in NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," (Reference 36), and NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA)" (Reference 75), the NRC staff requested justification for any cases in which a minimum joint HEP of less than 1E-05 was used. In its response to PRA RAI 09 (Reference 9), and PRA RAI 03 (Reference 12), the licensee explained that it updated the FPRA to use a minimum joint HEP value of 1×10^{-5} consistent with guidance in NUREG-1792 and that this update was reflected in the results of the aggregate analysis performed in response to PRA RAI 03. Based on the above, the NRC staff concludes that the licensee's response is acceptable because it used a minimum joint HEP value that is consistent with NRC guidance.

The NRC staff determined that the FM in support of the FPRA appears to use the guidance from NUREG-2178, Volume 1, (Reference 76), though it was not clear whether the licensee followed the guidance on modeling the effect of an obstructed plume. In its response to PRA RAI 10 (Reference 9), the licensee explained that it assumed in the detailed single compartment FM that the base of the fires for fixed ignition sources occurred at the top of the electrical cabinet. The licensee stated that there were no instances for which credit was taken for obstructed plume modeling where the base of the fire was assumed to be located at an elevation of less than one-half the cabinet height. Based on the above, the NRC staff concludes that the licensee's treatment of obstructed plume modeling is acceptable because it is consistent with NRC guidance.

The NRC staff identified that the main control board (MCB) "horseshoe" in the main control room (MCR) and the cabinets on the backside of the "MCB horseshoe" appear to be part of a single MCB enclosure consistent with the definition in FAQ 14-0008, "Close-out of Fire Probabilistic Risk Assessment Frequently Asked Question 14-0008 on Main Control Board Treatment" (Reference 57). It also appeared that the licensee credited a fire damage delay of 15 minutes in MCB fire scenarios due to the presence of solid barriers between MCB cabinets even though many of the MCB cabinets have open backs. The NRC staff requested that the licensee describe the MCB configuration and provide justification for how it modeled the cabinets in the "MCB horseshoe" and cabinets behind the "horseshoe." In its response to PRA RAI 11 (Reference 9), the licensee explained that the front panels that constitute the MCB horseshoe are not connected to the panels behind them. The licensee explained that these front panels and panels behind them are separated by a narrow walkway but do not constitute a single enclosure as defined in FAQ 14-0008. The license explained that there is open space above the top of the front and back panels to the ceiling of the MCR. The NRC staff concludes that because there is no MCB cabinet ceiling above the walkway or cables that run from the front to the back panels across the walkway, that the configuration at Hatch does not meet the criteria to be considered a single MCB enclosure as described in FAQ 14-0008. The license further explained that most of the cabinets accessible to the walkway have exposed open backs

and therefore, fires originating in the back panels were postulated to damage cables in the front panels across the walkway if the heat flux criterion is exceeded. In addition, the licensee stated that transient fires were postulated in the walkway leading to damage in both the back and front panels. The licensee explained that the front panels (i.e., the MCB panels) were treated using the guidance in Appendix L of NUREG/CR-6850, and that when applying Appendix L of NUREG/CR-6850 to the MCB that it did not credit a 15-minute delay for MCB cabinets with open backs, rather, it used a 10-minute delay in fire propagation and damage to credit steel-wall partitions between certain MCB cabinets consistent with guidance in Appendix S of NUREG/CR-6850 for fire spread across a single steel barrier.

Based on the above, the NRC staff concludes that the licensee's FM of the MCB and cabinets behind the MCB is acceptable because it is consistent with NRC guidance in FAQ 14-0008 and Appendices L and S of NUREG/CR-6850.

The NRC staff noted that the LAR did not describe how MCR abandonment scenarios due to loss of habitability (LOH) were modeled to account for the complexity associated with addressing the full range of fire impacts that can occur from fires in the MCR and the subsequent actions required for safe alternate shutdown. The NRC staff also identified that according to LAR Attachment W, Table W-2 MCR abandonment scenarios due to LOH are among the top risk contributors, therefore, the NRC staff requested that the licensee (1) explain how the CCDPs and the conditional large early release probabilities (CLERPS) were estimated, (2) explain how the full range of possible fire impacts were considered, (3) explain the range of CCDP and CLERP values demonstrating the range of MCR fire impacts possible, and (4) explain the range of CDF and LERF values. In its response to PRA RAI 12 (Reference 9), the licensee explained that the plant response for alternate shutdown due to LOH is fully integrated into the FPRA models and that individual failures associated with various actions needed to implement alternate shutdown were developed and integrated into the plant response model. The licensee further stated that development of the HEPs for these failures considered unit-specific considerations such as the complexity associated with the Unit 1 Remote Shutdown Panel (RSP) which consists of four individual panels located in different areas of the plant. The licensee described the required actions and identified the panel in each unit where command and control would be located. The licensee stated that coordination and communication problems were addressed in the FPRA HRA by adding additional timing requirements to the cognition time for certain key actions and by adding communication errors beyond those addressed by the EPRI HRA Calculator Cause Based Decision Tree Method (CBDTM). The licensee further stated that it reflected the full range of fire damage in the calculation of the CCDP and CLERP values for MCR abandonment scenarios due to LOH, and that the CCDP for these scenarios for Unit 1 ranged from $8x10^{-2}$ to 1.0 and for Unit 2 ranged from $7x10^{-2}$ to 1.0. The licensee further stated that the CLERP for Unit 1 ranged from $1x10^{-3}$ to $1x10^{-1}$ and for Unit 2 ranged from $1x10^{-4}$ to $4x10^{-2}$, and that the difference in these ranges between Unit 1 and Unit 2 reflect the difference in complexity in performing alternate shutdown between the units which is accounted for in the HRA using factors for stress. The CDF presented by the licensee for MCR abandonment scenarios due to LOH was $6.7x10^{-6}$ per year for Unit 1 and $6.9x10^{-6}$ per year for Unit 2 and the LERF presented by the licensee was $1.9x10^{-7}$ per year for Unit 1 and $1.8x10^{-7}$ per year for Unit 2. Based on the above, the NRC staff concludes that the licensee's treatment of MCR abandonment scenarios due to LOH is acceptable because the licensee performed detailed integrated modeling of fire impacts and alternate shutdown actions producing fire scenarios ranging from those for which a few functions are impacted to those that produce impacts for which successful shutdown is unlikely.

The NRC staff noted that the LAR did not describe how MCR abandonment scenarios due to loss of control (LOC) were modeled to account for the complexity associated with addressing the full range of fire impacts that can occur from fires in the MCR and the subsequent actions required for safe alternate shutdown. The NRC staff requested that the licensee: (1) identify the locations in the plant where a fire could lead to LOC and locations where alternate shutdown actions are performed that are credited in the FPRA, (2) explain how the full range of fire impacts were considered, (3) describe the range of CCDP and CLERP values demonstrating the range of MCR fire impacts possible, (4) describe the range of CDF and LERF values, (5) explain how command and control is performed and modelled for Unit 1 which has the complex RSP, and (6) explain how the decision to abandon the MCR due to LOC is made and was modeled in the FPRA. In its response to PRA RAI 13 (Reference 9), the licensee explained that alternate shutdown is only used for fires in the MCR, computer room, and CSR and is credited in the FPRA due to LOC for fires in the MCR and CSR. The licensee explained that the plant response for alternate shutdown due to LOC is fully integrated into the FPRA models and that individual failures associated with various actions needed to implement alternate shutdown were developed and integrated into the plant response model. The licensee stated that it reflected the full range of fire damage into the calculation of the CCDP and CLERP values for MCR abandonment scenarios due to LOC. The CCDP for LOC scenarios initiated by fires in the CSR and MCR ranged from $1x10^{-2}$ to 1.0 for Unit 1 and ranged from $1x10^{-2}$ to 1.0 for Unit 2. The CLERP for LOC scenarios initiated by fires in the CSR ranged from $1x10^{-5}$ to $1x10^{-1}$ for Unit 1 and ranged from $5x10^{-5}$ to $3x10^{-1}$ for unit 2, and for fires in the MCR ranged from $1x10^{-5}$ to $1x10^{-1}$ for unit 1 and ranged from $5x10^{-6}$ to $2x10^{-2}$ for Unit 2. The total CDF presented by the licensee for MCR abandonment scenarios due to LOC was $7.1x10^{-6}$ per year for Unit 1 and $1.1x10^{-5}$ per year for Unit 2. The total LERF presented by the licensee was $2.1x10^{-7}$ per year for Unit 1 and $5.5x10^{-7}$ per year for Unit 2. The licensee explained that the shift supervisor will move command and control to the designated primary RSP where the supervisor will maintain control over SSD progress and serve as coordinator. The licensee stated that once command and control has been moved, that operators will communicate via radios, sound powered phones or the plant paging system. The licensee stated that it reflected communication and coordination complexity in the HRA by adding additional timing requirements to the cognition time for certain key actions and by adding communication errors beyond those addressed by the EPRI HRA Calculator CBDTM. The licensee explained that its fire procedures identify fire areas that may require MCR abandonment due to LOC but that ultimately “MCR evacuation is at the discretion of the Shift Supervisor.” The licensee also stated, however, that fire response procedures identify specific fire areas that may require MCR abandonment due to a major fire and identify conditions that are evaluated as cues for LOC (i.e., LOC of containment heat removal capability, LOC of high-pressure injection and emergency depressurization systems, and LOC of low-pressure injection systems). The licensee stated that it specifically modeled failure of the decision to abandon the MCR in the FPRA. Based on the above, the NRC staff concludes that the licensee’s treatment of MCR abandonment scenarios due to LOC is acceptable because the licensee performed detailed integrated modeling of fire impacts and alternate shutdown actions producing a range of fire scenarios in which a few functions are impacted to scenarios in which successful shutdown is unlikely.

The NRC staff requested that the licensee explain: (1) how the risk contribution of fires originating in one unit is addressed for the other unit when fires can impact equipment and cables in the other unit, (2) how the contribution of fires in areas common to both units are addressed, and (3) explain the extent to which systems are shared between units and how these systems are modelled in the FPRA for each unit. In its response to PRA RAI 14 (Reference 9), the licensee explained that fire scenarios were developed and quantified for all unscreened plant analysis units (PAUs) regardless of which unit the ignition source is physically

located. Therefore, scenarios in which a fire originates in one unit and propagates to the other unit are addressed; such as the possibility that fire originates in one unit and propagates to the other unit in a common area. The licensee also explained that the reactor units have common power supplies and dependences which are explicitly modelled in the PRAs. The licensee further explained that the 1B EDG is shared between units but normally aligned to Unit 1 which means that operator actions are modeled to credit the 1B DG for Unit 2. Based on the information provided by the licensee, the NRC staff concludes that the licensee's modeling of dependencies between units is acceptable because the licensee addressed the risk contribution of fires originating in the opposite unit and because the licensee modeled system dependencies.

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

The NRC staff performed detailed reviews of the FM used to support the FREs in order to gain further assurance that the methods and approaches used for the application to transition to NFPA 805 (Reference 3) were technically adequate. NFPA 805 has the following requirements that pertain to FM used in support of the development of the FREs:

NFPA 805, Section 2.4.3.3, "On Acceptability," states that:

> The PSA approach, methods, and data shall be acceptable to the AHJ [authority having jurisdiction].

NFPA 805, Section 2.7.3.2, "Verification and Validation," states that:

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

NFPA 805, Section 2.7.3.3, "Limitations of Use," 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.

NFPA 805, Section 2.7.3.4, "Qualification of Users," states that:

> Cognizant personnel who use and apply engineering analysis and numerical models (e.g., fire modeling techniques) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations.

NFPA 805, Section 2.7.3.5, "Uncertainty Analysis," states that:

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

The following sections discuss the results of the NRC staff's reviews of the acceptability of the first requirement of the FM. The results of the NRC staff's review of compliance with the remaining requirements are discussed in Sections 3.9.3.2 through 3.9.3.5 of the SE.

3.4.3.3.1 Overview of Fire Models Used to Support the Fire Risk Evaluations

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

- Flame Height, Method of Heskestad (Reference 77)
- Plume Centerline Temperature, Method of Heskestad (Reference 77)
- Radiant Heat Flux, Method of Shokri and Beyler (Reference 78)
- Ceiling Jet Temperature, Method of Alpert (Reference 79)

These algebraic models are described in NUREG-1805, (Reference 31). Alpert's ceiling jet temperature correlation is described in FIVE, "EPRI Fire Induced Vulnerability Evaluation Methodology," (Reference 80), and serves as the basis for FDT[s] that are used to estimate sprinkler, smoke detector-and heat detector response times as documented in NUREG-1805, Chapters 10, 11, and 12, respectively. V&V of these algebraic models is documented in NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," (Reference 33)).

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

- CFAST, Version 6.1.1 and Version 7.2.1 (Reference 81)

This HGL correlation is described in NUREG-1805, and its V&V is documented in NUREG-1824, Supplement 1.

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

- Smoke Detection Actuation Correlation (Smoke Optical Density Method) (Reference 79), (Reference 82)
- Heat Detection Actuation Correlation (Reference 83)
- Corner and Wall Effects (Reference 84)
- Correlation for Flame Spread over Horizontal Cable Trays (FLASH-CAT) (Reference 34)
- Stratified Hot Layer Method (Reference 85)

The licensee used the plume radius as the horizontal ZOI where it exceeded the ZOI based on heat flux. The licensee implemented all algebraic fire models and empirical correlations in a workbook referred to as the Fire Modeling Workbook (FMWB).

The licensee used the ZOI approach as a screening tool to distinguish between fire scenarios that required further evaluation and those that did not require further evaluation. Qualified

personnel performed a plant walkdown to identify ignition sources and surrounding targets or SSC in compartments and 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 based its screening on the 98th percentile fire HRR from the NUREG/CR-6850 methodology (Reference 26) and NUREG-2178 (Reference 76).

CFAST, Version 6 and 7, was used for

- HGL temperature calculations
- CR abandonment calculations
- Temperature sensitive equipment HGL study

The V&V of CFAST is documented in NUREG-1824, Volume 5, as well as NUREG 1824, Supplement 1 (Reference 33).

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

<u>NRC Evaluation of FM in Support of the Hatch FPRA</u>

Attachment J of the LAR, as supplemented, includes the model identification, the purpose of the model application, technical references for the model, and the validation work available for the model.

In Attachment J of the LAR, as supplemented, (p. J-7), the licensee discussed the method used to determine an HRR adjustment factor for fires that are proximate to a wall or corner. In its response to FM RAI 01(a) (Reference 9), the licensee explained that for ignition sources located within two feet of a wall or corner, it multiplied the HRR by a modification factor of two, for fires located near a wall, and a factor of four for fires located near a corner. The licensee applied the modification factor in the FMWB, which it then used to perform the Hatch FPRA detailed FM calculations. The licensee indicated that this approach is consistent with the discussion presented in Appendix F of Attachment 3 to the NRC Inspection Manual.

The licensee stated that the ZOI calculations and HGL timing calculations for fires located close to a wall or corner were accounted for in the FMWB by increasing the HRRs. An increase in HRR causes an increase in ZOI in both the horizontal and vertical directions. The licensee further stated that if the expanded ZOI causes the ignition of additional secondary combustibles, then it incorporated the contribution of these burning secondary combustibles into the HRR curves produced using the FLASH-CAT model and into the calculation of the HGL temperature versus time.

The licensee stated that in the MCR abandonment calculations, transient fuel packages in the open, wall, and corner locations were evaluated using CFAST which reflects the condition wherein fires located near a wall boundary or near a corner would experience a reduced air entrainment and a force imbalance on the plume.

Based on the information provided by the licensee, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee's approach to account for wall and corner effects follows the guidelines in Inspection Manual Chapter 0609, Appendix F (Fire Protection Significance Determination Process), and is consistent with the "image" method, which is commonly used to simulate wall and corner fires in zone fire models, such as CFAST.

The NRC staff could not determine whether there are any administrative controls in place to minimize the likelihood of fires involving those electrical cabinets with open doors during maintenance activities in the plant and describe how cabinets with temporarily open doors are treated. In its response to FM RAI 01(b) (Reference 9), the licensee explained that the FM analysis assumes that electrical cabinets with normally closed doors are maintained closed. This assumption is based on existing plant procedures and administrative controls that require compliance with applicable industrial/electrical safety requirements, prevention of foreign material from entering the maintenance area, and proper housekeeping practices related to fire prevention and protection of the equipment. The licensee stated that for the brief period cabinet doors may be temporarily open, administrative procedures provide reasonable assurance that the likelihood of a fire in these cabinets is minimal. The licensee also stated that electrical cabinet doors are unlikely to be left open when maintenance or measurement activities are not in progress, and to ensure this assumption remains valid, the condition of cabinet doors will be included in the monitoring program. Table S-3 of the LAR, as supplemented, describes the implementation items that will be completed prior to the implementation of the new NFPA 805 FPP and the following implementation item has been added to Table S-3, as supplemented:

> Implementation Item IMP-22: Verification of the condition of electrical cabinet doors to meet fire modeling assumptions will be included in the monitoring program.

Based on the above, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee added Implementation Item, IMP-22 to LAR Table S-3, as supplemented, to minimize the likelihood of fires involving those electrical cabinets with open doors during maintenance and measurements activities in the plant.

In LAR Attachment J, as supplemented, (p. J-8) the licensee discussed the method used to predict the growth and spread of a fire within a vertical stack of horizontal cable trays (FLASH-CAT).

In its response to FM RAI 01(c) (Reference 9), the licensee explained that the method for evaluating secondary combustible cable tray HRR profiles used the FLASH-CAT model provided in NUREG-7010. The Hatch FPRA calculated the time to ignition of the secondary combustibles in one of the following two ways: (1) the time to ignition was equal to the time at which the flame height reached the first unprotected cable tray, or (2) it used a conservative time of 1 minute, based on the minimum ignition time for cables as described in Appendix H of NUREG/CR-6850.

The licensee stated that for all Unit 1 fire zones and fire zones that are shared between Unit 1 and Unit 2 (i.e., common fire areas/zones), the FLASH-CAT model used the HRR per unit area (HRRPUA) and horizontal flame spread rates for thermoplastic cables. The licensee noted that for Unit 2 fire zones in which it used FLASH-CAT, the model used the HRRPUA and horizontal flame spread rates for thermoset cables and that per its calculation, 97.39 percent of the cable insulation in Unit 2 could be classified as thermoset. The licensee further stated that employing a weighted average approach, as recommended in NUREG-7010, the HRRPUA would be 152 kW/m^2 (an increase of 2 kW/m^2) and the flame spread rate would be 1.15 m/hour (an

increase of 0.05 m/hr.). The results from sensitivity cases using the weighted average values for the FLASH-CAT analysis showed negligible changes in the results, and therefore, the values for thermoset are appropriate for Unit 2.

Based on the above, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee followed the guidance in NUREG/CR-7010.

In LAR Attachment J, as supplemented, (p. J-10) the licensee discussed the heat soak method used to estimate the time to damage for cables subject to varying time dependent temperatures.

In its response to FM RAI 01(d) (Reference 9), the licensee stated that its procedure describes the methodology employed in the Hatch FPRA for evaluating the time-to-damage for generic cables exposed to a time-dependent temperature of heat flux. It employed the damage integral heat soak method in the same manner as documented in Prairie Island Response to NFPA 805 LAR, PRA RAI 21.01, (Reference 86).

Based on the above, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee provided an appropriate technical basis and V&V to justify the use of the particular heat soak method used at Hatch to determine ignition and damage delays, and also demonstrated that its method appropriately accounts for the effect of preheating. Further, the heat soak method also accounts for the damage accrued at a target over time and when the HRR is not at a steady state.

Based on the information provided by the licensee, it was not apparent to the NRC how non-cable secondary combustibles were identified and accounted for. In its response to FM RAI 01(e) (Reference 9), the licensee stated that the Hatch FPRA identifies areas where it observed significant foam insulation (via walkdowns) and documents the analysis of those scenarios in which the ignition of these non-cable secondary combustibles could result in an increase in the scenario HRR and target impacts of that scenario. The licensee further stated that if the foam insulation was ignited and resulted in an increased HRR that was greater than 15 percent of the primary ignition source, it revised the HRR profile for the scenario to include the contribution of the burning foam, and that due to uncertainty in the HRR values for the foam material, it screened combustible foam sources with estimated HRRs less than 15 percent of the primary ignition source from this analysis. The licensee further stated that with this larger HRR value, it determined the revised (i.e., larger) ZOI for the ignition source scenario, and that if these additional targets resulted in an increase in the fire risk for that scenario, it incorporated this increase into the model results.

The licensee stated that currently no above ground high density polyethylene (HDPE) piping is present at plant Hatch, but that in the future, if any plant change involves installation of HDPE material or other noncable secondary combustibles, as part of the 805 process, the change will be reviewed and the impact on the FPRA will be captured and addressed accordingly.

Based on the information provided by the licensee, the NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that the potential contribution of non-cable intervening combustibles was properly accounted for in the FM analysis and because no above ground HDPE piping is present.

Based on the information provided by the licensee, it was not apparent to how the model assumptions in terms of location and HRR of transient combustibles in a fire area or zone will not be violated during and post-transition. In its response to FM RAI 01(f) (Reference 9), the

licensee explained that it included all of the open floor areas in the fire zone in the transient zone areas which ensures that all possible locations of transient combustibles were considered in the Hatch FPRA. The licensee also stated that the size of a defined transient zone was larger than the size of a ZOI as calculated using FM tools (i.e., as described in Appendix F of NUREG/CR-6850), and that this approach accounted for uncertainties in the specific location of a transient source within a transient zone and ensured that these specific model assumptions would not be violated during or post-transition.

The licensee stated that walkdowns were performed to validate the appropriateness of the HRR chosen for modeling transient fires in that fire zone and that in most cases, it used the 98th percentile peak HRR value of 317 kW. The licensee further stated that for those fire zones with posted combustible controls, it used a reduced peak HRR of 65 kW and that control of combustible materials is governed in accordance with its procedures which requires immediate removal of waste, debris, scrap, packing materials, and other combustibles from an area immediately following the completion of work or at the end of the shift, whichever comes first (NFPA 805, Ch.3, Section 3.3.1.2(3)). The licensee stated that combustible storage and staging areas are designated, and limits are established on the types and quantities of storage materials (NFPA 805, Ch. 3, Section 3.3.1.2(5) and that general housekeeping is maintained in accordance with its procedures.

The NRC staff concludes that the licensee’s response to the RAI is acceptable because the licensee used an acceptable approach to categorize transient combustibles in terms of their nature and HRR characteristics.

In LAR Attachment J, as supplemented, (p. J-5) the licensee discussed the HGL calculations using CFAST that calculates the upper and lower gas layer temperature, interface height, and soot concentration. In its response to FM RAI 01(g) (Reference 9), the licensee explained that it analyzed high energy arcing fault (HEAF) scenarios following the empirical approach described in Appendix M of NUREG/CR-6850 and that HEAFs have two distinct phases: an initial energetic phase (the arcing explosion), and an ensuing fire phase, and it determined the HRRs in HEAF scenarios as follows:

1. The licensee assigned a peak HRR value to the switchgear or load center where the HEAF was postulated following the guidance described in its PRA analysis. It assumed that the peak HRR was achieved at the time of the explosion (i.e., no fire growth phase was credited).
2. The licensee determined the combustibles within the ZOI of the fault and assigned HRRs to those combustibles. No growth profile was credited to combustibles in the ZOI.
3. The licensee assigned HRRs of exposed cable secondary combustibles using the FLASH-CAT model.

The licensee stated that tray covers and fire-resistant wraps that were within the ZOI of the HEAF were assumed to be damaged during initial fault event and were not credited to reduce the area of the cable trays ignited by the fire. Therefore, the HRR curve incorporated into the CFAST model included (1) the peak HRR value of the switchgear or load center, (2) the immediate involvement of trays within the HEAF ZOI, and (3) flame spread through the exposed trays modeled in accordance with FLASH-CAT.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that its analysis to model HEAF-induced fires either followed the NUREG/CR-6850 guidance or is more conservative.

Based on the information provided by the licensee, the NRC staff could not determine the technical justification for the approach the licensee used that makes structural steel scenarios more conservative. In its response to FM RAI 01(h) (Reference 9), the licensee stated that consistent with the ASME/ANS FPRA Standard, it based the risk-relevant scenarios in the structural steel analysis on the following conditions only: (1) exposed structural steel, and (2) a high-hazard fire source was present. The licensee further stated that since it did not perform a detailed structural analysis, this screening analysis assumed that the failure of only one exposed structural steel was sufficient to justify further evaluation of the scenario. The licensee further stated that a high-hazard fire refers to a large and long-duration fire consistent with the clarification note to Table 4-2.6-7 of the ASME/ANS FPRA Standard and that examples of these fires include (1) ignited oil running down to lower elevations of the Turbine Building or (2) fires involving ignition of the entire content of oil reservoirs. The licensee further stated that the evaluation of the scenarios included an assessment of target damage that ranged from damage to targets outside the PAU where the fire originates to collapse of the building in question.

The licensee stated that it did not refine the distance to exposed structural steel (which could be used in a determination of flame impingement or in a heat transfer analysis), rather, it assumed in the screening analysis that any high-hazard fire was capable of damaging exposed structural steel if both were located in the same PAU.

The licensee stated that, consistent with the conservative nature of this study, the quantified structural steel scenarios did not credit manual suppression actions that might have been taken during the fire, and assumed failure of the structural steel immediately upon failure of any available automatic suppression system, and that it did not consider the time required to heat steel up to its structural failure temperature.

The licensee stated that the approach employed in the Structural Steel Scenario Selection Analysis Report should be considered bounding because it results in the inclusion of all fire scenarios, for which a structural collapse is conceivable, without requiring a detailed structural and/or heat transfer analysis.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that the structural steel scenarios were modeled conservatively.

Based on the information provided by the licensee, the NRC staff could not determine how the size of the opening between the exposing and exposed compartments in the CFAST HGL calculations was determined, and whether or not the opening sizes were representative of conditions in the plant. In its response to FM RAI 01(i) (Reference 9), the licensee explained that in the Hatch FPRA Multi-Compartment Analysis (MCA), the horizontal vent between the exposing and exposed fire compartments in the CFAST model was a standard double-doorway (Width = 1.83 m, Height = 2.12 m), and that the double doorway was centered on the wall shared by the two compartments and was entirely open. The licensee further stated that fire effects flowed freely to the exposed compartment once the depth of the HGL exceeded the distance between the ceiling of the exposing compartment and the height of the doorway and that the open doorway was conservative for this multi-compartment application as follows:

1. The open double-doorway provided a less restrictive flow path than the small horizontal vents between the compartments and the “outside”, which were added in the CFAST model to account for typical leakage in compartment construction. This ensured that the majority of the fire effects would migrate through the open door rather than to the outside via the leakage vents.

2. The open double-doorway was a bounding configuration compared to typical plant conditions, because it allowed fire effects to migrate from one compartment to another. Other more typical openings between compartments are usually provided with protective features, such as dampers, closed fire doors, or penetrations seals, and would not allow fire effects to pass between compartments.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee's assumption concerning the size of the opening between the exposing and exposed compartments assumed in the MCA is bounded by the conservatisms in the analysis.

In Attachment J of the LAR, as supplemented, (p. J-5) the licensee discussed the HGL calculations using CFAST that evaluates the time at which CR abandonment is necessary based on smoke obscuration and average HGL temperature. In its response to FM RAI 01(j) (Reference 9), the licensee explained that the MCR abandonment report documents how volume obstructions were subtracted from the gross area volume for each space included in the CFAST. The licensee further noted that the volume obstructions include electrical enclosures and structural components (e.g. columns), and that it assumed conservatively that the volume of the nonstructural obstructions (e.g., ductwork, smaller utilities, and conduits) in the interstitial space to be ten percent of the gross volume of the interstitial space, which was based on field observations and a review of the major ductwork in the areas. The licensee also stated it assumed conservatively the volume of the contents in each restroom (e.g. storage cabinets) to be five percent of the gross restroom volume and that the MCBs are not enclosed (i.e., have open backs), and therefore were not included as a volume obstruction.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee accounted properly for the effective compartment volume reductions in the CFAST MCR abandonment calculations.

In Attachment J of the LAR, as supplemented, (p. J-5) the licensee discussed the HGL calculations using CFAST that evaluates the time at which CR abandonment is necessary based on smoke obscuration and average HGL temperature. In its response to FM RAI 01(k) (Reference 9), the licensee explained that the MCR abandonment calculations assumed that doors between the CR, surrounding spaces, between the electrical equipment rooms, and surrounding spaces remained closed, however, exfiltration and boundary leakage occurred via small gaps and openings in the enclosure boundary. The licensee stated that the effect of this configuration was to maximize the severity of the fire conditions within the CR by limiting leakage of fire effects to locations outside of those rooms.

The licensee stated that it determined boundary leakage from leakage fractions and the internal boundary area of the enclosure, and that the baseline assumption in the abandonment calculation was that the boundary leakage corresponds to tight construction as defined in the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering. The licensee stated that the analysis evaluated the model sensitivity to uncertainty in the boundary leakage fraction and showed that the baseline results were conservative or were not significantly sensitive to large uncertainties in this parameter, and that because the it assumed the boundary leakage to be uniformly distributed, it was defined as a single vent having a height equal to the enclosure height and a width that was determined from the leakage area and vent height.

The licensee stated that adequate airflow to support full development of the postulated fire was provided via mechanical ventilation or via natural ventilation, and that validation studies

confirmed that the scenarios evaluated fall within the equivalence ratio validation range which ensures that the fires analyzed would not be limited by ventilation conditions.

The NRC staff concludes that the licensee's response to the RAI is acceptable, because the licensee's assumptions concerning natural leakage vents are consistent with the SFPE Handbook of Fire Protection Engineering (Reference 82), which is an authoritative publication.

Based on the information provided by the licensee, it was not apparent how raceways, with a mixture of thermoplastic and thermoset cables, are treated in terms of damage thresholds. In its response to FM RAI 02 (Reference 9), the licensee stated that:

(a) The Hatch Nuclear Plant Unit 1 FPRA model used the assumption that all cables within Unit 1 were of thermoplastic type. Cables in fire zones shared by Unit 1 and Unit 2 and raceways in these zones that may contain a mixture of cables types were assumed conservatively to be and modeled as thermoplastic cables. Therefore, thermoplastic damage criteria was used for all Unit 1 cables. Since thermoplastic cables have lower damage thresholds than thermoset cables, assuming all cables are thermoplastic for Hatch Unit 1 is conservative.

(b) A risk-based analysis was conducted to determine the percentage of thermoset cable in Unit 2. The total lengths of Unit 2 cables were extracted from the equipment, raceway and cables database. The cable code associated with each cable was translated to an insulation or jacket type using plant records. The insulation or jacket type was than categorized as either thermoplastic and thermoset. Some of the Unit 2 cables had unknown insulation or jacket materials and were not used in the analysis. Of the Unit 2 cables that were identified as thermoplastic or thermoset, 97.39% was determined to be thermoset, given the thermoset classifications listed in the SNC specification document for cables used at Hatch. Based on this result it was concluded that the thermoset damage criteria for cables is appropriate tor Unit 2.

(c) In accordance with Appendix H.2 of NUREG/CR-6850, for major components such as motors, valves, etc., the fire vulnerability was assumed to be limited by the vulnerability of the power, control, or instrument cables supporting the component. All non-cable components used the damage threshold of the cable type that the associated unit is composed of (i.e., thermoplastic for Unit 1, thermoset for Unit 2).

The licensee further stated that Instrument air lines and pipes, verified by walkdowns, were determined to be carbon steel, stainless steel, or copper with Swagelok, and mechanical connections, given that there are no brazed connections or soldered copper joints, instrument air loss is not postulated due to fire exposure to the air lines.

In its response to FM RAI 02 (Reference 9), the licensee explained that it performed an investigation to classify each of the various cable types as either thermoset or thermoplastic based upon cable jacket or insulation material, and that it combined this information with that in Appendix H of NUREG/CR-6850, Volume 2, to determine the thermal damage criteria for thermoplastic targets.

The NRC staff concludes that the licensee's responses to the RAIs are acceptable because the licensee characterized the installed cabling in Hatch, in accordance with the guidance in

NUREG/CR-6850, Volume 2, and the damage thresholds used in the FM analysis are consistent with the type of cabling installed in each area of the plant.

3.4.3.3.2 Conclusion for Section 3.4.2.3

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 Section 3.9.3.2 of the SE, 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 Probabilistic Risk Assessment Model Quality During Self-Approval

As discussed in Section 2.6 of the SE, 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 Section 2.4.2 of the SE), 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 Section 3.4.3.2 of the SE, the NRC staff concludes that the FPRA model is of sufficient technical adequacy for use to support transition to NFPA 805, subject to the completion of the modifications in Table S-2 and the implementation items in Table S-3 (Reference 12). In Section 4.7.3 of the LAR, "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 IEPRA model. In addition, the licensee has a program for ensuring that developers and users of these models are trained and qualified appropriately. 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, QA requirements is provided in Section 3.9.3 of the SE.

3.4.3.5 Conclusions Regarding Fire Probabilistic Risk Assessment Quality

Based on the information provided by the licensee, the NRC staff concludes that the licensee's PRA is consistent with the guidance in RG 1.174 (Reference 18), Section 2.3, and RG 1.205 (Reference 4), Section 4.3, regarding the technical adequacy of the PRA used to support risk assessment for transition to NFPA 805.

The NRC staff concludes that the PRA model will adequately represent the as-built, as-operated, and as-maintained 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 concludes that the PRA model conforms to the applicable industry standards for IE and FPRAs at an appropriate CC. The NRC staff concludes 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.

The FPRA used to support RI self-approval of changes to the FPP must use an acceptable PRA approach and acceptable methods and data. The NRC staff concludes that the changes already made to the baseline FPRA model to incorporate acceptable methods, as detailed in the response to PRA RAI 03 (Reference 12), and discussed above, subject to completion of all implementation items described in updated LAR Table S-3 (Reference 12), demonstrate that NFPA 805 criteria are satisfied and the PRA is acceptable for use to support self-approval changes to the FPP.

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 8), the licensee stated that it based its FRE methodology on the requirements in NFPA 805 and guidance in Section C.2.2.4 of RG 1.205 and NEI 04-02. The NRC based its SE of the licensee's FREs on information provided in Section 4.5.2 of the LAR, Attachment C of the LAR, Attachment W of the LAR, 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 8), as supplemented, 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 Section 4.2.4 of the LAR, "Fire Area Transition," the licensee indicated that all VFDRs evaluated by the FPRA were categorized as separation issues. The separation related VFDRs can generally be categorized into the following four types of plant configurations: (1) inadequate separation resulting in fire-induced damage of process equipment or associated cables required for the identified success path; (2) inadequate separation resulting in fire-induced spurious operation of equipment that may defeat the identified success path; (3) inadequate separation resulting in fire-induced failure of process monitoring instrumentation or associated cables required for the identified success path; (4) combinations of the above configurations. The NRC staff requested that the licensee discuss the treatment of VFDRs not modelled in the FPRA. In its response to PRA RAI 15.c (Reference 9), the licensee explained that not all VFDRs identified by the NSCA were included in the FPRA for one of three reasons: (1) differences in the system requirements between the best estimate FPRA models and the NSCA system requirements, (2) differences in credited indications between the NSCA and the FPRA for required cues to support PRA credited actions, and (3) conservatisms in the NSCA that resulted in a non-credible failure in the FPRA that have been refined for realism in the FPRA model. Based on the information provided by the licensee, the NRC staff concludes that VFDRs identified in the NSCA but not addressed in the FPRA cannot impact the application given they were not required to be modelled in the FPRA as contributing to CDF or LERF.

In Section W.2.1 of the LAR, "Methods Used to Determine Changes in Risk," and in response to PRA RAI 15.a (Reference 9), the licensee described how the change-in-risk associated with

VFDRs is performed. The licensee explained that the change-in-risk associated with each fire area is obtained by calculating the difference between the CDF and LERF of a compliant plant configuration and the variant (post-transition) plant configuration. The total change-in-risk was obtained by summing the change-in-risk for each fire area. The licensee explained that the types of model adjustments to remove VFDRs from the compliant plant consisted of a bounding evaluation case and a detailed FRE case. The licensee explained that in the bounding evaluation case, the total area post-transition CDF and LERF is used as the change-in-risk for the VFDRs in the area to “bound” the change-in-risk associated with VFDRs. The licensee explained that in the detailed FRE case that basic events associated with a VFDR function (e.g., decay removal or injection) were set to their nominal value in the compliant plant model which has the effect of removing the VFDR from the compliant plant model. In the post-transition plant model, the fire-induced failures due to the VFDRs are modeled. In its response to PRA RAI 15.d (Reference 9), the licensee explained and further clarified in its response to PRA RAI 15.d.01.a (Reference 11), that the risk reduction plant modifications are credited in both the compliant and transition plant. Based on the information provided by the licensee, the NRC staff concludes that the licensee’s treatment of the risk reduction modifications produces a conservative estimate of the change in risk, and therefore the VFDRs and risk reduction modifications are treated properly in the change in risk calculation.

In its responses to PRA RAI 15.b (Reference 9), and PRA RAI 15.b.01.a (Reference 11), for MCR abandonment scenarios due to LOH or LOC, the licensee clarified that the change-in-risk calculations are performed in the same manner as other scenarios except that fire separation is accomplished by crediting the RSP as the alternate shutdown path not impacted by the fire. The licensee stated that the compliant plant model for MCR abandonment is different from the post transition plant model for these scenarios in that local RAs are not needed to recover from fire damage in the compliant model, and accordingly, failures that challenge SSD from the RSP necessitating a RA are considered VFDRs. Based on the information provided by the licensee, the NRC staff concludes that the licensee’s treatment of MCR abandonment scenarios due to LOH or LOC is acceptable because the licensee considers failures that challenge SSD from the RSP necessitating a RA, VFDRs.

The NRC staff requested that the licensee explain the basis for applying a “surrogate” value of $7x10^{-2}$ as the lower bound limit for the CCDP of MCR abandonment scenarios in the compliant plant model, and justification that this approach does not contribute to under estimation of the change-in-risk. In its response to PRA RAI 15.b.01.b (Reference 11), and PRA RAI 03 (Reference 12) and (Reference 13), the licensee explained that it replaced the “surrogate” value used to limit the CCDP for MCR abandonment to $7x10^{-2}$ in the aggregate analysis, with modeling consistent with the fire area risk modeling value used elsewhere. Based on the information provided by the licensee, the NRC staff concludes that this treatment is acceptable because the licensee removed the surrogate value from the FPRA compliant plant model and replaced it with acceptable FPRA modeling.

The NRC staff concludes 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. Based on LAR Attachment W, 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 Attachment C of the LAR, "NEI 04-02, Table B-3 – Fire Area Transition," Attachment G of the LAR, "Recovery Actions Transition," and Attachment W of the LAR, "Fire PRA Insights," during its evaluation of the additional risk presented by the NFPA 805 RAs credited at Hatch. Section 3.2.5 of the SE describes the identification and evaluation of RAs.

The licensee used the guidance in RG 1.205, Revision 1, and FAQ 07-0030 (Reference 46), for addressing RAs, which included the definition of a PCS and RA. Accordingly, any actions required to transfer control to, or operate equipment from the PCS were not considered RAs per the RG 1.205 guidance and in accordance with NFPA 805. Conversely, any OMAs required to be performed outside the CR and not at the PCS were considered RAs.

Attachment G of the LAR, Table G-1, as supplemented, identifies a large number of RAs required to meet risk and DID criteria, and indicated which RAs were required to resolve VFDRs, which RAs were credited to reduce risk, and which RAs were credited for DID. The licensee also identified in LAR Attachment G, Table G-1, as supplemented, those actions taken at the PCS. However, as discussed above, actions taken at the PCS are not considered RAs. In LAR Attachment G the licensee stated that its RSPs for each Hatch unit meet the definition of a PCS provided in RG 1.205 and FAQ 07-0030.

The additional risk of RAs for each fire area is presented in LAR Attachment W, Table W-4 and W-5, as supplemented. LAR Section W.2.1 states that the licensee conservatively determined the additional risk of credited RAs using the post-transition case model by calculating the difference in CDF and LERF with the failure probabilities of RAs associated with VFDRs set to their nominal value versus those RAs set to zero failure probability. This approach is equivalent to calculating the contribution to change-in-risk from scenarios that credit the RAs. This calculation is consistent with guidance in FAQ 07-0030, because the calculation determines the change-in-risk associated with performing the RAs compared to a deterministic resolution of completely resolving the VFDR. In its response to PRA RAI 15.a (Reference 9), the licensee stated that it conservatively assumed the additional risk for fire areas where it used a conservative bounding approach. The NRC staff notes that the additional risk of RAs is not bounded by the total change in risk if non-VFDR related risk reduction modifications are credited only in the post transition case in the change in risk. In its response to PRA RAI 15.d.01 (Reference 11), the licensee clarified that it did not credit plant modifications for a risk decrease in the transition change in risk. Based on the information provided by the licensee, the NRC staff concludes that the licensee's responses to the RAIs are acceptable because the licensee followed the guidance in FAQ 07-0030 and conservatively assumed the additional risk for fire areas where a conservative bounding approach was used.

The total additional risk of RAs was obtained by summing the additional risk of RAs for each fire area. LAR Attachment W, Table W-4 and W-5, as supplemented, states that the total additional risk of RAs for Unit 1 is an increase in CDF of 4.19×10^{-7} per year and an increase in LERF of 1.18×10^{-8} per year, and for Unit 2 is an increase in CDF of 5.13×10^{-7} per year and an increase in LERF of 2.84×10^{-8} per year. Accordingly, the total additional risk of RAs is below the risk acceptance guidelines in RG 1.174, Revision 2, of 1×10^{-5} per year for CDF and 1×10^{-6} per year for LERF.

In Attachment G of the LAR, as supplemented, the licensee reviewed all RAs for adverse impact and resolved each action. None of the RAs listed in LAR Attachment G, Table G-1, as supplemented, was found to have an adverse impact on the FPRA. All RAs listed were evaluated against the feasibility criteria provided in FAQ 07-0030 and RG 1.205. Additionally, regarding the feasibility of RAs, the licensee included an action in LAR Table S-3, Implementation Item IMP-11 to "revise the response procedures to ensure guidance is provided to operators for all credited RAs and actions for DID, with consideration for the Time Critical and Time Sensitive Action program, as applicable." The NRC staff concludes that this action is acceptable because it is an obligation required to be completed by the proposed license condition.

The NRC staff concludes that the licensee's methods for determining the additional risk of RAs are acceptable, because they are consistent with RG 1.205, Section 2.2.4.1 and FAQ 07-0030. Therefore, the NRC concludes that the additional risk of RAs meets the requirements of NFPA-805 Sections 4.2.4 and 2.4.4.1.

3.4.6 Cumulative Risk and Combined Changes

In Table S-2 of the LAR, as supplemented (Reference 12), the licensee identified the modifications it is going to implement for NFPA 805 transition. In its response to PRA RAI 15.d.01 (Reference 11), the licensee clarified that it did not credit plant modifications for a risk decrease in the transition change-in-risk. Therefore, the NRC staff concludes that the licensee's application to transition to a RI/PB FPP is not a combined change request per Section 1.1 of RG 1.174, Revision 2.

The total plant CDF and LERF are in general estimated by adding the risk assessment results for IE, fire, and seismic events. The licensee provided an estimate of the contributors to the total CDF and LERF in LAR Attachment W, Section W.2.1, Table W-1 (Reference 12), as supplemented, which are presented below in SE Table 3.4.6. The risk estimates provided in Table 3.4.6 of the SE show that the total CDF values for both units are below $1x10^{-4}$ per year and the total LERF values for both units are below $1x10^{-5}$ per year.

Table 3.4.6: CDF and LERF for Hatch after Transition to NFPA 805

Hazard Group[1]	**Unit 1**		**Unit 2**	
	CDF (/year)	**LERF (/year)**	**CDF (/year)**	**LERF (/year)**
Internal Events	1.13×10^{-5}	8.99×10^{-7}	8.69×10^{-6}	5.62×10^{-7}
Seismic	1.7×10^{-6}	2.31×10^{-7}	2.01×10^{-6}	2.38×10^{-7}
Fire	5.24×10^{-5}	1.89×10^{-6}	5.38×10^{-5}	3.39×10^{-6}
TOTAL[2]	6.55×10^{-5}	3.02×10^{-6}	6.45×10^{-5}	4.19×10^{-6}

Notes:
1. External hazard risk other than seismic risk was screened as being insignificant.
2. This table reflects the values presented Table W-1 of the final LAR Attachment W dated December 13, 2019.

The licensee provided the ∆CDF and ∆LERF estimates for each fire area that is not deterministically compliant, in accordance with NFPA 805, Section 4.2.3, "Deterministic

Approach." In its response to PRA RAI 03 (Reference 12), the licensee provided change-in-risk estimates in Attachment W of the LAR, Tables W-4 and W-5, as supplemented, based on the final FPRA after implementing some FPRA model and method refinements which use NRC-accepted methods. The risk estimates for these fire areas address the planned modifications and administrative controls that will be implemented as part of the transition to NFPA 805 at Hatch, as well as RAs to reduce VFDR risk. For Unit 1, the total CDF change-in-risk was reported to be 8.12×10^{-6} per year, and total LERF change-in-risk was reported to be 4.88×10^{-7} per year. For Unit 2 the total CDF change-in-risk was reported to be 8.31×10^{-6} per year, and the total LERF change-in-risk was reported to be 5.74×10^{-7} per year. Therefore, the change in CDF and LERF are below the RG 1.174 acceptance guidelines for the corresponding total CDF and LERF. The change-in-risk values presented for each of the individual fire areas are also all less than the total ΔCDF and ΔLERF for each unit, and, therefore, also less than the risk acceptance guidelines in RG 1.174 of 1×10^{-5} per year for ΔCDF and 1×10^{-6} per year for ΔLERF. Accordingly, the NRC staff concludes that the RG 1.205 guidance is satisfied and that the total ΔCDF and ΔLERF meet the acceptance guidelines in RG 1.174.

The NRC identified that the assumption made in the FPRA that untraced cables are failed in all fire sequences is a conservative approach for modeling untraced cables in the post-transition plant model, but this approach can lead to underestimation of the change-in-risk when used in the compliant plant model. The NRC staff requested that the licensee: (1) explain how the untraced cables were treated in the compliant and post-transition plant models, (2) justify that the assumptions made about untraced cables do not contribute to underestimation of the transition change-in-risk, and (3) justify that any other major model conservatisms do not contribute to underestimation of the transition change-in-risk. In its response PRA RAI 16 (Reference 9), the licensee explained that the extent of untraced cables is about 15 percent of the FPRA components with cables, and that components with untraced cables were treated in the FPRA by globally failing them in the compliant and post-transition plant FPRA models. The licensee indicated that the impact on total fire risk from this treatment was "largely" from always assuming failure of the feedwater system in a fire scenario and effectively not crediting the system in the FPRA. The licensee stated that no VFDRs are associated with the feedwater system, and, therefore, concluded that the untraced cables do not contribute to the underestimation of the change-in-risk. The NRC staff requested that the licensee provide a discussion of whether any of the uncredited systems are associated with a VFDR and could contribute to underestimation of the transition change-in-risk, and if so, justification for this treatment. In its response to PRA RAI 16.01 (Reference 11), the licensee explained that there are no systems assumed to be failed after a fire in the FPRA (i.e., not credited) that are associated with a VFDR. The NRC staff concludes that the conservatism associated with modeling systems with untraced cables as failed in the compliant FPRA model is acceptable because it does not lead to underestimation of the transition change-in-risk.

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

3.4.7 Uncertainty and Sensitivity Analyses

The NRC staff identified that a key source of uncertainty in the Hatch LAR to adopt 10 CFR 50.69, "Risk-informed Categorization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors, (Reference 87), for the IEPRA model, was the assumed

conditional probability of $1x10^{-2}$ used to account for the loss of NPSH on the low-pressure emergency core cooling system (ECCS) pumps following emergency containment venting.

The NRC staff requested that the licensee provide a description of the basis for the assumed conditional probability of $1x10^{-2}$, indicating the degree of uncertainty, and justify that the assumption does not impact the fire risk estimates. In its response to PRA RAI 18 (Reference 12), the licensee explained that it revised the FPRA model to remove the uncertainty associated with the assumed conditional failure probability, therefore, rather than justifying that the uncertainty had no impact on the application, the licensee explained that it used the thermal hydraulic and operator response analysis to model the dependency of low-pressure ECCS pump NPSH on torus conditions and the operator's ability to throttle flow following containment venting. Based on the information provided by the licensee, the NRC staff concludes that the licensee's treatment of the uncertainty associated with low-pressure ECCS pump NPSH is acceptable because the licensee removed the uncertainty using more refined modeling.

3.4.8 Conclusion for Section 3.4

Based on the information provided by the licensee, regarding the fire risk assessment methods, tools, and assumptions used to support transition to NFPA 805, the NRC staff concludes that:

- The licensee's PRA used to perform the risk assessments in accordance with NFPA 805, Section 2.4.4, "Plant Change Evaluations," and Section 4.2.4.2 "Use of Fire Risk Evaluations," is of sufficient quality to support the application to transition to NFPA 805. The NRC staff concludes that the PRA approach, methods, tools and data are acceptable and are in accordance with NFPA 805, Section 2.4.3.3.

- The licensee stated that it has completed changes to the PRA model which replaced unacceptable approaches, data, and methods identified during the LAR review with acceptable approaches, data, and methods, as described. Therefore, the NRC staff concludes that the FPRA model may be used to support post-transition self-approval of FPP changes subject to completion of all implementation items identified in LAR, Table S-3 as supplemented, because the identified acceptable methods will be used unless replaced by other acceptable methods.

- Final LAR Table S-3, Implementation Item IMP-19 includes an action to update the FPRA to reflect the as-built, as-operated plant and if the updated FPRA does not meet the risk acceptance guidelines found in RG 1.174, Rev. 2, measures will be taken to reduce the FPRA risk.

- The transition process included a detailed review of fire protection DID and safety margins as required by NFPA 805. The NRC staff concludes the licensee's evaluation of DID and safety margins to be acceptable. The licensee's process followed the NRC-endorsed guidance in NEI 04-02, Revision 2, and is consistent with the approved NRC staff guidance in RG 1.205, Revision 1, which provides an acceptable approach for meeting the requirements of 10 CFR 50.48(c).

- The total additional risk of RAs for Unit 1 is an increase in CDF of $4.19x10^{-7}$ per year and an increase in LERF of $1.18x10^{-8}$ per year. For Unit 2 the total

additional risk of RAs is an increase in CDF of 5.13×10^{-7} per year and an increase in LERF of 2.84×10^{-8} per year. Accordingly, the total additional risk of RAs is below the risk acceptance guidelines in RG 1.174, Revision 2, of 1×10^{-5} per year for CDF and 1×10^{-6} per year for LERF.

- The risk increase associated with the transition to NFPA 805 satisfies the guidance in RG 1.205, Revision 1, RG 1.174, Section 2.4, and NUREG-0800, Section 19.2 regarding acceptable risk.

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

3.5 Nuclear Safety Capability Assessment Results

In Section 3.2.1 of the SE, 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 Section 3.5.1 of the SE, 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 Section 3.5.3 of the SE, the NRC staff provides its review of the NSCA results for NPO at Hatch.

3.5.1 Nuclear Safety Capability Assessment Results by Fire Area

Section 2.2.4 of NFPA 805, "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.

Section 2.4.2.4 of NFPA 805, "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. Chapter 4 of NFPA 805, provides the methodology for both the deterministic and PB approach to meet the NSPC. Section 4.2.2 of NFPA 805, states that the PB approach shall be permitted to use deterministic methods for simplifying assumptions within the fire area.

Hatch is a dual unit BWR with approximately 128 individual fire areas, and each fire area is composed of one or more fire zones. The licensee performed the NSCA on a fire area basis. The licensee stated that fire areas were established to support the development of the FPRA. Attachment C of the LAR, as supplemented, provides the results of these analyses on a fire area basis and identified the fire zones within the fire areas.

Attachment C of the LAR, Table C-2, as supplemented, provides a summary of the Hatch NFPA 805 compliance basis and required fire protection systems and features for each fire compartment. Attachment C of the LAR, as supplemented, also provides the results of the fire area transition review for at-power modes of operation and summarizes Hatch compliance with Chapter 4 of NFPA 805, for each fire area. Attachment C of the LAR, as supplemented indicates that both the PB approach and deterministic approaches allowed by Section 4.2.4 of

NFPA 805, were used. For each fire area, LAR Attachment C, as supplemented, included the following information:

- The regulatory bases for the new FPP, including fire detection and suppression systems required to meet 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

Chapter 4 of NFPA 805, 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 area. For each fire area, Attachment C of the LAR, Table C-2, as supplemented, identifies if the fire suppression and detection systems installed in the area 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.

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

Based on statements provided in Attachment C of the LAR, as supplemented, the NRC staff concludes that the licensee's treatment of this issue is acceptable because the licensee adequately identified the fire detection and suppression systems required to meet the NFPA 805 NSPC on a fire area basis.

3.5.1.2 Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

Sections 2.4.2.4 and 4.2.1 of NFPA 805, require that the effects of fire suppression activities on the ability to achieve the NSPC be evaluated for each fire area. Each fire area included in Attachment C of the LAR, as supplemented, Table C-1, includes 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 areas, 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 areas, the licensee concluded that fire suppression activities will not impact the ability to achieve the NSPC in accordance with NFPA 805, Section 4.2.1.

As described in the LAR, certain plant-specific features such as spray nozzle configurations, spray barriers, containment dikes, and floor drainage systems would prevent water accumulation and intrusion into credited NSCA equipment due to sprinkler spray. In addition, fire brigade members are trained to protect safety-related equipment from fire and water damage by giving consideration to the type of fire suppression method used (fire extinguisher, water, direct/indirect attack, etc.) based on the type of fire, extent, intensity and the need to limit the quantity of water in areas containing energized electrical and safety-related equipment, and by discharging water in a judicious manner in such a way as to limit the amount of overspray beyond the immediate area of the fire. Therefore, fire suppression activities are not expected to adversely affect achieving the NSPC.

For fire area 1017, the licensee stated that switchgear 1R23S004 is potentially susceptible to damage due to water spray from the sprinkler in fire zone 0014K and that LAR Table S-2, as supplemented, Item 1 will modify the door fusible link configuration between the suppression system in fire zone 0014K and the switchgear in Fire Area 1017 to address the issue.

Based on 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 or will not adversely affect achievement of the NSPC with the completion of a modification which is required to be completed in accordance with the license condition.

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). Each of these exemptions is further detailed in LAR Attachment K. In LAR Section 4.2.3, the licensee stated that no licensing actions will be transitioned into the NFPA 805 FPP as previously approved.

Based on the NRC staff's review of the licensing actions identified and described in Attachment K to the LAR, the NRC staff concludes that the licensing actions are identified by applicable fire area and the existing exemptions are no longer required to support the proposed license amendments because the licensee has performed EEEEs in accordance with NEI 04-02 (Reference 7) as endorsed by RG 1.205, (Reference 4), and/or transitioned the applicable fire areas as performance-based in accordance with NFPA 805 Section 4.2.4.2.

3.5.1.4 Existing Engineering Equivalency Evaluations

Section 2.2.7 of NFPA 805, "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.

Section 4.2.2 of the LAR, "Existing Engineering Equivalency Evaluation Transition", 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 in the LAR, the EEEEs that demonstrate that a fire protection system or feature is adequate for the hazard. The licensee identified and summarized the EEEEs for each fire area, as applicable, in LAR Attachment C, as supplemented. 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, as supplemented, 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

The licensee stated that for those fire areas where deterministic criteria were not met, VFDRs were identified and evaluated using PB methods in Section 4.5.2.2 of the LAR, "Fire Risk Approach." VFDR identification, characterization, and resolutions were identified and summarized in Attachment C of the LAR, as supplemented, for each fire area. Documented variances were all represented as separation issues. The following strategies were used by the licensee in resolving the VFDRs:

- An FRE determined that applicable risk, DID, and safety margin criteria were satisfied without further action; or
- An FRE determined that applicable risk, DID, and safety margin criteria were satisfied with a credited RA; or
- An FRE determined that applicable risk, DID, and safety margin criteria were satisfied with a RA-DID; or
- An FRE determined that applicable risk, DID, and safety margin criteria were satisfied with a plant modification(s), as identified in the LAR Table S-2, as supplemented.

For all fire areas where the licensee used the PB approach to meet the NSPC, each VFDR and the associated disposition have been described in Attachment C of the LAR, as supplemented. Based on review of the VFDRs and associated resolutions as described in Attachment C of the LAR, as supplemented, the NRC staff concludes that the licensee's identification and resolution of the VFDRs is acceptable because the licensee identified, characterized, and resolved all VFDRs as summarized in Attachment C of the LAR for each fire area in accordance with NEI 04-02 for compliance with NFPA 805, Section 4.2.4.

3.5.1.6 Recovery Actions

Attachment G of the LAR, as supplemented, lists the RAs identified in the resolution of VFDRs in LAR Attachment C for each fire area. The RAs identified include both actions considered necessary to meet risk acceptance criteria, as well as actions relied upon as DID. The NRC

staff reviewed Section 4.2.1.3 of the LAR, "Establishing Recovery Actions," and Attachment G of the LAR, as supplemented, to evaluate whether the licensee meets the associated requirements for the use of RAs per NFPA 805. The details of the NRC staff's review for RAs are described in Section 3.2.5 of the SE, "Establishing Recovery Actions." The NRC staff's evaluation of the additional risk of RAs credited to meet the risk acceptance guidelines is provided in Section 3.4.5 of the SE.

In the LAR, the licensee stated that RAs are identified in Attachment G, as supplemented. The licensee stated that it evaluated the set of RAs necessary to demonstrate the availability of a success path for the NSPC for additional risk using the process described in NEI 04-02, FAQ 07-0030, and RG 1.205.

The NRC staff reviewed Section 4.2.1.3 of the LAR, "Establishing Recovery Actions," and Attachment G, as supplemented, to evaluate whether the licensee meets the associated requirements for the use of RAs per NFPA 805. The NRC staff's evaluation of the licensee's process for identifying RAs and assessing their feasibility is provided in Section 3.2.5 of the SE, "Establishing Recovery Actions," and concluded that the licensee followed the endorsed guidance of NEI 04-02 and RG 1.205 to identify and evaluate RAs in accordance with NFPA 805, and that the feasibility criteria applied to RAs are acceptable because the criteria conforms with the endorsed guidance contained in NEI 04-02 and because the licensee will be in compliance with the regulation upon completion of implementation items described in LAR Table S-3, as supplemented, that would be required to be completed by the proposed license condition.

3.5.1.7 Plant Fire Barriers and Separations

Except for the 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 area boundaries as part of the RI/PB FPP.

Fire area boundaries are established for those areas described in Attachment C of the LAR, 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 Attachment A of the LAR (Section 3.1 of SE).

3.5.1.8 Electrical Raceway Fire Barrier Systems

The licensee stated that the ERFBS installed at Hatch complies with the requirements of NFPA 805, Section 3.11.5, with no additional clarification. Each fire area using ERFBS is identified in LAR Attachment C. In fire areas with deterministic compliance, the ERFBS met the requirements of NFPA 805, Section 4.2.3. In fire areas with PB compliance, the ERFBS were analyzed using the PB approach in accordance with NFPA 805, Section 4.2.4. There were no VFDRs associated with ERFBS.

Based on the information provided by the licensee, the NRC staff concludes 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 Attachment C of the LAR, as supplemented, for those fire areas that used a deterministic approach in accordance with NFPA 805, Section 4.2.3, the NRC staff concludes that each of the fire areas analyzed using the deterministic approach meets the associated criteria of NFPA 805, Section 4.2.3. This conclusion is based on:

- The licensee's documented compliance with NFPA 805, Section 4.2.3,
- The licensee's statement that the success path will be free of fire damage without reliance on RAs,
- The licensee's assessment that the fire suppression activities in the fire area will have no impact on the ability to meet the NSPC, and
- The licensee's determination of the automatic fire suppression and detection systems and other fire protection feature (i.e., ERFBS) required to meet the NSPC.

For those fire areas that used the PB approach in accordance with NFPA 805, Section 4.2.4, the NRC staff concludes that each fire area has been analyzed properly, and that compliance with the NFPA 805 requirements was demonstrated as follows:

- Deviations from the pre-NFPA 805 fire protection licensing basis that were transitioned to the NFPA 805 licensing basis were reviewed for applicability, as well as continued validity, and found acceptable.
- VFDRs were evaluated and either found to be acceptable based on an integrated assessment of risk, DID, and safety margins; or modifications or RAs were identified with actions planned or implemented to address the issue.
- RAs used to demonstrate the availability of a success path to achieve the NSPC were evaluated and the additional risk of their use determined, reported, and found to be acceptable. The licensee's analysis appropriately identified the fire protection SSC required to meet the NSPC, including fire suppression and detection systems and other fire protection feature (i.e., ERFBS).
- Fire area boundaries (ceilings, walls, and floors), such as fire barriers, fire barrier penetrations, and through penetration fire stops, are established.
- Credited ERFBS were documented on a fire area basis, verified to be installed consistent with tested configurations and rated accordingly, and evaluated using an EEEE.

The NRC staff concludes that each fire area utilizing the deterministic or PB approach meets the applicable requirements of NFPA 805, Section 4.2.

3.5.2 Clarification of Prior NRC Approvals

In LAR Attachment T, the licensee stated that there are no elements of the pre-transition FPP licensing basis that require clarification of prior NRC approval.

3.5.3 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 Section 4.3 of the LAR, "Non Power Operational Modes," and Attachment D of the LAR, to evaluate the licensee's treatment of potential fire impacts during NPO modes. In Section 4.3 of the LAR, the licensee stated that it used the process outlined in NEI 04-02 and FAQ 07-0040 and demonstrated that the NSPC are met during NPO modes.

3.5.3.1 NPO Strategy and Plant Operating States

In Section 4.3 of the LAR and Attachment D, the licensee stated that the process used to demonstrate that the NSPC are met during NPO modes is consistent with the guidance contained in NEI 04-02 and FAQ 07-0040. As described in Attachment D of the LAR, the licensee's procedures utilize the DID concept to minimize shutdown risk and maximize the availability of critical components and station systems that ensure nuclear safety during shutdown conditions. The licensee stated that HREs are outage activities or plant configurations during shutdown, which would make the plant more susceptible to an event causing the loss of NPO KSFs.

3.5.3.2 NPO Analysis Process

In Section 4.3.1 of the LAR, the licensee stated that its goal is to ensure that contingency plans are established when the plant is in an NPO mode where the risk is intrinsically high. Section 4.3 of the LAR discusses these additional controls and measures, and that during low-risk periods, normal risk management controls, as well as fire prevention/protection processes and procedures, will be used.

In Section 4.3.2 of the LAR, the licensee stated that based on FAQ 07-0040 "Clarification on Non-Power Operations," the plant operating states (POSs) were considered for identifying equipment and cable selection to support the KSFs of DHR capability, electrical power availability, inventory control, reactivity control, spent fuel pool cooling, containment integrity, and associated support functions (e.g., PSW, heating, ventilation and air conditioning (HVAC), etc.).

3.5.3.3 NPO KSFs and SSC Used to Achieve Performance

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

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

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

- Identified the SSC required to meet the KSFs during the POS analyzed.

- Identified the location of these SSC and their associated cables.

- Performed analyses on a fire area basis to identify pinch points where one or more KSF could be lost as a direct result of fire-induced damage.

- Planned/implemented modifications to appropriate procedures in order to employ a fire protection strategy for reducing risk at these pinch points during HREs.

In Attachment D of the LAR, the licensee stated that POS 1, POS 2, and POS 3 were evaluated as part of the NPO review process in accordance with FAQ 07-0040. The licensee also defined the KSFs and stated that the following KSFs were evaluated against the POSs for inclusion in the NPO transition review:

- DHR
- Electrical Power Availability
- Inventory Control
- Reactivity Control
- Spent Fuel Pool Cooling
- Containment Integrity
- Support System (e.g., PSW, HVAC, etc.)

The licensee stated that the evaluation determined that Reactivity Control and Containment Closure KSF could be excluded from consideration. The licensee further stated that the evaluation of the Spent Fuel Pool Cooling system determined that detailed modeling of the system for specific fire induced component damage was not necessary due to adequate DID. The licensee stated that various modes of operation for each system used to satisfy each KSF were reviewed, and that it developed a comprehensive list of equipment.

The licensee stated that the review identified the components necessary to accomplish the KSFs using a methodology consistent with that identified for the Hatch At-Power Analysis, including identification of components that could spuriously operate and impair the KSF path. The components comprising each KSF path were compared to the population of components contained in the Hatch At-Power Analysis. The majority of equipment required to maintain the NPO KSFs is the same as that required to safely shutdown the plant while at power. Some SSD components have a different functional requirement during NPO modes. In those cases where a component's NPO function was not appropriately addressed in the SSD model, the licensee performed additional circuit analysis and routing.

Based on its review of the information provided by the licensee, as supplemented, the NRC staff concludes that the licensee used methods consistent with the guidance provided in NEI 04-02, FAQ 07-0040 and RG 1.205 to identify the equipment required to achieve and maintain the fuel in a safe and stable condition during NPO modes. Furthermore, the NRC staff concludes that the licensee has a process in place to ensure that fire protection DID measures will be implemented to achieve the KSFs during plant outages and that any required actions will be completed through implementation items identified in LAR Table S-3, as supplemented, which would be required to be completed by the proposed license condition.

3.5.3.4 NPO Pinch Point Resolutions and Program Implementation

The licensee stated that it performed a deterministic fire separation analysis to identify pinch points (i.e., areas where redundant equipment and cables credited for a given KSF fail due to fire damage). The licensee further stated that 128 fire areas at each unit were analyzed, and that 87 fire areas at Hatch, Unit No. 1, and 88 fire areas at Hatch, Unit No. 2, were found to have at least one success path following the postulated fire. The licensee further stated that overall, 41 areas at Hatch, Unit No. 1, and 40 areas at Hatch, Unit No. 2, were found to have pinch points resulting in the potential loss of all success paths for one or more KSFs.

In Attachment D of the LAR, the licensee stated that to minimize fire risk, fire areas with identified pinch points were evaluated and plant controls were considered that are consistent with FAQ 07-0040. Table S-3 of the LAR, as supplemented, Implementation Item IMP-12, addresses the NPO implementation plan, and the NRC staff concludes that this action is acceptable because it will incorporate the provisions of NFPA 805 in the FPP and would be required by the proposed license condition. The licensee stated that the strategies will include but not be limited to the following based on the actions from FAQ 07-0040:

- Restriction of hot work in fire areas during periods of increased vulnerability;
- Verification of operable detection and/or suppression in the vulnerable areas;
- Limitation of transient combustible materials in fire areas during periods of increased vulnerability;
- Use of plant configuration changes/realignment (e.g., removing power from equipment once it is placed in its desired position);
- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (e.g., surveillance cameras) during periods of increased vulnerability;
- Use of plant procedures to mitigate potential losses of KSFs;
- Identification and monitoring *in-situ* ignition sources for “fire precursors” (e.g., equipment temperatures); and
- Reschedule the work to a period with lower risk or higher DID.

NFPA 805 requires that the NSPC be met during any operational mode or condition, including NPO. Based on the information provided by the licensee, the NRC staff concludes that the licensee will provide reasonable assurance that the NSPC are met during NPO modes and HREs because evaluations of power-operated components needed to support NPO KSF have been performed and revisions to procedures in order to employ a fire protection strategy for reducing risk at these pinch points during HREs are included in LAR Table S-3, as supplemented, and are required to be completed by the proposed license condition.

3.5.4 Conclusion for Section 3.5

The NRC staff reviewed the licensee's RI/PB FPP as described in the LAR, and its supplements, to evaluate the NSCA results. The licensee used the PB approach in accordance with NFPA 805, 4.2.4.

For those fire areas that utilized a PB approach, the NRC staff verified the following:

- The engineering evaluations from the existing FPP were evaluated and found to be valid and acceptable for meeting the requirements of NFPA 805, as allowed by NFPA 805, Section 2.2.7.

- Exemptions not carried over from existing FPP were evaluated and found no longer needed for transitioning to proposed RI/PB FPP.

- Fire suppression effects were evaluated and found to have no adverse impact on the ability to achieve and maintain the NSPC for each fire area.

- All VFDRs were evaluated using the FRE PB approach (in accordance with NFPA 805, Section 4.2.4.2), to address risk impact, DID, and safety margin, and found to be acceptable.

- All RAs necessary to demonstrate the availability of a success path were evaluated with respect to the additional risk presented by their use and found to be acceptable in accordance with NFPA 805, Section 4.2.4.

- All RA-DIDs were properly documented for each fire area.

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

Accordingly, the NRC staff concludes that there is reasonable assurance that each fire area utilizing the PB approach does so in accordance with NFPA 805, Section 4.2.4.

The NRC staff's review of the licensee's analysis and outage management process during NPO modes concluded that the licensee provided reasonable assurance that the NSPC will be met during NPO modes and HREs, and that the licensee used methods consistent with the guidance provided in RG 1.205, NEI 04-02, and FAQ 07-0040. The NRC staff's review also concluded that for low risk NPO evolutions the normal FPP DID actions are credited for addressing the risk impact of those fires that potentially affect one or more trains of equipment that provide a KSF required during NPO modes but would not be expected to cause the total loss of that KSF. For NPO evolutions identified as HREs, additional fire protection DID measures will be taken to manage risk in fire areas containing known pinch points or where pinch point may arise because of equipment taken out of service. The NRC staff concludes that this overall approach for fire protection during NPO modes is acceptable.

3.6 Radioactive Release Performance Criteria

As discussed in Section 2.0 of the SE, 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 an NPP. The NRC staff has endorsed, with exceptions, NEI 04-02 as it provides methods acceptable to the NRC staff for establishing an RI/PB FPP consistent with NFPA 805. Additional guidance is provided in FAQ 09-0056. 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.

NFPA 805, Chapter 1 defines the radioactive release goals, objectives, and performance criteria that must be met by the FPP in the event of a fire at NPPs in any plant operational mode.

The licensee is implementing the methodology described in NEI 04-02, Revision 2 to transition Hatch from the current fire protection licensing basis to NFPA 805. As described in Section 4.4.1 of the LAR, the licensee applied the clarifying guidance of FAQ 09-0056 in evaluating the radiological release performance criteria. More specifically, the licensee reviewed fire pre-plans, fire brigade training materials and engineering controls being credited during all plant operating modes; including full power and NPO. The NRC staff reviewed the licensee's assessment in order to determine if the existing FPP with its planned modifications would meet the radioactive release performance criteria, in accordance with 10 CFR 50.48(a) and (c) using the guidance in RG 1.205 and NUREG-0800, Section 9.5.1.2.

The licensee's radioactive release review states that the FPP, as defined in the fire strategies, and fire brigade training materials will be compliant with the requirements of NFPA 805 and the guidance in NEI 04-02 and RG 1.205 upon completion of the implementation items identified in Table S-3 of the LAR, as supplemented. The licensee addressed the potential for radioactive releases due to fire suppression activities by evaluating and revising fire strategies and training materials.

The licensee performed a review of compartments (fire areas/zones) that included a screening process to determine what compartments have a potential for a radioactive release. Areas outside of the radiologically controlled area (RCA) were viewed as having no risk and were screened out from further review. Compartments were identified based on whether common smoke and runoff control systems were present. Information for the compartments that were screened in is provided in Attachment E of the LAR. The fire strategies were evaluated for the screened-in compartments to ensure that the locations that have potential for radioactive releases are subject to specific steps for the containment and monitoring of potentially contaminated smoke and fire suppression runoff water.

The screened-in areas included plant areas in the RCA such as the Hatch Units 1 and 2 Reactor Buildings, Refueling Floors, and Waste Gas Treatment Building. For these plant areas, the licensee's review identified that the engineering controls, pre-fire plans and training materials were sufficient to contain and monitor gaseous and liquid effluent. Engineering controls credited for containment of gaseous (e.g., filtered exhaust systems) and liquid (e.g., floor drains that transfer to the floor drain tank for processing) effluents are documented in Attachment E of the LAR. No credit was taken for operator action to assure radiological releases were limited. The NRC staff's review determined that the existing engineering controls for these areas were adequate because the gaseous effluent is filtered and monitored prior to discharge, and the liquid effluent is collected, processed, and monitored prior to discharge.

The licensee's review also identified the "Bounded Areas" compartment. This compartment includes areas such as the Sealand Storage Facility, Laundry Storage Building, Units 1 and 2 CSTs/Pumps, and Low Level Radwaste Facility, where there are not sufficient engineering controls for containment of gaseous and liquid effluents. For these areas the licensee performed a quantitative evaluation (i.e., dose calculation) to address the alternative to engineering controls for the Bounded Areas. The Hatch quantitative evaluation demonstrates that in the Bounded Areas any potential releases of contaminated gaseous or liquid effluents resulting from a fire involving radioactive contents are below 10 CFR Part 20 limits and satisfy the acceptance criteria of FAQ 09-0056.

In its response to Health Physics RAI 01 (Reference 9), the licensee described the quantitative evaluation to demonstrate that the 10 CFR Part 20 limits will not be exceeded, for areas where containment and engineering controls could not be relied upon. The licensee's quantitative evaluation considered a release of radioactivity contained within a maximally loaded low-specific activity (LSA) container due to a fire and/or fire suppression activity in an area with insufficient engineering controls to control or prevent airborne or liquid effluents to the unrestricted area boundary. The source term was calculated using the RADMAN radwaste computer program assuming a 10 mrem/hour dose at 2 meters from the surface of the container, the highest expected waste density, and the Hatch site radwaste source term.

The NRC staff reviewed the licensee's methodology, critical assumptions, input parameters and resulting doses for the quantitative evaluation. The licensee based its evaluation on conservative assumptions and the methodology included using the NRC Radiological Assessment System for Consequence Analysis (RASCAL) computer software to calculate the airborne pathway dose consequences and a spreadsheet to calculate the liquid pathway doses. Based on the review of the licensee's quantitative evaluation the NRC staff concludes that the licensee has adequately demonstrated that any radioactive material released during fire suppression activities would not exceed the public dose limits of 10 CFR Part 20.

The licensee reviewed the pre-fire plans and fire brigade training materials to ensure they were consistent with the fire strategies in terms of containment, removal and monitoring of potentially contaminated smoke and fire suppression water. Those training materials needing to be revised were identified and documented. The revised training materials will identify potentially contaminated areas, provide instruction for communication with radiation protection personnel, and describe precautions that should be undertaken for the containment and safe removal of contaminated smoke and water runoff in the potentially contaminated areas. The training materials will also describe the presence and use of monitored ventilation and drainage systems. The licensee's planned procedure changes, process updates, and training for affected plant personnel, that will be completed prior to the implementation of the FPP are documented in Table S-3 of the LAR, as supplemented, and would be required to be completed in accordance with the proposed license condition.

Based on the information provided by the licensee, as supplemented; the licensee's use of revised fire strategies; the results of the NRC staff's evaluation of the identified engineering controls used to manage suppression water and combustion products; and the development and implementation of newly revised fire brigade training and procedures, the NRC staff concludes that the licensee's RI/PB FPP provides reasonable assurance that radiation releases to any unrestricted area resulting from the direct effects of fire suppression activities at Hatch are ALARA and are not expected to exceed the radiological dose limits in 10 CFR Part 20. The NRC staff concludes that the licensee's proposed RI/PB FPP complies with the requirements specified in NFPA 805, Sections 1.3.2, 1.4.2, and 1.5.2, and that this approach is acceptable.

3.7 NFPA 805 Monitoring Program

Section 4.6 of the LAR, “Monitoring Program”, describes the program the licensee will implement to monitor the availability, reliability, and performance of FPP systems and features at Hatch 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.

The NRC staff reviewed Section 4.6 of the LAR, “Monitoring Program,” that the licensee developed to monitor availability, reliability, and performance of FPP systems and features after the transition to NFPA 805. The NRC staff focused on the critical elements related to the monitoring program, including the selection of FPP systems and features to be included in the program, the attributes of those systems and features that will be monitored, and the methods for monitoring those attributes. Implementation of the monitoring program will occur on the same schedule as the NFPA 805 RI/PB FPP implementation, which the NRC staff concludes is acceptable (see Section 2.7 of the SE).

The licensee stated that it will develop an NFPA 805 monitoring program consistent with the NRC-approved version of FAQ 10-0059 “Close-out of National Fire Protection Association Standard 805 Frequently Asked Question 10-0059 On National Fire Protection Association Standard 805, Monitoring,” (Reference 53). The licensee further stated that the monitoring program will incorporate phases for scoping, screening using risk criteria, risk target value determination, and monitoring implementation, and that the scope of the program will include fire protection systems and features, NSCA equipment, SSC relied upon to meet radioactive release criteria, and FPP elements.

Based on the above, the NRC staff concludes that the licensee’s NFPA 805 monitoring program and development and implementation process is acceptable and assures that the licensee will implement an effective program for monitoring risk-significant fires because it:

- Establishes the appropriate SSC to be monitored;
- Uses an acceptable screening process for determining the SSC to be included in the monitoring program;
- Establishes availability, reliability, and performance criteria for the SSC being monitored; and
- Requires corrective actions when SSC availability, reliability, and performance criteria targets are exceeded in order to bring performance back within the required range.

However, since the final values for availability and reliability, as well as the performance criteria for the SSC being monitored, have not been established for the monitoring program as of the date of this SE, completion of the licensee's NFPA 805 monitoring program is an implementation item addressed in Table S-3 of the LAR, as supplemented, Implementation Item IMP-14.

The NRC staff concludes that completion of the monitoring program on the same schedule as the implementation of NFPA 805 is acceptable because the monitoring program will be completed with the other implementation items as described in Table S-3 of the LAR, as supplemented, 365 days after NRC approval, which is prior to completion of the modifications to achieve full compliance with 10 CFR 50.48(c) (which is by the startup of the second refueling outage (for each unit) after issuance of the SER).

The NRC staff reviewed the licensee's RI/PB FPP and concludes that there is reasonable assurance that the licensee's monitoring program will meet the requirements specified in Sections 2.6, 2.6.1, 2.6.2, and 2.6.3 of NFPA 805 because the licensee identified an action to implement the fire protection monitoring program in accordance with the NRC-approved version of FAQ 10-0059 and included that action as implementation item IMP-14 in Table S-3 of the LAR, as supplemented, that would be required by the proposed license condition.

3.8 Post-Implementation Plant Change Evaluation Process

The NRC staff reviewed Section 4.7.2 of the LAR, "Compliance with Configuration Control Requirements in Section 2.7.2 and 2.2.9 of NFPA 805", for compliance with the NFPA 805 PCE process requirements to address potential changes to the 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.

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

Section 4.7.2 of the LAR states that the PCE 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.

In the LAR, the licensee stated that the PCE process begins by defining the change or altered condition to be examined and the baseline configuration. The baseline is defined by the design basis and licensing basis. The licensee also stated that the baseline is defined as that plant condition or configuration that is consistent with the design and licensing basis, and that the

changed or altered condition or configuration that is not consistent with the design and licensing basis is defined as the proposed alternative.

The licensee stated that once the definition of the change is established, a screening is then performed to identify and resolve minor changes to the FPP, and the screening is consistent with fire protection regulatory review processes currently in place at nuclear plants under traditional licensing bases. The licensee further stated that 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 88), and that the process will address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.).

The licensee stated that the characteristics of an acceptable screening process that meets the "assessment of the acceptability of risk" requirement of Section 2.4.4 of NFPA 805 are: (1) the quality of the screen is sufficient to ensure that potentially greater than minimal risk increases receive detailed risk assessments appropriate to the level of risk; (2) the screening process must be documented and be available for inspection by the NRC; and (3) the screening process does not pose undue evaluation or maintenance burden. The licensee further stated that if any of the above is not met, then it would proceed to the risk evaluation step.

The licensee stated that the screening is followed by engineering evaluations that may include FM and risk assessment techniques, and the results of these evaluations are then compared to the acceptance criteria. The licensee stated that changes that satisfy the acceptance criteria of NFPA 805, Section 2.4.4 and the license condition (see Attachment M of the LAR) can be implemented within the framework provided by NFPA 805, and that the changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The licensee also stated that the acceptance criteria require that the resultant change in core damage frequency (CDF) and large early release frequency (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 licensee stated that the risk evaluation involves the application of FM analyses and risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change and that in certain circumstances, an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions, provided the use of such assumptions does not unnecessarily challenge the acceptance criteria.

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

The licensee stated its FPP configuration is defined by the program documentation and, to the greatest extent possible, the existing configuration control processes for modifications, calculations and analyses, and FPP license basis reviews will be utilized to maintain configuration control of the FPP documents. The licensee further stated the configuration control procedures, which govern the various Hatch documents and databases that currently exist, will be revised to reflect the new NFPA 805 licensing bases requirements. This action is included in Table S-3 of the LAR, as supplemented, Implementation Item IMP-16. The NRC

staff concludes that this action is acceptable because it will incorporate the provisions of NFPA 805 in the FPP and would be required by the proposed license conditions.

The licensee stated that several NFPA 805 document types, such as NSCA supporting information and non-power mode NSCA treatment, etc., generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The licensee further stated the new procedures will be modeled after the existing processes for similar types of documents and databases, and system level design basis documents will be revised to reflect the NFPA 805 role that the system components now play. This action is included in Implementation Item IMP-15, which is included in Table S-3 of the LAR, as supplemented. The NRC staff concludes that this action is acceptable because it will incorporate the provisions of NFPA 805 in the FPP and would be required by the proposed license conditions.

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

- Deterministic Approach: Comply with NFPA 805, Chapter 3, and Section 4.2.3 requirements; or

- PB 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 will be used to determine if the proposed change could be implemented "as is" or whether prior NRC approval of the proposed change is required.

The licensee stated that this process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174, which require the use of qualified individuals and procedures that require calculations be subject to independent review and verification, record retention, peer review, and a CAP that ensure appropriate actions are taken when errors are discovered.

Since NFPA 805 always requires the use of a PCE regardless of what element requires the change, the NRC staff concludes that in accordance with the requirements of NFPA 805, if FPP impacts are determined to exist as a result of the proposed change, the issue would be resolved by utilizing the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the Hatch NFPA 805 fire protection license condition to assess the acceptability of the proposed change. This process will be used to determine if prior NRC approval of the proposed change is required.

The NRC staff concludes that the licensee's PCE 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 PCE 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 Section 3.4 of the SE). Therefore, the NRC staff concludes that the licensee's PCE process is acceptable. Actions to ensure implementation of the PCE process are completed are included as a requirement in the new FPP license condition.

3.9 Program Documentation, Configuration Control, and Quality Assurance

Section 2.7 of NFPA 805, "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 Section 4.7 of the LAR, the licensee described how it will meet these requirements.

3.9.1 Documentation

Section 2.7.1 of NFPA 805, "Content," specifies the documentation requirements for an FPP based on NFPA 805. Section 4.7.1 of the LAR, "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, Hatch has documented analyses to support compliance with 10 CFR 50.48(c). The analyses are being performed in accordance with SNC'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 Section 4.7.1 of the LAR, the NRC staff has reasonable assurance that the licensee will comply with the documentation requirements of NFPA 805, Section 2.7.1.

3.9.2 Configuration Control

Section 2.7.2 of NFPA 805, "Configuration Control," specifies the requirements to maintain the FPP documentation up-to-date. Section 4.7.2 of the LAR states, in part, that:

> Program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) is subject to SNC 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.

Section 4.7.2 of the LAR further states:

> The Hatch 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 Hatch documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements.

Table S-3 of the LAR, as supplemented, includes Implementation Item IMP-15 to develop the new NFPA 805 configuration control procedures and processes.

Section 4.7.3 of the LAR 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 Hatch maintains an as-built, as operated PRA model of the plant.

Sections 3.4.3.4 and 3.9.3 of the SE describe the NRC staff's review of the licensee's process for updating and maintaining the Hatch FPRA to reflect plant changes made after completion of the transition to NFPA 805.

Based on the statements in Sections 4.7.2 and 4.7.3 of the LAR, the NRC staff has reasonable assurance that the licensee will comply with the configuration control requirements of NFPA 805, Section 2.7.2. Actions related to implementation of the configuration control process are a requirement in the new FPP license condition.

3.9.3 Quality

Section 2.7.3 of NFPA 805, "Quality," specifies the QA 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 Section 4.7.3 of the LAR, the licensee stated that it will maintain the existing fire protection QA program. 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. 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

Section 2.7.3.1 of NFPA 805, requires that each analysis, calculation, or evaluation performed be independently reviewed. In Section 4.7.3 of the LAR, the licensee stated that it requires that the calculations and evaluations in support of the NFPA 805 LAR, exclusive of the FPRA, be performed within the scope of the QA program, which requires independent review as defined

by its procedures. Based on the above, the NRC staff has reasonably assurance that the licensee will comply with NFPA 805, Section 2.7.3.1.

3.9.3.2 Verification and Validation

Section 2.7.3.2 of NFPA 805, requires each calculational model or numerical method to be verified and validated through comparison to test results or other acceptable models. In Section 4.7.3 of the LAR, 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.

Volumes 1-7 of NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," (Reference 32), and Supplement 1 (Reference 33), documents the V&V of five selected fire models commonly used to support applications of RI/PB fire protection at NPPs. The seven volumes of this NUREG-series report provide technical documentation concerning the predictive capabilities of a specific set of fire dynamics calculation tools and fire phenomenological models that may be used for the analysis of fire hazards in postulated NPP scenarios. When used within the limitations of the fire models and considering the identified uncertainties, these models may be employed to demonstrate compliance with the requirements of 10 CFR 50.48(c).

Accordingly, for those FM elements performed by the licensee using the V&V applications contained in NUREG-1824, Supplement 1, to support the transition to NFPA 805, the NRC staff concludes that the use of these models is acceptable, provided that the intended application is within the appropriate limitations of the model, as identified in NUREG-1824, Supplement 1.

In Attachment J of the LAR, as supplemented, the licensee identified the use of several empirical correlations that are not addressed in NUREG-1824, Supplement 1 (see Section 3.4.3.3.1 of the SE). The NRC staff reviewed these correlations, as well as the related material provided in the LAR, in order to determine whether the licensee adequately demonstrated alignment with specific portions of the applicable NUREG-1824, Supplement 1, guidance.

The NRC staff concludes that the theoretical bases of the models and empirical correlations used in the FM calculations that were not addressed in NUREG-1824, Supplement 1, were identified by the NRC staff and described in authoritative publications, peer reviewed journal articles or conference papers, or national research laboratory reports (Reference 77), (Reference 78), (Reference 79), (Reference 81), (Reference 82), (Reference 83), (Reference 84), (Reference 85).

The FM employed by the licensee in the development of the 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 above, the NRC staff concludes that this approach provides reasonable assurance that the FM used in the development of the fire scenarios for the Hatch FREs is appropriate, and thus, acceptable for use in transition to NFPA 805 because the V&V of the empirical correlations used by the licensee were consistent with either NUREG-1824, Supplement 1, authoritative publications, peer reviewed journal articles, or national research laboratory reports.

3.9.3.3 Limitations of Use

Section 2.7.3.3 of NFPA 805, states that:

> Acceptable engineering methods and numerical models shall only be used for applications to the extent these methods have been subject to verification and validation. These engineering methods shall only be applied within the scope, limitations, and assumptions prescribed for that method.

In Section 4.7.3 of the LAR, the licensee 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 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 Section 3.9.3.3 of the SE, the NRC staff concluded 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 Hatch 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

Section 2.7.3.4 of NFPA 805, 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 NPPs, NPP fire protection, and power plant operations.

In Section 4.7.3 of the LAR, 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 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 Table S-3 of the LAR, as supplemented, as Implementation Item IMP-18.

Based on the statements in Section 4.7.3 of the LAR, the NRC staff has reasonable assurance that the licensee will comply with the requirements of NFPA 805, Section 2.7.3.4. Actions related to user qualification for FM are a requirement in the new FPP license condition (see Section 4.0 of the SE).

3.9.3.5 Uncertainty Analysis

Section 2.7.3.5 of NFPA 805, 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 89), 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 90).

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 Section 4.7.3 of the LAR, 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 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 provided by the licensee, the NRC staff concludes that the licensee performed an appropriate uncertainty analysis for the application and that there is reasonable assurance that the licensee will comply with NFPA 805, Section 2.7.3.5.

4.0 FIRE PROTECTION LICENSE CONDITION

As discussed in Section 2.4.2 of the SE, the licensee proposed a new FPP license condition (Attachment M of the LAR) (Reference 8), (Reference 12), to replace its current license conditions 2.C(3) 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, (Reference 4), with some plant-specific changes.

Overall, the proposed license condition provides structure and detailed criteria to allow the licensee to make changes to the Hatch FPP, without prior NRC approval, if the requirements of NFPA 805 (Reference 3), regarding engineering analyses, FREs, and PCEs 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 plant modifications and implementation items are completed, the risk analysis will not be consistent with the as-built, as-operated and as-maintained plant. The NRC staff's evaluation of the self-approval process for FPP changes (post-transition) is contained in Section 2.6 of the SE.

The proposed license condition also references the plant-specific modifications that must be completed for Hatch 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 are completed. However, as discussed in Section 2.7 of the SE, the licensee also credited the completion of the implementation items in Table S-3 of the LAR, as supplemented, in its analysis supporting the LAR. These implementation items must also be completed for Hatch to fully transition to NFPA 805. Therefore, the NRC staff will require the completion of these implementation items, in addition to the plant modifications, 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 Hatch license. These changes include spelling out some undefined acronyms, using scientific notation instead of calculator notation, and other editorial changes, such as re-numbering the FOL pages. In addition, the amendments approving the LAR are referenced instead of the SE. The NRC staff finds these editorial changes acceptable.

The following license condition is included in the revised renewed facility operating licenses and will replace Renewed Facility Operating License Nos. NPF-5, Condition 2.C(3), for Unit No. 1, and Renewed Facility Operating License Nos. DPR-57, Condition 2.C(3), for Unit No. 2:

(3) <u>Fire Protection</u>

Southern Nuclear Operating Company shall implement and maintain in effect all provisions of the approved fire protection program that comply with 10 CFR 50.48(a) and 10 CFR 50.48(c), as specified in *the* licensee amendment request dated April 4, 2018, supplemented by letters dated May 28, August 9, October 7, and December 13, 2019, and February 5, and March 13, 2020, and as approved in the NRC safety evaluation (SE) dated June 11, 2020. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

A risk assessment of the change must demonstrate that the acceptance criteria below are met. The risk assessment approach, methods, and data shall be acceptable to the NRC and shall be appropriate for the nature and scope of the change being evaluated; be based on the as-built, as-operated, and as-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.

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

(2) Prior NRC review and approval is not required for individual changes that result in a risk increase less than 1 $x10^{-7}$/year (yr) for CDF and less than 1 x 10^{-8} /yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

- 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 NRC safety evaluation report dated June 11, 2020, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

(c) Transition License Conditions

(1) Before achieving full compliance with 10 CFR 50.48(c), as specified by (2) 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 (b)(2) above.

(2) The licensee shall implement the modifications described in Attachment S2, Table S-2, "Plant Modifications Committed," of SNC letter NL-19-1475, dated December 13, 2019, to its facility to complete transition to full compliance with 10 CFR 50.48(c) by the startup of the second refueling outage (for each unit) after issuance of the NRC SE. The licensee shall maintain appropriate compensatory measures in place until completion of the modifications delineated above.

(3) The licensee shall implement the items as listed in Attachment S2, Table S-3, "Implementation Items," of SNC letter NL-19-1475, dated December 13, 2019, within 365 days after the issuance of the NRC SE. An exception to this statement is for the completion date for Implementation Item IMP-19. This item will be completed for each unit at a time not to exceed 180 days after all modifications for the respective unit (as listed in Attachment S2, Table S-2) are operable.

Based on the technical review in Section 3.0 of the SE above, the NRC staff finds it acceptable to replace the old Fire Protection license condition with the new Fire Protection license condition above.

5.0 SUMMARY

The NRC staff reviewed the LAR to transition Hatch 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 concluded the proposed change to TS 5.4.1 acceptable. As discussed in Section 2.4.2 of the SE, the licensee identified the need to revise the FPP license conditions 2.C(3) and provided a proposed new license condition. As discussed in Section 4.0 of the SE,

the NRC staff will impose a new FPP license condition to replace license conditions 2.C(3) in their 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 modifications in Table S-2 of the LAR, as supplemented, and the implementation items in Table S-3, as supplemented. As discussed in Section 4.0 of the SE, the NRC staff will impose a license condition with the issuance of the amendments to ensure that the modifications in Table S-2 of the LAR, as supplemented, and implementation items in Table S-3 of the LAR, as supplemented are completed in accordance with a schedule that is acceptable to the NRC staff.

In Attachment L of the LAR, 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 Section 3.1.4 of the SE, in accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff concluded that the proposed PB methods in Approval Requests 1–14 are acceptable alternatives to the corresponding NFPA 805, Chapter 3, requirements.

The license condition discussed in Section 4.0 of the SE, in conjunction with NFPA 805, will permit the licensee to make certain changes to the Hatch FPP without prior NRC approval. The license condition provides a structure and detailed criteria to allow the licensee to self-approve changes to the Hatch FPP if the requirements of NFPA 805 regarding engineering analyses, FREs, and PCEs 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 Hatch 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 Hatch 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 Hatch will meet the requirements of GDC 3, 10 CFR 50.48(a), and 10 CFR 50.48(c). In addition, the NRC staff finds it acceptable to re-number the FOL pages to accommodate the Fire Protection license condition.

As discussed in Section 2.5 of the SE, the NRC staff determined that the exemptions to 10 CFR Part 50, Appendix R, for Hatch will no longer be applicable if the proposed amendments to transition to NFPA 805 are approved. Therefore, the NRC staff concludes that it is acceptable to rescind the current exemptions to 10 CFR Part 50, Appendix R, identified by the licensee with the approval of the proposed license amendments.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the NRC staff notified the State of Georgia, official on March 5, 2020, of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on March 26, 2019 (84 FR 11340). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1 U.S. Nuclear Regulatory Commission, "Branch Technical Position (BTP) APCSB 9.5-1, Guidelines for Fire Protection for Nuclear Power Plants," May 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" (August 23, 1976)," February 24, 1977 (ADAMS Accession No. ML070660458).

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

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

5 U.S. Nuclear Regulatory Commission, "SECY-98-058: Development of a Risk-Informed, Performance-Based Regulation for Fire Protection at Nuclear Power Plants," March 26, 1998 (ADAMS Accession No. ML992910106).

6 U.S. Nuclear Regulatory Commission, "SECY-00-0009: Rulemaking Plan, Reactor Fire Protection Risk-Informed, Performance-Based Rulemaking," January 13, 2000 (ADAMS Accession No. ML003671923).

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

8 Gayheart, Cheryl, A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant - Unit 1 and 2, License Amendment Request to Adopt NFPA-805 Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," April 4, 2018 (ADAMS Accession Nos. Package ML18096A936 including ML18096A955; ML18096A956 and ML18096A957; and ML18096A959 non-publicly available).

9 Michael, D., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission Meier, "Edwin I. Hatch Nuclear Plant Units 1 & 2, Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) - NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating," May 28, 2019 (ADAMS Accession Nos.: Pacakge ML19151A444 including ML19151A421; and ML19151A422 non-publicly available).

10 Coleman, Jamie, M., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant, Units 1 & 2 - Supplemental Response to Request for Additional Information Regarding License Amendment Request for Transition to 10 CFR 50.48(c) - NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor," August 9, 2019 (ADAMS Accession Nos. Package ML19225D178 including ML19225D132; and ML19225D134 non-publicly available).

11 Gayheart, C. A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Units 1 and 2 Response to Request for

Additional Information Regarding Application to Adopt National Fire Protection Association Standard 805," October 7, 2019 (ADAMS Accession No. ML19280C812).

12 Gayheart, C. A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Units 1 and 2 Response to Request for Additional Information Regarding the License Amendment Request to Transition to 10 CFR 50.48(c) - NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating," December 13, 2019 (ADAMS Package Accession Nos.: Package ML19351D121 including ML19351D130; and ML19351D127 non-publicly available).

13 Gayheart, C. A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, "Edwin I. Hatch Nuclear Plant Units 1 and 2, Supplemental Response to RAI Regarding the License Amendment Request to Transition to 10 CFR 50.48(c) - NFPA 805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants," February 5, 2020 (ADAMS Accession No. ML20036F588).

14 Cheryl, A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission Gayheart, "Edwin I. Hatch Nuclear Plant Units 1 and 2, Supplemental Response Regarding the License Amendment Request to Transition to 10 CFR 50.48(c) - NFPA-805 Performance Based Standard for Fire Protection for Light Water Reactor Generating Plants," March 13, 2020 (ADAMS Accession No. ML20073M564).

15 Lamb, John, U.S. Nuclear Regulatory Commission, E-mail to Lowery, Ken, G., and Burns, Pamela Diane, Southern Nuclear Operating Company, "Hatch 1 and 2 - LAR to Adopt NFPA 805 Fire Protection Standard, EPID L-2018-LLA-0107, Request for Additional Information," March 29, 2019 (ADAMS Accession No. ML19088A009).

16 Lamb, John, U.S. Nuclear Regulatory Commission, E-mail to Enfinger, Timothy, Lee, Southern Nuclear Operating Company, L., "RAI - Edwin I. Hatch Nuclear Plant, LAR Regarding Adoption of NFPA 805, Request for Additional Information, EPID L-2018-LLA-0107," August 8, 2019 (ADAMS Accession No. ML19220A036).

17 Nuclear Energy Institute NEI 00-01, Revision 2, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Washington, DC, May 2009 (ADAMS Accession No. ML091770265).

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

19 U.S. Nuclear Regulatory Commission, "Regulatory Guide 1.200, Revision 2: An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," March 2009 (ADAMS Accession No. ML090410014).

20 American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) standard ASME/ANS RA-Sa-2009, "ASME/ANS RS-SA-2009: Addenda to ASME/ANS PRA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2, 2009.

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

22 U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Section 9.5.1.2, Revision 0 , "Standard Review Plan, "Risk-Informed, Performance-Based Fire Protection Program."," December 2009 (ADAMS Accession No. ML092590527).

23 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan," Chapter 19.1, Revision 3, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests after Initial Fuel Load," September 2012 (ADAMS Accession No. ML12193A107).

24 U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan," Chapter 19.2, "NUREG-0800, Standard Review Plan, 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).

25 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG/CR-6850, Final Report, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 1: Summary and Overview," EPRI - Palo Alto, CA, September 2005 (ADAMS Accession No. ML052580075).

26 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG/CR-6850, Final Report, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," EPRI - Palo Alto, CA, September 2005 (ADAMS Accession No. ML052580118).

27 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," EPRI - Palo Alto, CA, September 2010 (ADAMS Accession No. ML103090242).

28 Correia, Richard, P., memorandum to Joseph G. Giitter, U.S. Nuclear Regulatory Commission, "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," June 14, 2013 (ADAMS Package Accession No. ML13165A194).

29 U.S. Nuclear Regulatory Commission (NRC) and Sandia National Laboratories, NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE), Volumes 1, 2 and 3, Sandia National Laboratories - Albuquerque, NM," April 2008 (ADAMS Accession Nos. ML081190230, ML081190248, and ML081190261).

30 U.S. Nuclear Regulatory Commission and Sandia National Laboratories, NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results, Final Report, Sandia National Laboratories - Albuquerque, NM," April 2012 (ADAMS Package Accession No. ML121460107).

31 U.S. Nuclear Regulatory Commission, NUREG-1805, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program - Final Report," December 2004 (ADAMS Accession No. ML043290075).

32 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," 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 (FDS) EPRI - Palo Alto, CA, May 2007, (ADAMS Accession Nos. ML071650546, ML071730305, ML071730493, ML071730499, ML071730527, ML071730504, ML071730543, respectively).

33 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG-1824, Supplement 1, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications, Final Report, EPRI - Palto Alto, CA," November 2016 (ADAMS Accession No. ML16309A011).

34 U.S. Nuclear Regulatory Commission (NRC), National Institute of Standards and Technology (NIST) and University of Dayton Research Institute, NUREG/CR-7010, Volume 1, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," NIST - Gaithersburg, MD; University of Dayton Research Institute - Dayton, OH, July 2012 (ADAMS Accession No. ML12213A056).

35 U.S. Nuclear Regulatory Commission, Brooklyn National Laboratory, and Sandia National Laboratories, NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making - Main Report," Brooklyn National Laboratory - Upton, NY; Sandia National Laboratories - Albuquerque, NY, March 2009 (ADAMS Accession No. ML090970525).

36 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines - Final Report," EPRI - Palo Alto, CA, July 2012 (ADAMS Accession No. ML12216A104).

37 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG), Final Report," EPRI - Palo Alto, CA, November 2012 (ADAMS Accession No. ML12314A165).

38 U.S. Nuclear Regulatory Commission, Generic Letter 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations," April 10, 2006 (ADAMS Accession No. ML053620142).

39 National Fire Protection Association (NFPA), Standard 101, "Life Safety Code," NFPA 101, Quincy, Massachusetts.

40 National Fire Protection Association (NFPA), Standard 30, "Flammable and Combustible Liquids Code," NFPA 30, Quincy, Massachusetts.

41 National Fire Protection Association (NFPA), Standard 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hotwork," NFPA 51B, Quincy, Massachusetts.

42 National Fire Protection Association (NFPA), Standard 72, "National Fire Alarm and Signaling Code," NFPA 72, Quincy, Massachusetts.

43 National Fire Protection Association Standard (NFPA), Standard 76, "Standard for the Fire Protection of Telecommunications Facilities," NFPA 76, Quincy, Massachusetts.

44 National Fire Protection Association (NFPA), Standard 241, "Standard for Safeguarding Construction Alteration, and Demolition Operations," NFPA 241, Quincy, Massachusetts.

45 National Fire Protection Association (NFPA), Standard 262, "Standard Method of Test for Flame Travel and Smoke of Wires and Cables for Use in Air-Handling Spaces," NFPA 262, Quincy, Massachusetts.

46 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Frequently Asked 07-0030 on Establishing Recovery Actions," February 4, 2011 (ADAMS Accession No. ML110070485).

47 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Frequently Asked 07-0038 on Lessons Learned on Multiple Spurious Operations," February 3, 2011 (ADAMS Accession No. ML110140242).

48 Nuclear Energy Institute, NEI 00-01, Revision 1, "Guidance for Post Fire Safe Shutdown Circuit Analysis," Washington, DC, January 2005 (ADAMS Accession No. ML050310295).

49 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).

50 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).

51 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Frequently Asked 08-0054 on Demonstrating Compliance with Chapter 4 of National Fire Protection Association 805," March 10, 2015 (ADAMS Accession No. ML15016A280).

52 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 4, 2011 (ADAMS Accession No. ML102920405).

53 Casto, Greg, A., U.S. Nuclear Regulatory Commission, memorandum to APLB Files, "Close-out of National Fire Protection Association Standard 805 Frequently Asked Question 10-0059 On National Fire Protection Association Standard 805 Monitoring," August 7, 2018 (ADAMS Package Accession No. ML18208A454).

54 Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association 805 Frequently Asked Question 12-0064: Hot Work/Transient Fire Frequency Influence Factors," January 17, 2013 (ADAMS Accession No. ML12346A488).

55 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).

56 Hamzehee, Hossein, G., U.S. Nuclear Regulatory Commission, Memorandum to APLA Files, "Close-out of Fire Probabilistic Risk Assessment Frequently Asked Question 13-0005 on Cable Fires Special Cases: Self-Ignited and Caused by Wedling and Cutting," December 3, 2013 (ADAMS Accession No. ML13319B181).

57 Hamzehee, Hossein, G., U.S. Nuclear Regulatory Commission, Memorandum to APLA Files, "Close-out of Fire Probabilistic Risk Assessment Frequently Asked Question 14-0008 on Main Control Board Treatment," July 22, 2014 (ADAMS Accession No. ML14190B307).

58 U.S. Nuclear Regulatory Commission, letter to Whitmer, C.F., Georgia Power Company, "Forwards Amend #60 to Facil Oper Lic #DPR-57 rev Tech Specs w/Safety Evaluation & Notice of Issuance.," October 4, 1978 (ADAMS Legacy Library Accession No: 7810140046).

59 Stinson, L.M., Southern Nuclear Operating Company, Inc., letter to U.S. Nuclear Regulatory Commission, "Joseph M. Farley Nuclear Plant, Edwin I. Hatch Nuclear Plant, Voglte Electric Generating Plant, NRC Generic Letter 2006-03 Response," June 9, 2006 (ADAMS Accession No. ML061600376).

60 Electric Power Research Institute, "Technical Report TR-1006756 - Fire Protection Equipment Surveillance Optimization and Maintenance Guide, Final Report," Palo Alto, CA, July 2003.

61 American Society for Testing and Materials, "Standard E84 - Standard Test Method for Surface Burning Characteristics of Building Materials, ASTM-E84," West Conshohocken, Pennsylvania, 1983.

62 Institute of Electrical and Electronics Engineers (IEEE), "Standard 383 - Standard for Qualifying Class 1E Electric Cables and Field Splices for Nuclear Power Generating Stations," IEEE 383, New York, New York, 2003.

63 National Fire Protection Association (NFPA), Standard 70, "National Electric Code, NFPA 70," Quincy, Massachusetts.

64 National Fire Protection Association (NFPA), Standard 22, "Standard for Water Tanks for Private Fire Protection, NFPA 22," Quincy, Massachusetts.

65 National Fire Protection Association (NFPA), Standard 24, "Standard for the Installation of Private Fire Service Mains and Their Appurtenances, NFPA 24, ," Quincy, Massachusetts.

66 National Fire Protection Association (NFPA), Standard 20, "Standard for the Installation of Stationary Pumps For Fire Protection, NFPA 20," Quincy, Massachusetts.

67 National Fire Protection Association (NFPA), Standard 14, "Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems, NFPA 14," Quincy, Massachusetts.

68 U.S. Nuclear Regulatory Commission, NUREG-0800, Standard Review Plan, Revision 0, Section 9.5.1.1, "Fire Protection Program - Review Responsibilities," February 2009 (ADAMS Accession No. ML090510170).

69 Nuclear Energy Institute, NEI 05-04, Revision 2, "Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard," (Washington, DC, November 2008 (ADAMS Accession No. ML083430462).

70 Nuclear Energy Institute, "NEI 05-04/07-12/12-06 Appendix X: Close-Out of Facts and Observations (F&Os)," February 21, 2017 (ADAMS Accession No. ML17086A451).

71 Nuclear Energy Institute, NEI 07-12, Revision 1, "Fire Probabilistic Risk Assessment Peer Review Process Guidlines," Washington, DC, June 2010 (ADAMS Accession No. ML102230070).

72 Nuclear Energy Institute, NEI 12-13, "External Hazards PRA Peer Review Process Guidelines," August 2012 (ADAMS Accession No. ML12240A027).

73 Giitter, Joseph, and Ross-Lee, Mary Jane, U.S. Nuclear Regulatory Commission, letter to Mr. Greg Krueger, NEI , "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)," May 3, 2017 (ADAMS Accession No. ML17079A427).

74 Giitter, Joseph, U.S. Nuclear Regulatory Commission, letter to Mr. Biff Bradley, Nuclear Energy Institute (NEI), "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).

75 U.S. Nuclear Regulatory Commission (NRC), Science Applications International Corporation, and Sandia National Laboratories, NUREG-1792, "Good Practices for Implementing Human Reliability Analysis (HRA), Final Report," Science Aplications International Corporation - Lakewood, CO; Sandia National Laboratories - Albuquerque, NM, April 2005 (ADAMS Accession No. ML051160213).

76 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), NUREG-2178, Volume 1, "Refining And Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) - Peak Heat Release Rates and Effect of Obstructed Plume," December 2015 (ADAMS Accession No. ML152666A516).

77 Heskestad, Gunnar, "Chapter 13 - Fire Plumes, Flame Height, and Air Entrainment," in *Safety Fire Protection Engineering (SFPE) Handbook of Fire Protection Engineering, 5th Edition*, Morgan J. Hurley, Ed., Springer, New York, New York, 2016.

78 Beyler, Craig, L., "Chapter 66 - Fire Hazard Calculations for Large, Open Hydrocarbon Fires," in *Safety Fire Protection Engineering (SFPE) Handbook of Fire Protection Engineering, 5th Edition*, 5th ed., Morgan J. Hurley, Ed., Springer, New York, New York, 2016.

79 Alpert, Ronald, L., "Chapter 14 - Ceiling Jet Flows," in *Safety Fire Protection Engineering (SFPE) Handbook of Fire Protection Engineering, 5th Edition*, 5th ed., Morgan J. Hurley, Ed., Springer, New York, New York, 2016.

80 Electric Power Research Institute (EPRI), "Technical Report TR-100370 - Fire Induced Vulnerability Evaluation (FIVE), Revision 1," TR-100370, May 01, 1992.

81 National Institute of Standard and Technology (NIST), "NIST Special Publication 1041, Revision 6 - CFAST – Consolidated Model of Fire Growth and Smoke Transport (Version 6) Software Development and Model Evaluation Guide," NIST Special Publication 1041, Gaithersburg, MD, December 2008.

82 Custer, Richard, L.P.; Meacham, Brian, J.; and Schifiliti, Robert, P., "Chapter 40 - Design of Detection Systems," in *Safety Fire Protection Engineering (SFPE) Handbook of Fire Protection Engineering, 5th Edition*, 5th ed., Morgan J. Hurley, Ed., Springer, New York, New York, 2016.

83 Budnick, E.K., Evans, D.D., and Nelson, H.L., "Chapters 3 - 9: Simplified Fire Growth Calculations," in *National Fire Protection Association (NFPA) Fire Protection Handbook,*

19th Edition, 19th ed.: National Fire Protection Association (NFPA), Quincy, Massachusetts, 2003.

84 Lattimer, Brian, Y., "Chapter 25 - Heat Transfers from Fires to Surfaces," in *Safety Fire Protection Engineering (SFPE) Handbook of Fire Protection Engineering, 5th Edition*, 5th ed., Morgan J. Hurley, Ed., Springer, New York, New York, 2016.

85 National Institute of Standards and Technology (NIST), "CFAST - Consolidated Fire and Smoke Transport (Version 7) Volume 1: Technical Reference Guide," NIST Technical Note 1889 v1, Gaithersburg, MD, December 15, 2015.

86 Northard, Scott, Northern States Power Company, letter to U.S. Nuclear Regulatory Commission, "Prairie Island Nuclear Generating Plant Units 1 and 2, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactors - Response to Request for Additional Information (CACs ME9734 & ME9735)," March 6, 2017 (ADAMS Accession No. ML17065A339).

87 Gayheart, Cheryl, A., Southern Nuclear Operating Company, letter to U.S. Nuclear Regulatory Commission, Edwin I. Hatch Nulcear Plant Units 1 and 2, "Application to Adopt 10 CFR 50.69, Risk-informed Categrization and Treatment of Structures, Systems, and Components for Nuclear Power Reactors," June 7, 2018 (ADAMS Accession No. ML18158A583).

88 Marion, Alex, Nuclear Energy Institute (NEI), letter to Hannon, John, U.S. Nuclear Regulatory Commission, NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," June 2003 (ADAMS Accession No. ML031780500).

89 U.S. Nuclear Regulatory Commission, NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Main Report, March 2009 (ADAMS Accession No. ML090970525).

90 U.S. Nuclear Regulatory Commission (NRC) and Electric Power Research Institute (EPRI), "NUREG-1934, Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG), Final Report," EPRI - Palo Alto, CA, November 2012 (ADAMS Accession No. ML12314A165).

Principal Contributors: NRC
Jay Robinson, Naeem Iqbal, JS Hyslop, Thin Dinh

Pacific Northwest National Laboratories
Garill Coles

Date: June 11, 2020

Attachments:

A. Abbreviations and Acronyms

ATTACHMENT

Abbreviations and Acronyms

Acronym	Definition
AC	alternating current
ADAMS	Agencywide Documents Access and Management System
AEC	Atomic Energy Commission
AHJ	authority having jurisdiction
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
AOP	abnormal operating procedure
APCSB	Auxiliary and Power Conversion Systems Branch
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Material
BTP	Branch Technical Position
BWR	boiling water reactor
BWROG	Boiling Water Reactor Owners Group
°C	degrees Celsius
CAP	corrective action program
CBDTM	Cause Based Decision Tree Method
CC	capability category
CCDP	conditional core damage probability
CDF	core damage frequency
CFAST	consolidated model of fire and smoke transport
CFR	*Code of Federal Regulations*
CHRISTIFIRE	Cable Heat Release, Ignition, and Spread in Tray Installations During Fire
CLERP	conditional large early release probabilities
CR	control room
CR	Condition Report
CS	core spray
CSR	cable spreading room
CST	condensate storage tank
DHR	decay heat removal
DC	direct current
DG	diesel generator
DHR	decay heat removal
DID	defense-in-depth
ECCS	emergency core cooling system
EDG	emergency diesel generator
EEEE	existing engineering equivalency evaluation
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier system

Acronym	Definition
ERO	emergency response organization
°F	degrees Fahrenheit
F&O	fact and observation
FAQ	frequently asked question
FDS	fire dynamics simulator
FDT	fire dynamics tool
FLASH-CAT	Flame Spread over Horizontal Cable Tray
FM	Fire model or fire modeling
FMWB	Fire Modeling Workbook
FOL	Facility Operating License
FPE	fire protection engineering
FPP	fire protection program
FPRA	fire probabilistic risk assessment
FR	*Federal Register*
FRE	fire risk evaluation
FSAR	final safety analysis report
GAL	gallon
GDC	General Design Criterion/Criteria
GL	Generic Letter
HDPE	high density polyethylene
HEAF	high energy arcing fault
HEP	human error probability
HFE	human failure event
HGL	hot gas layer
HPCI	high pressure coolant injection
HRA	human reliability analysis
HRE	high(er) risk evolution
HRR	heat release rate
HRRPUA	heat release rate per unit area
HVAC	heating, ventilation, and air conditioning
IA	independent assessment
IEEE	Institute of Electrical and Electronics Engineers
IEPRA	internal events probabilistic risk assessment
KSF	key safety function
LAR	license amendment request
LERF	large early release frequency
LOC	loss of control
LOH	loss of habitability
LPCI	low pressure coolant injection
LSA	low-specific activity
MCA	multi-compartment analysis
MCB	main control board

Acronym	Definition
MCC	motor control center
MCR	main control room
MHIF	multiple high impedance faults
MSO	multiple spurious operations
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NPO	non-power operations
NPP	nuclear power plant
NPSH	net positive suction head
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSCA	nuclear safety capability assessment
NSPC	nuclear safety performance criteria
OMA	operator manual action
OSP	offsite power
PA	public address
PAU	physical analysis unit
PB	performance-based
PCE	plant change evaluation
PCS	primary control station
PORV	power-operated relief valve
POS	plant operating state
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSI	pounds per square inch
PSW	plant service water
PVC	polyvinyl chloride
PWR	Pressurized water reactor
QA	quality assurance
RA	recovery action
RAI	request for additional information
RASCAL	Radiological Assessment System for Consequence Analysis
RCA	radiologically controlled area
RCIC	reactor core isolation cooling
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	residual heat removal
RI	Risk-Informed
RI/PB	Risk-Informed, Performance-Based
RPV	reactor pressure vessel
RSP	remote shutdown panel

Acronym	Definition
SBLC	standby liquid control
SE	safety evaluation
SER	safety evaluation report
SFPE	Society of Fire Protection Engineers
SNC	southern nuclear operating company
SR	supporting requirement
SRV	safety relief valve
SSA	safe shutdown analysis
SSC	structures, systems, and components
SSD	safe shutdown
SSEL	safe shutdown equipment list
SW	service water
TS	technical specifications
UAMs	unreviewed analysis method
UFSAR	updated final safety analysis report
UL	Underwriters Laboratories
V&V	verification and validation
VFDR	variance from deterministic requirement
yr	year
ZOI	zone of influence

SUBJECT: EDWIN I. HATCH NUCLEAR PLANT, UNIT NOS. 1 AND 2, ISSUANCE OF AMENDMENTS NOS. 304 AND 249, REGARDING LICENSE AMENDMENT REQUEST TO ADOPT NFPA-805 PERFORMANCE BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR ELECTRIC GENERATING PLANTS (2001 EDITION) (EPID L-2018-LLA-0107) DATED JUNE 11, 2020

DISTRIBUTION:
Public
LPL2-1 File Copy
RidsACRS_MailCTR Resource
RidsRgn2Mail Center Resource
RidsNrr [Tech Branch] Resource
RidsNrrDraAplb Resource
RidsNrrPMHatch Resource (hard copy)
RidsNrrLAKGoldstein Resource
RidsNrrDorlLpl2-1 Resource

ADAMS Accession No. ML20066F592 *via memo **via email

OFFICE	NRR/LPL2-1/PM	NRR/LPL2-1/LA	NRR/DRA/APLB/BC
NAME	JLamb	KGoldstein**	JWhitman*
DATE	03/05/2020	04/03/2020	03/04/2020
OFFICE	OGC - NLO	NRR/LPL2-1/BC	NRR/LPL2-1/PM
NAME	AGhoshNaber**	MMarkley**	JLamb
DATE	05/07/2020	06/11/2020	06/11/2020

OFFICIAL RECORD COPY