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

January 28, 2015

Mr. William R. Gideon, Vice President
Brunswick Steam Electric Plant
P.O. Box 10429
Southport, NC 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 – ISSUANCE OF AMENDMENT REGARDING TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c) (TAC NOS. ME9623 AND ME9624)

Dear Mr. Gideon:

The U.S. Nuclear Regulatory Commission (NRC) has issued the enclosed Amendment No. 266 to Renewed Facility Operating License No. DPR-71, and Amendment No. 294 to Renewed Facility Operating License No. DPR-62, for the Brunswick Steam Electric Plant (Brunswick), Units 1 and 2, respectively. The amendments change the license and Technical Specifications (TSs) of the units in response to your application dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014. Carolina Power & Light Company (now Duke Energy Progress, Inc. (the licensee)), submitted a license amendment request to revise the fire protection program in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.48(c), for Brunswick and to change the license and TSs accordingly. The application dated September 25, 2012, and several of the supplements, were submitted before the name of the licensee in the operating licenses was changed to "Duke Energy Progress, Inc." on October 21, 2013.

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

The amendments revise the fire protection license condition in each unit's license and TS 5.4.1.d. As a result of placing the new license condition in each unit's license, the NRC is issuing additional pages for each unit due to repagination of subsequent license pages. The only changes to the licenses are the changes to the fire protection license condition.

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

Sincerely,

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-324 and 50-325

Enclosures:
1. Amendment No. 266 to DPR-71
2. Amendment No. 294 to DPR-62
3. Safety Evaluation

cc w/enclosures: Distribution via Listserv

ENCLOSURE 1

AMENDMENT NO. 266

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DUKE ENERGY PROGRESS, INC.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

DOCKET NO. 50-325

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-325

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 266
Renewed License No. DPR-71

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

   A. The application for amendment by Duke Energy Progress, Inc. (Duke Energy, the licensee), dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 266, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended as indicated in the attachment to this license amendment, and Paragraph 2.B.(6) of Renewed Facility Operating License No. DPR-71 is hereby amended to read as follows:

(6) Fire Protection

Duke Energy Progress, Inc. 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 license amendment request dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014, and as approved in the safety evaluation dated January 28, 2015. 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 Brunswick. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

   The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805,

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

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

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

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

   Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated January 28, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

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

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the licensee's fire protection program may not be made

without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the 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 safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-71
and Technical Specifications

Date of Issuance: January 28, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 266

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-71

DOCKET NO. 50-325

Replace the following pages of Renewed Facility Operating License No. DPR-71 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Pages 3 through 7	Pages 3 through 10

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

REMOVE	INSERT
5.0-5	5.0-5

(2) 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 Final Safety Analysis Report, as supplemented and amended;

(3) 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 as fission detectors in amounts as required;

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

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2;

(6) Fire Protection

Duke Energy Progress, Inc. 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 license amendment request dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014; and as approved in the safety evaluation dated January 28, 2015. 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 Brunswick. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

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

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

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

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

   Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated January 28, 2015, 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. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the 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 safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation of LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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 1: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and it subject to all applicable provisions hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts thermal.

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 266, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 203 to Renewed Facility Operating License DPR-71, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 203. For SRs that existed prior to Amendment 203 including SRs with modified acceptance criteria and SRs whose frequency of

performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 203.

(a) Effective June 30, 1982, the surveillance requirements listed below need not be completed until July 15, 1982. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2 shall apply.

Specification 4.3.3.1, Table 4.3.3-1, Items 5.a and 5.b

(b) Effective July 1, 1982, through July 8, 1982, Action statement "a" of Technical Specification 3.8.1.1 shall read as follows:

ACTION:

a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within two hours and at least once per 12 hours thereafter; restore at least two offsite circuits and four diesel generators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(3) Deleted by Amendment No. 206.

D. The licensee 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 plans, which contain Safeguards Information protected under 10 CFR 73.21 are entitled: "Physical Security Plan, Revision 2," and "Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006, and "Guard Training and Qualification Plan, Revision 0," submitted by letter dated September 30, 2004.

The licensee 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 licensee's CSP was approved by License Amendment No. 258, as supplemented by changes approved by License Amendment Nos. 261 and 265.

E. This license is subject to the following additional conditions for the protection of the environment:

a. Deleted per Amendment 54, 3-11-83

b. Deleted per Amendment 54, 3-11-83

c. The licensee shall comply with the effluent limitations contained in National Pollutant Discharge Elimination System Permit No. NC0007064

issued pursuant to Section 402 of the Federal Water Pollution Control Act, as amended.

F. In accordance with the requirement imposed by the October 8, 1976, order of the United States Court of Appeals for the District of Columbia Circuit in Natural Resources Defense Council v. Nuclear Regulatory Commission, No. 74-1385 and 74-1586, that the Nuclear Regulatory Commission "shall make any licenses granted between July 21, 1976 and such time when the mandate is issued subject to the outcome of the proceedings herein," the license issued herein shall be subject to the outcome of such proceedings.

G. Deleted by Amendment No. 206.

H. This license is effective as of the date of issuance and shall expire at midnight on September 8, 2036.

I. Deleted per Amendment No. 70 dated 5-25-84.

J. Deleted per Amendment No. 70 dated 5-25-84.

K. Deleted by Amendment No. 206.

L. Power Uprate License Amendment Implementation

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment (Amendment No. 183):

(1) Deleted by Amendment No. 206.

(2) Deleted by Amendment No. 206.

(3) Fuel Pool Decay Heat Evaluation

The decay heat loads and the decay heat removal systems available for each refueling outage shall be evaluated, and bounding or outage specific analyses shall be used for various refueling sequences. Where a bounding engineering evaluation is in place, a refueling specific assessment shall be made to ensure that the bounding case encompasses the specific refueling sequence. In both cases (i.e., bounding or outage specific evaluations), compliance with design basis assumptions shall be verified.

(4) Deleted by Amendment No. 206.

(5) Deleted by Amendment No. 206.

M. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, CP&L* may make changes to the programs and activities described in the supplement without prior Commission approval, provided that CP&L* evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

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*On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc."

N. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Duke Energy Progress, Inc. shall complete these activities no later than September 8, 2016, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

O. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

P. Mitigation Strategy License Condition

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

(1) 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

(2) 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

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

Q. 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.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 262, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachments:
1. Unit 1 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

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, Appendix A, November 1972;

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

c. Quality assurance for effluent and environmental monitoring; and

d. All programs and manuals specified in Specification 5.5.

ENCLOSURE 2

AMENDMENT NO. 294

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DUKE ENERGY PROGRESS, INC.

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

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

DUKE ENERGY PROGRESS, INC.

DOCKET NO. 50-324

BRUNSWICK STEAM ELECTRIC PLANT, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 294
Renewed License No. DPR-62

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

   A. The application for amendment by Duke Energy Progress, Inc. (Duke Energy, the licensee), dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 294, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

In addition, the license is amended as indicated in the attachment to this license amendment, and Paragraph 2.B.(6) of Renewed Facility Operating License No. DPR-62 is hereby amended to read as follows:

(6) Fire Protection

Duke Energy Progress, Inc. 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 license amendment request dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014; and as approved in the safety evaluation dated January 28, 2015. 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 Brunswick. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

   The licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805,

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

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

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

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

   Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated January 28, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

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

1. Before achieving full compliance with 10 CFR 50.48(c), as specified by 2. below, risk-informed changes to the

licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the 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 safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window; then, in that case, completion of the implementation items, except Item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

3. This license amendment is effective as of its date of issuance and shall be implemented within 180 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Shana R. Helton

Shana R. Helton, Chief
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Renewed Facility
Operating License No. DPR-62
and Technical Specifications

Date of Issuance: January 28, 2015

ATTACHMENT TO LICENSE AMENDMENT NO. 294

TO RENEWED FACILITY OPERATING LICENSE NO. DPR-62

DOCKET NO. 50-324

Replace the following pages of Renewed Facility Operating License No. DPR-62 with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
Page 3 through 8	Page 3 through 10

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

REMOVE	INSERT
5.0-5	5.0-5

(3) 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 as fission detectors in amounts as required;

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

(5) Pursuant to the Act and 10 CFR Parts 30 and 70 to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Brunswick Steam Electric Plant, Unit Nos. 1 and 2, and H. B. Robinson Steam Electric Plant, Unit No. 2.

(6) Fire Protection

Duke Energy Progress, Inc. 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 license amendment request dated September 25, 2012, as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014; and as approved in the safety evaluation dated January 28, 2015. 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 Brunswick. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

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

component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

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

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

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

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

   Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated January 28, 2015, 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. below, risk-informed changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact, as described in 2. above.

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the 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 safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications

3. The licensee shall complete all implementation items, except Item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window, then in that case completion of the implementation items, except Item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation LAR Attachment S, Table S-2, Item 9 within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

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

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2923 megawatts (thermal).

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 294, are hereby incorporated in the license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Technical Specifications.

For Surveillance Requirements (SRs) that are new in Amendment 233 to Renewed Facility Operating License DPR-62, the first performance is due at the end of the first surveillance interval that begins at implementation of Amendment 233. For SRs that existed prior to Amendment 233,

including SRs with modified acceptance criteria and SRs whose frequency of performance is being extended, the first performance is due at the end of the first surveillance interval that begins on the date the Surveillance was last performed prior to implementation of Amendment 233.

(a) The end of the current surveillance period for the surveillance requirements listed below may be extended beyond the time limit specified by Technical Specification 4.0.2a. After May 1, 1982, the plant shall not be operated in Conditions 1, 2, or 3 until the surveillance requirements listed below have been completed. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2a shall apply.

Specification 4.3.1.1; Table 4.3.1-1, items 9 & 10
4.3.1.2
4.3.1.3; Table 3.3.1-2, item 10
4.3.2.1; Table 4.3.2-1, items 1.d & 1.f
4.3.2.3; Table 3.3.2-3, item 1.a.1
4.3.3.2; Table 4.3.3-1, items 4.c & 4.f
4.5.2.a
4.8.1.1.2.d.2
4.8.1.1.2.d.3
4.8.1.1.2.d.6
4.8.1.1.2.d.7

(b) Effective June 30, 1982, the surveillance requirements listed below need not be completed until restart for Cycle 5 or July 15, 1982, whichever occurs first. The unit shall not be operated in Conditions 1, 2 or 3 until the surveillance requirements listed below have been completed. Upon accomplishment of the surveillances, the provisions of Technical Specification 4.0.2 shall apply.

Specification 4.3.3.1 Table 4.3.3-1, Items 5.a and 5.b.

(c) Effective July 1, 1982, through July 8, 1982, Action statement "a" of Technical Specification 3.8.1.1 shall read as follows:

<u>ACTION</u>:

a. With either one offsite circuit or one diesel generator of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirements 4.8.1.1.1.a and 4.8.1.1.2.a.4 within two hours and at least once per 12 hours thereafter; restore at least two offsite circuits and four diesel generators to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

(3) Deleted by Amendment No. 236.

(4) Equalizer Valve Restriction

The valves in the equalizer piping between the recirculation loops shall be closed at all times during reactor operation, except for one bypass valve which is left open to prevent pressure build-up due to ambient and conduction heating of the water between the equalizer valves.

(5) Deleted by Amendment No. 233.

(6) The licensee 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 plans, which contain Safeguards Information protected under 10 CFR 73.21, are entitled: "Physical Security Plan, Revision 2," and "Safeguards Contingency Plan, Revision 2," submitted by letter dated May 17, 2006, and "Guard Training and Qualification Plan, Revision 0," submitted by letter dated September 30, 2004.

The licensee 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 licensee's CSP was approved by License Amendment No. 286, as supplemented by changes approved by License Amendment Nos. 289 and 293.

D. This license is subject to the following additional conditions for the protection of the environment:

a. Deleted per Amendment 79, 3-11-83

b. Deleted per Amendment 79, 3-11-83

c. Deleted per Amendment 79, 3-11-83

d. The licensee shall comply with the effluent limitations contained in National Pollutant Discharge Elimination System Permit No. NC0007064 issued pursuant to Section 402 of the Federal Water Pollution Control Act, as amended.

E. This license is effective as of the date of issuance and shall expire at midnight on December 27, 2034.

F. Deleted per Amendment No. 98 dated 5-25-84.

G. Deleted per Amendment No. 98 dated 5-25-84.

H. Deleted by Amendment No. 236.

I. Power Uprate License Amendment Implementation

The licensee shall complete the following actions as a condition of the approval of the power uprate license amendment (Amendment No. 214):

(1) Deleted by Amendment No. 236.

(2) Deleted by Amendment No. 236.

(3) Fuel Pool Decay Heat Evaluation

The decay heat loads and the decay heat removal systems available for each refueling outage shall be evaluated, and bounding or outage specific analyses shall be used for various refueling sequences. Where a bounding engineering evaluation is in place, a refueling specific assessment shall be made to ensure that the bounding case encompasses the specific refueling sequence. In both cases (i.e., bounding or outage specific evaluations), compliance with design basis assumptions shall be verified.

(4) Deleted by Amendment No. 236.

(5) Deleted by Amendment No. 236.

J. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the UFSAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, CP&L* may make changes to the programs and activities described in the supplement without prior Commission approval, provided that CP&L* evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

K. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. Duke Energy Progress, Inc. shall complete these activities no later than December 27, 2014, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

L. All capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. Any changes to the BWRVIP ISP capsule withdrawal schedule, including spare capsules, must be approved by the NRC prior to implementation. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

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* On April 29, 2013, the name "Carolina Power & Light Company" (CP&L) was changed to "Duke Energy Progress, Inc."

M. Mitigation Strategy License Condition

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

(1) 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

(2) 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

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

N. 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.

3. Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 290, are hereby incorporated into this license. Duke Energy Progress, Inc. shall operate the facility in accordance with the Additional Conditions.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

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

Attachments:
1. Unit 2 – Technical Specifications – Appendices A and B

Date of Issuance: June 26, 2006

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, Appendix A, November 1972;

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

c. Quality assurance for effluent and environmental monitoring; and

d. All programs and manuals specified in Specification 5.5.

ENCLOSURE 3

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED

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

AMENDMENT NOS. 266 AND 294 TO RENEWED FACILITY OPERATING

LICENSE NOS. DPR-71 AND DPR-62

DUKE ENERGY PROGRESS, INC.

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 AND 50-324

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SAFETY EVALUATION

ATTACHMENTS

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

SAFETY EVALUATION BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

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BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2

DOCKET NOS. 50-325 and 50-324

1.0 INTRODUCTION

1.1 Background

The U.S. Nuclear Regulatory Commission (NRC) started developing fire protection requirements in the 1970s, and in 1976, the NRC published comprehensive fire protection guidelines in the form of Branch Technical Position (BTP) 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 evaluation reports (SERs) or supplements to SERs. In 1980, to resolve issues identified in those reports, the NRC amended its regulations for fire protection in operating nuclear power plants and published its Final Rule, Fire Protection Program for Operating Nuclear Power Plants, in the *Federal Register* (FR) on November 19, 1980 (45 FR 76602), adding Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.48, "Fire Protection," and Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." Section 50.48(a)(1) requires each holder of an operating license, and holders of a combined operating license issued under 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; identify the positions responsible for the program and the authority delegated to those positions; outline the plans for fire protection, fire detection and suppression capability, and limitation of fire damage. Section 50.48(a)(2) states that the fire protection plan must describe the specific features necessary to implement the program described in paragraph (a)(1) including administrative controls and personnel requirements; automatic and manual fire detection and suppression systems; and the means to limit fire damage to

structures, systems, and components (SSCs) to ensure the capability to safely shut down the plant. Section 50.48(a)(3) requires that the licensee retain the fire protection plan and each change to the plan as a record until the Commission terminates the license, 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 fire protection program (FPP) design requirements and elements, determining required fire protection systems and features, applying PB requirements, and administering fire protection for existing light-water reactors during operation, decommissioning, and permanent shutdown. It provides for the establishment of a minimum set of fire protection requirements, but allows PB or deterministic approaches to be used to meet performance criteria.

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

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

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, 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, a 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, and fire modeling calculations to show that the criteria are met. Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the performance criteria.

It also specifies that at least one success path to achieve the nuclear safety performance criteria shall be maintained free of fire damage by a single fire.

RG 1.205 also states, in part, that:

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

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

RG 1.205 also states, in part, that:

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

RG 1.205 provides the NRC staff's position on NEI 04-02, Revision 2 (Reference 7), and offers additional information and guidance to supplement the NEI document and assist licensees in meeting the NRC's regulations in 10 CFR 50.48(c) related to adopting a risk-informed, performance-based (RI/PB) FPP. RG 1.205 endorses the guidance of NEI 04-02, Revision 2, subject to certain exceptions, as providing methods acceptable to the staff for adopting an FPP consistent with the 2001 edition of NFPA 805 and complying with the regulations in 10 CFR 50.48(c).

Accordingly, Duke Energy Progress, Inc. (Duke, the licensee), requested a license amendment to allow the licensee to revise the Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick) FPP in accordance with 10 CFR 50.48(c) and change the license and technical specifications (TSs) accordingly.

1.2 Requested Licensing Action

By letter to the NRC dated September 25, 2012 (Reference 8), as supplemented by letters dated December 17, 2012 (Reference 9); June 28, 2013 (Reference 10); July 15, 2013 (Reference 11); July 31, 2013 (Reference 12); August 29, 2013 (Reference 13); September 30, 2013 (Reference 14); February 28, 2014 (Reference 15); March 14, 2014 (Reference 16); April 10, 2014 (Reference 17); June 26, 2014 (Reference 18); August 15, 2014 (Reference 19); August 29, 2014 (Reference 20); November 20, 2014 (Reference 21); and December 18, 2014

(Reference 22), the licensee submitted an application for a license amendment to transition the Brunswick FPP from 10 CFR 50.48(b) to 10 CFR 50.48(c), NFPA 805, "Performance-Based Standard for Fire Protection For Light Water Reactor Electric Generating Plants," 2001 Edition. The supplemental letters were in response to the NRC staff's requests for additional information (RAIs) dated May 15, 2013 (Reference 23); January 14, 2014 (Reference 24); February 12, 2014 (Reference 25); June 4, 2014 (Reference 26); and July 24, 2014 (Reference 27). The licensee's supplemental letters dated June 28, July 15, July 31, August 29 and September 30, 2013; and February 28, March 14, April 10, June 26, August 15, August 29, November 20, and December 18, 2014, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards consideration determination as published in the *Federal Register* (FR) on August 13, 2013 (78 FR 49300).

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

Specifically, the licensee requested to transition from the existing deterministic fire protection licensing basis - established in accordance with all provisions of the approved FPP as described in the Brunswick Final Safety Analysis Report (FSAR) and as approved in the SEs dated November 22, 1977 (Reference 28), as supplemented April 2, 1979 (Reference 29); June 11, 1980 (Reference 30); December 30, 1986 (Reference 31); December 6, 1989 (Reference 32); July 28, 1993 (Reference 33); and February 10, 1994 (Reference 34) to a risk-informed, performance-based (RI/PB) fire protection program in accordance with 10 CFR 50.48(c), that uses risk information, in part, to demonstrate compliance with the fire protection and nuclear safety goals, objectives, and performance criteria of NFPA 805. As such, the proposed FPP at Brunswick is referred to as RI/PB throughout this SE.

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

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

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

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

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

The licensee proposed a new fire protection license condition reflecting the new RI/PB FPP licensing basis, as well as revisions to the TSs that address this change to the current FPP licensing basis. Section 2.4.2 and Section 4.0 of this SE discuss in detail the license condition, and Section 2.4.3 discusses the TS changes.

2.0 REGULATORY EVALUATION

Section 50.48, "Fire Protection," of 10 CFR provides the NRC requirements for nuclear power plant fire protection. Section 50.48 includes specific requirements for requesting approval for an RI/PB FPP based on the provisions of NFPA 805 (Reference 3). Section 50.48(c)(3)(i) of 10 CFR states, in part, that:

> A licensee may maintain a fire protection program that complies with NFPA 805 as an alternative to complying with [10 CFR 50.48(b)] for plants licensed to operate before January 1, 1979, or the fire protection license conditions for plants licensed to operate after January 1, 1979. The licensee shall submit a request to comply with NFPA 805 in the form of an application for license amendment under [10 CFR] 50.90. The application must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof.

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

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

The intent of 10 CFR 50.48(c)(3)(ii) is given in the statement of considerations for the Final Rule, Voluntary Fire Protection Requirements for Light Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative (69 FR 33536, 33548; June 16, 2004), which states, in part, that:

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

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

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

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

Alternatively, licensees who want to use RI or PB alternatives to comply with NFPA 805 must obtain approval by submitting an LAR as required in 10 CFR 50.48(c)(4). This regulation further provides that:

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

In addition to the conditions outlined by the rule that require licensees to submit an LAR for NRC review and approval in order to adopt an RI/PB FPP, a licensee may submit additional elements of its FPP for which it wishes to receive specific NRC review and approval, as set forth in Regulatory Position C.2.2.1 of RG 1.205 (Reference 4). Inclusion of these elements in the NFPA 805 LAR is meant to alleviate uncertainty in portions of the current FPP licensing bases as a result of the lack of specific NRC approval of these elements. Regulatory guides are not

substitutes for regulations, and compliance with them is not required. Methods and solutions that differ from those set forth in 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 purpose of the FPP established by NFPA 805 is to provide assurance, through a defense-in-depth (DID) philosophy, that the NRC's fire protection objectives are satisfied. NFPA 805, Section 1.2, "Defense-in-Depth," states the following:

> Protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount to this standard. The fire protection standard shall be based on the concept of defense in depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:
>
> (1) Preventing fires from starting.
>
> (2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage.
>
> (3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

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

2.1 Applicable Regulations

The following regulations address fire protection:

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

  > Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed

to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

- GDC 5, "Sharing of structures, systems, and components," to 10 CFR Part 50, Appendix A:

  Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

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

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

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

2.2 Applicable Staff Guidance

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

- RG 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, issued December 2009 (Reference 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 provides exceptions to the NEI 04-02 guidance where required. Should a conflict occur between NEI 04-02 and this RG, the regulatory positions in RG 1.205 govern.

- The 2001 edition of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants" (Reference 3), specifies the minimum fire protection requirements for existing light-water nuclear power plants during all phases of plant operations, including shutdown, degraded conditions, and decommissioning, which had not been explicitly addressed by previous requirements and guidelines. NFPA 805 was developed to provide a comprehensive RI/PB standard for fire protection. The NFPA 805 Technical Committee on Nuclear Facilities is composed of nuclear plant licensees, the NRC, insurers, equipment manufacturers, and subject matter experts. The standard was developed in accordance with NFPA processes, and consisted of a number of technical meetings and reviews of draft documents by committee and industry representatives. The scope of NFPA 805 includes goals related to nuclear safety, radioactive release, life safety, and plant damage/business interruption. The standard addresses fire protection requirements for nuclear plants during all plant operating modes and conditions, including shutdown and decommissioning, which had not been explicitly addressed by previous requirements and guidelines. NFPA 805 became effective on February 9, 2001.

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

- NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 2 (Reference 35), 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 (MSOs). The risk-informed method is intended for application by licensees to determine the risk significance of identified circuit failure issues related to MSOs.

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 2, issued May 2011 (Reference 36), provides the NRC staff's

recommendations for using risk information in support of licensee-initiated licensing basis changes to a nuclear power plant that require such review and approval. The guidance provided does not preclude other approaches for requesting licensing basis changes. Rather, RG 1.174 is intended to improve consistency in regulatory decisions in areas in which the results of risk analyses are used to help justify regulatory action. As such, the RG provides general guidance concerning one approach that the NRC has determined to be acceptable for analyzing issues associated with proposed changes to a plant's licensing basis and for assessing the impact of such proposed changes on the risk associated with plant design and operation.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009 (Reference 38), provides guidance to licensees for use in determining the technical adequacy of the base probabilistic risk assessment (PRA) used in a 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 needed to support a regulatory submittal.

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

- American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 38), provides guidance for PRAs used to support RI decisions for commercial light-water reactor nuclear power plants and prescribes a method for applying these requirements for specific applications. The standard gives guidance for a Level 1 PRA of internal and external hazards for all plant operating modes. In addition, the standard provides guidance for a limited Level 2 PRA sufficient to evaluate large early release frequency (LERF). The only hazards explicitly excluded from the scope are accidents resulting from purposeful human-induced security threats (e.g., sabotage). The standard applies to PRAs used to support applications of 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 39), provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP. The NRC staff developed the RG to provide a comprehensive fire protection guidance document and to identify the scope and depth of fire protection that the staff would consider acceptable for nuclear power plants.

- NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (Reference 40), 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 41), 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 42), provides the NRC staff with guidance for evaluating the risk information used by a licensee to support permanent RI changes to the licensing basis.

- NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," Volumes 1 and 2 and Supplement 1 (References 43, 44, and 45), presents a compendium of methods, data, and tools to perform a fire probabilistic risk assessment (FPRA) and develop associated insights. In order to address the need for improved methods, the NRC Office of Nuclear Regulatory Research (RES) and Electric Power Research Institute (EPRI) embarked upon a program to develop a state-of-art FPRA methodology. Both RES and EPRI provided specialists in fire risk analysis, fire modeling (FM), electrical engineering, human reliability analysis, and systems engineering for methods development. A formal technical issue resolution process was developed to direct the deliberative process between RES and EPRI. The process ensures that divergent technical views are fully considered, yet encourages consensus at many points during the deliberation. Significantly, the process provides that each party maintain its own point of view if consensus is not reached. Consensus was reached on all technical issues documented in NUREG/CR-6850. The methodology documented in this report reflects the current state-of-the-art in FPRA. These methods are expected to form a basis for risk-informed analyses related to the plant FPP. Volume 1, the Executive Summary, provides general background and overview information, project insights and conclusions. Volume 2 provides the detailed discussion of the recommended approach, methods, data, and tools for conduct of an FPRA. Supplement 1 provides certain FPRA methods enhancements.

- Interim Technical Guidance provided in a Memorandum from Richard P. Correia, RES, to Joseph G. Giitter, NRR, titled "Interim Technical Guidance on Fire-Induced Circuit Failure Mode Likelihood Analysis," dated June 14, 2013 (Reference 46), discusses that, based on new experimental information documented in NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)" issued April 2008 (Reference 47), and NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," issued April 2012 (Reference 48), the effect of any control power transformer reduction to the hot short-induced spurious operation likelihood could not be substantiated.

- NUREG-1805, "Fire Dynamics Tools (FDT^s): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 49), provides quantitative methods, known as FDT^s, to assist regional fire protection inspectors in performing fire hazard analysis. The FDT^s are intended to assist fire protection inspectors in performing RI evaluations of credible fires that may cause critical damage to essential safe shutdown equipment.

- NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1 through 7 (Reference 50), provide technical documentation regarding the predictive capabilities of a specific set of fire models for the analysis of fire hazards in nuclear power plant scenarios. This report is the result of a collaborative program with the EPRI and the National Institute of Standards and Technology (NIST). The selected models are:

  (1) FDT^s developed by NRC (Volume 3);

  (2) The Fire-Induced Vulnerability Evaluation, Revision 1 (FIVE) developed by EPRI (Volume 4);

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

  (4) The zone model MAGIC developed by Electricite de France (Volume 6); and

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

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

- NUREG/CR-7010, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," Volume 1 (Reference 51), describes Phase 1 of the CHRISTIFIRE testing program

conducted by NIST. The overall goal of this multiyear program is to quantify the burning characteristics of grouped electrical cables installed in cable trays. This first phase of the program focuses on horizontal tray configurations. CHRISTIFIRE addresses the burning behavior of a cable in a fire beyond the point of electrical failure. The data obtained from this project can be used for the development of fire models to calculate the heat release rate (HRR) and flame spread of a cable fire.

- NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 52), provides guidance on how to treat uncertainties associated with PRA in RI decision-making. The objectives of this guidance include fostering an understanding of the uncertainties associated with PRA and their impact on the results of PRA and providing a pragmatic approach to addressing these uncertainties in the context of the decision-making. To meet the objective of the NUREG, it is necessary to understand the role that PRA results play in the context of the decision process. To define this context, NUREG-1855 provides an overview of the RI decision-making process itself.

- NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, Final Report" (Reference 53), presents the state-of-the-art in fire human reliability analysis (HRA) practice. This report was developed jointly between RES and EPRI to develop the methodology and supporting guidelines for estimating human error probabilities (HEPs) for human failure events 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 54), 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 nuclear power plant fire hazard analyses. The report also provides information to assist fire model users in applying this technology in the nuclear power plant environment.

- Generic Letter (GL) 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations" (Reference 55), requested that licensees evaluate their facilities to confirm compliance with the existing applicable regulatory requirements in light of the information provided in this GL and, if appropriate, take additional actions. Specifically, NRC testing revealed that, for the configurations tested, Hemyc and MT fire barriers failed to provide the protective function intended for compliance with existing regulations.

- NFPA 101, "Life Safety Code" (Reference 56), 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 74), provides requirements for the safe storage, handling, and use of flammable and combustible liquids.

- NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection" (Reference 57), provides requirements for the selection and installation of pumps to ensure that systems will work as intended to deliver adequate and reliable water supplies in a fire emergency.

- NFPA 14, "Standard for the Installation of Standpipe and Hose Systems" (Reference 58), provides the minimum requirements for the installation of standpipes and hose systems to ensure that systems will work as intended to deliver adequate and reliable water supplies in a fire emergency. NFPA 14 covers all system components and hardware, including piping, fittings, valves, and pressure-regulation devices, as well as system requirements; installation requirements; design; plans and calculations; water supply; and system acceptance.

- NFPA 10, "Standard for Portable Fire Extinguishers" (Reference 59), provides requirements to ensure that portable fire extinguishers will work as intended to provide a first line of defense against fires of limited size.

2.3 NFPA 805 Frequently Asked Questions

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

Table 2.3-1, NFPA 805 Frequently Asked Questions

FAQ #	FAQ Title and Summary	Reference No.	SE Section
06-0022	Electrical Cable Flame Propagation Tests • This FAQ provides a list of acceptable cable flame propagation tests.	60	3.1.1.2

FAQ #	FAQ Title and Summary	Reference No.	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 or at primary control station(s). ▪ Determination of which RAs are required by the NFPA 805 fire protection program. ▪ Evaluate the additional risk presented by the use of RAs. ▪ Evaluate the feasibility of the identified RAs. ▪ Evaluate the reliability of the identified RAs.	61	3.2.5
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 fire PRA and Nuclear Safety Capability Assessment to include MSOs of concern. ▪ Step 4 – Evaluate for NFPA 805 compliance. ▪ Step 5 – Document the results.	62	3.2.4
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 strategy to the endorsed industry guidance, NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 1. In short, the process has the licensee: ▪ Assemble industry and plant-specific documentation; ▪ Determine which sections of the guidance are applicable; ▪ Compare the existing safe shutdown methodology to the applicable guidance; and ▪ Document any discrepancies.	63	3.2.1
07-0040	"Non-Power Operations (NPO) Clarifications" • This FAQ clarifies an acceptable NFPA 805 NPO program. The process includes: ▪ Selecting NPO equipment and cabling. ▪ Evaluation of NPO Higher Risk Evolutions (HRE). ▪ Analyzing NPO key safety functions (KSF). ▪ Identifying plant areas to protect or "pinch points" during NPO HREs and actions to be taken if KSFs are lost.	64	3.5.3

FAQ #	FAQ Title and Summary	Reference No.	SE Section
08-0046	"Incipient Fire Detection Systems" • This FAQ provides guidance for modeling non-suppression probability when an incipient fire detection system is installed in electrical cabinets outside the Main Control Room.	65	3.2.6
08-0048	"Revised Fire Ignition Frequencies" • This FAQ provides an acceptable method for using updated fire ignition frequencies in the licensee's fire PRA. The method involves the use of sensitivity studies when the updated fire ignition frequencies are used.	66	3.4.7
08-0050	"Manual Non-Suppression Probability" • This FAQ updates the treatment of manual suppression and fire brigade response. The update includes a process to adjust the non-suppression analysis for scenario-specific fire brigade responses.	67	3.4.2.3.2
08-0054	"Demonstrating Compliance with Chapter 4 of NFPA 805" • This FAQ provides an acceptable process to demonstrate Chapter 4 compliance for transition: ▪ Step 1 – Assemble documentation ▪ Step 2 – Document Fulfillment of Nuclear Safety Performance Criteria ▪ Step 3 – Variance From Deterministic Requirements (VFDR) Identification, Characterization, and Resolution Considerations ▪ Step 4 – Performance-Based Evaluations ▪ Step 5 – Final VFDR Evaluation ▪ Step 6 – Document Required Fire Protection Systems and Features	68	3.4.3 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.	69	3.6

FAQ #	FAQ Title and Summary	Reference No.	SE Section
10-0059	"NFPA 805 Monitoring Program" • This FAQ provides clarification regarding the implementation of an NFPA 805 monitoring program for transition. It includes: ▪ Monitoring program analysis units; ▪ Screening of low safety significant SSCs; ▪ Action level thresholds; and ▪ The use of existing monitoring programs.	70	3.7

2.4 Orders, License Conditions, and Technical Specifications

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

2.4.1 Orders

The NRC staff reviewed LAR Section 5.2.3, "Orders and Exemptions," and LAR Attachment O, "Orders and Exemptions," with regard to NRC-issued Orders pertinent to Brunswick 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 Brunswick are maintained. The licensee discussed the affected orders and exemptions in LAR Attachment O.

The licensee requested that 14 exemptions be rescinded, and determined that no orders need to be superseded or revised to implement an FPP at Brunswick that complies with 10 CFR 50.48(c).

The licensee's review included an assessment of docketed correspondence files and electronic searches, including the NRC's Agencywide Documents Access and Management System (ADAMS). The review was performed to ensure that compliance with the physical protection requirements, security orders, and adherence to commitments applicable to Brunswick are maintained. The NRC staff accepts the licensee's determination that 14 exemptions should be rescinded as listed in LAR Attachment K, "Existing Licensing Action Transition," and that no Orders need to be superseded or revised to implement NFPA 805 at Brunswick. See SE Section 2.5 for the NRC staff's detailed evaluation of the exemptions being rescinded.

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

the mitigation measures required by 10 CFR 50.54(hh)(2) because 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 in regard to 10 CFR 50.54(hh)(2) is acceptable.

2.4.2 License Conditions

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

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

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

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

Overall, the licensee's proposed revised license condition allows self-approval for FPP changes that meet the requirements of NFPA 805 with regard to engineering analyses, fire risk evaluations (FREs), and plant change evaluations (PCEs). The NRC staff's evaluation of the self-approval process for FPP changes (post-transition) is contained in SE Section 2.6. The license condition also references the plant-specific modifications, and associated implementation schedules that must be accomplished at Brunswick to complete transition to NFPA 805 and comply with 10 CFR 50.48(c). In addition, the license condition includes a requirement that appropriate compensatory measures will remain in place until implementation of the specified plant modifications is completed. These modifications and implementation schedules are identical to those identified elsewhere in the LAR, as discussed by the NRC staff in Sections 2.7.1 and 2.7.2, and reviewed in Section 3.0, of this SE.

SE Section 4.0 provides the NRC staff's review of the proposed Brunswick FPP license condition.

2.4.3 Technical Specifications

The NRC staff reviewed LAR Section 5.2.2, "Technical Specifications," and LAR Attachment N, "Technical Specification Changes," with regard to proposed changes to the Brunswick TSs that are being revised or superseded during the NFPA 805 transition process. According to the LAR, the licensee conducted a review of the Brunswick TSs to determine which, if any, TS sections will be impacted by the transition to an RI/PB FPP based on 10 CFR 50.48(c). The licensee identified changes to the TSs needed for Brunswick adoption of the new fire protection licensing basis and provided applicable justification listed in LAR Attachment N.

The licensee identified one TS change that involves deleting TS 5.4.1.d, which requires that procedures be established, implemented, and maintained for FPP implementation. Specifically, the licensee stated that deleting TS 5.4.1.d is acceptable for adoption of the new fire protection licensing basis since the requirement for establishing, implementing, and maintaining fire protection procedures is contained in 10 CFR 50.48(a) and 10 CFR 50.48(c). The regulations in 10 CFR 50.48(c) approve the incorporation of NFPA 805 by reference and NFPA 805, Section 3.2.2, "Procedures," states that "Procedures shall be established for implementation of the fire protection program."

Based on the information provided by the licensee, the NRC staff concludes that the proposed deletion is acceptable because TS 5.4.1.d is an administrative control (i.e., a procedure the licensee puts in place to establish, implement, and maintain the fire protection program as required by the licensee's fire protection license condition and 10 CFR 50.48(a), 10 CFR 50.48(c), and NFPA 805, Section 3.2.3), and would be redundant to the NFPA 805 requirement to establish FPP procedures. NFPA 805 requires the licensee to establish FPP procedures, and 10 CFR 50.48(a) would become the fire protection licensing basis of Brunswick. In addition, failure by the licensee to establish FPP procedures would result in non-compliance with 10 CFR 50.48(c)(1), which is the licensee's fire protection licensing basis. Changes to fire protection administrative controls are controlled by the proposed fire protection license condition. See SE Section 4.0.

2.4.4 Updated Final Safety Analysis Report

The NRC staff reviewed LAR Section 5.4 "Revision to the UFSAR," which states:

> After the approval of the LAR, in accordance with 10 CFR 50.71(e), the Brunswick UFSAR will be revised. The content will be consistent with NEI 04-02.

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

2.5 <u>Rescission of Exemptions</u>

The NRC staff reviewed LAR Section 5.2.3, "Orders and Exemptions," Attachment O, "Orders and Exemptions," and LAR Attachment K, "Existing Licensing Action Transition," with regard to previously-approved exemptions to Appendix R to 10 CFR Part 50, which the transition to an

FPP licensing basis in conformance with NFPA 805 will supersede. These exemptions will no longer be required since upon approval of the RI/PB FPP in accordance with NFPA 805, Appendix R will not be part of the licensing basis for Brunswick.

The licensee previously requested and received NRC approval for 14 exemptions from 10 CFR Part 50 Appendix R. These exemptions were discussed in detail in LAR Attachment K. The licensee stated that the exemptions are either compliant with 10 CFR 50.48(c), or are no longer necessary because either an RI/PB analysis was performed and the configuration was determined to be not risk significant, or redundant equipment is available outside of the area to maintain and achieve safe shutdown, or there is no corresponding requirement under NFPA 805. The licensee requested, in accordance with the requirements of 10 CFR 50.48(c)(3)(i), that all the exemptions be rescinded.

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

- The exemption was found to be unnecessary since the underlying condition has been evaluated using RI/PB methods (FM and/or FRE) and found to be acceptable and no further actions are necessary by the licensee.

- The exemption was found to be appropriate as a qualitative engineering evaluation that meets the deterministic requirements of NFPA 805 and is carried forward as part of the engineering analyses supporting NFPA 805 transition.

The following exemptions, originally approved by NRC SEs dated November 22, 1977 (Reference 28); April 6, 1979 (Reference 71); June 11, 1980 (Reference 30); December 30, 1986 (Reference 31); December 6, 1989 (Reference 32); July 28,1993 (Reference 33); and February 10, 1994 (References 34), are rescinded as requested by the LAR and the underlying condition has been evaluated using RI/PB methods and found to be acceptable with no further actions because DID and sufficient safety margin is maintained:

- Exemption from the Appendix R, Section III.G.3 requirements for the installation of fire suppression systems in the control room and cable spreading rooms.

- Exemption from the Appendix R, Section III.G.3 requirements for the installation of fire suppression systems in the control building cable vaults.

- Exemption from the Appendix R, Section III.G.3 requirements for the Installation of automatic detection and fixed suppression requirements for the east yard.

- Exemption from the Appendix R, Section III.G.2 requirements for fire area RB1-1.

- Exemption from the Appendix R, Section III.G.2 requirements for fire area RB2-1.

- Exemption from the Appendix R, Section III.G.2 requirements for safe shutdown system separation for fire area DG-1.

- Exemption from the Appendix R, Section III.G.2 requirements for providing an automatic suppression system for fire area DG-8.

- Exemption from the Appendix R, Section III.G.3 requirements for alternative shutdown areas in the control and diesel generator buildings.

- Exemption from the Appendix R, Section III.J requirements for emergency lighting for fire zone CB-23 in fire area CB-23E.

- Exemption from the Appendix R, Section III.J requirements for emergency lighting for the east yard.

- Exemption from the Appendix R, Section III.G.3 requirements for the installation of fixed suppression for fire area TB1.

- Exemption from the Appendix R, Section III.G.2 requirements for the safe shutdown separation features and unrated penetrations for fire area RB1-6.

- Exemption from the Appendix R, Section III.G.2 requirements for the safe shutdown separation features and unrated penetrations for fire area RB2-6.

- Exemption from the Appendix R, Section III.G.2 requirements for separation of combustibles for elevations 4 feet and 20 feet of the service water building.

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

- None

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

Upon completion of the implementation of the RI/PB fire protection program and issuance of the license condition discussed in SE Section 2.4.2, changes to the approved FPP must be evaluated by the licensee to ensure that they are acceptable.

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

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

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

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

2.6.1 Post-Implementation Plant Change Evaluation Process

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

LAR Section 4.7.2 states that the PCE process will consist 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 licensee stated that the baseline is defined as that plant condition or configuration that is consistent with the design basis and licensing basis (NFPA 805 licensing basis post-transition) and that the changed or altered condition or configuration that is not consistent with the design basis and licensing basis is defined as the proposed alternative.

The licensee stated that once the definition of the change is established, a screening is then performed to identify and resolve minor changes to the FPP and that the screening will be consistent with fire protection regulatory review processes currently in place at nuclear plants under traditional licensing bases. The licensee stated that the screening process is modeled after the NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," (Reference 72), a process that will address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.). The licensee further stated in LAR Section 4.7.2 that if 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 not met, then the licensee will proceed to the risk evaluation step of the PCE process.

The licensee stated that the risk evaluation screening will be followed by engineering evaluations and that the results of the evaluations are 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 to the LAR) can be implemented within the framework provided by NFPA 805, and that changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The licensee further stated that the acceptance criteria require that the resultant change in core damage frequency (CDF) and LERF be consistent with the license condition, and that the acceptance criteria will 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 risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change. The licensee also stated that, in certain circumstance an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions, provided the use of such assumptions does not unnecessarily challenge the acceptance criteria.

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

The licensee stated that the fire protection program configuration is defined by the program documentation and that, to the greatest extent possible, the existing configuration control processes for modifications, calculations and analyses, and fire protection program license basis reviews will be utilized to maintain configuration control of the fire protection program documents. The licensee further stated that the configuration control procedures which govern the various Brunswick documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements. In LAR Attachment S, Table S-2, Implementation Item 4, the licensee included the action to "update configuration control procedures to reflect the new NFPA 805 licensing bases requirements." The NRC staff considers this action acceptable because it will result in compliance with NFPA 805 and because the action is required by the proposed license condition.

The licensee stated that several NFPA 805 document types, such as Nuclear Safety Capability Assessment (NSCA) Supporting Information and Non-Power Mode NSCA Treatment, 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 that the new procedures will be modeled after the existing processes for similar types of documents and databases. The licensee further stated that system level design basis documents will be revised to reflect the NFPA 805 role that the systems and components now play. In LAR Attachment S, Table S-2, Implementation Item 4, the licensee included the action to "update configuration control procedures to reflect the new NFPA 805 licensing bases requirements." The NRC staff considers this action acceptable because it will result in compliance with NFPA 805 and because the action is required by the proposed license condition.

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

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

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

The licensee stated that this process follows the requirements in NFPA 805 and the guidance outlined in RG 1.174 (Reference 36), which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered.

Since NFPA 805 always requires the use of a PCE, regardless of what element requires the change, the NRC staff concludes that, in accordance with the requirements of NFPA 805, if FPP impacts are determined to exist as a result of the proposed change, the issue would be resolved by utilizing the NFPA 805 change process developed in accordance with NEI 04-02, RG 1.205, and the Brunswick NFPA 805 fire protection license condition to assess the acceptability of the proposed change. This process will be used to determine if prior NRC approval of the proposed change is required.

Based on the information provided by the licensee, the NRC staff concludes that the licensee's PCE process is considered acceptable because it meets the guidance in NEI 04-02, Revision 2 (Reference 7), as well as RG 1.205, Revision 1 (Reference 4), and addresses attributes for using fire risk evaluations in accordance with NFPA 805. Section 2.4.4 of NFPA 805 requires that PCEs consist of an integrated assessment of risk, DID and safety margin. NFPA 805, Section 2.4.3.1 requires that the probabilistic safety assessment (PSA) use CDF and LERF as measures for risk. NFPA 805, Section 2.4.3.3 requires that the risk assessment approach, methods, and data shall be acceptable to the Authority Having Jurisdiction (AHJ), which is the NRC. NFPA 805, Section 2.4.3.3 also requires that the PSA be appropriate for the nature and scope of the change being evaluated, be based on the as-built and as-operated and maintained plant, and reflect the operating experience at the plant.

The licensee's PCE process includes the required delta risk calculations, uses risk assessment methods acceptable to the NRC, uses appropriate risk acceptance criteria in determining acceptability, involves the use of a fire PRA of acceptable quality, and includes an integrated assessment of risk, DID, and safety margin as discussed above.

2.6.2 Requirements for the Self-Approval Process Regarding Plant Changes

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

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

However, before achieving full compliance with 10 CFR 50.48(c) by completing the plant modifications and implementation items discussed in SE Section 2.7 (i.e., during full implementation of the transition to NFPA 805), the proposed license condition provides that RI changes to the licensee's fire protection program may not be made without prior NRC review and approval unless the changes have been demonstrated to have no more than a minimal risk impact using the screening process discussed above because the risk analysis is not consistent with the as-built, as-operated, and maintained plant since the items have not been completed. In addition, the condition requires the licensee to ensure that fire protection DID and safety margin are maintained during the transition process. The "Transition License Conditions" in the proposed NFPA 805 license condition include the appropriate acceptance criteria and other attributes to form an acceptable method for meeting Regulatory Position C.3.1 of RG 1.205, Revision 1 (Reference 4), with respect to the requirements for fire protection program changes during transition, and therefore, demonstrate compliance with 10 CFR 50.48(c).

The proposed NFPA 805 license condition also includes a provision for self-approval of changes to the fire protection program that may be made on a qualitative, rather than an RI basis. Specifically, the license condition states that prior NRC review and approval are not required for changes to the NFPA 805, Chapter 3 fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the NFPA 805, Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805, Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical

arrangement (i.e., has not impacted its contribution toward meeting the nuclear safety and radioactive release performance criteria), using a relevant technical requirement or standard.

Use of this approach does not fall under NFPA 805, Section 1.7, "Equivalency," because the condition can be shown to meet the NFPA 805, Chapter 3 requirement. Section 1.7 of NFPA 805 is a standard format used throughout NFPA standards. It is intended to allow owner/operators to use the latest state of the art fire protection features, systems, and equipment, provided the alternatives are of equal or superior quality, strength, fire resistance, durability, and safety. However, the intent is to require approval from the authority having jurisdiction because not all of these state-of-the-art features are in current use or have relevant operating experience. This is a different situation than the use of functional equivalency since functional equivalency demonstrates that the condition meets the NFPA 805 code requirement.

Alternatively, the licensee may use an engineering evaluation to demonstrate that changes to certain NFPA 805, Chapter 3 elements are acceptable because the changes are "adequate for the hazard." Prior NRC review and approval would not be required for alternatives to four specific sections of NFPA 805, Chapter 3 listed below, for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is adequate for the hazard. A qualified fire protection engineer shall perform the engineering evaluation and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement (with respect to the ability to meet the nuclear safety and radioactive release performance criteria), using a relevant technical requirement or standard. NFPA 805, Section 2.4, states that engineering analysis is an acceptable means of evaluating a fire protection program against performance criteria. Engineering analyses shall be permitted to be qualitative or quantitative. Use of qualitative engineering analyses by a qualified fire protection engineer to determine that a change has not affected the functionality of the component, system, procedure or physical arrangement is allowed by NFPA 805, Section 2.4.

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

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

The engineering evaluations described above (i.e., functionally equivalent and adequate for the hazard) are engineering analyses governed by the NFPA 805 guidelines. In particular, this means that the evaluations must meet the requirements of NFPA 805, Section 2.4, "Engineering Analyses," and NFPA 805, Section 2.7, "Program Documentation, Configuration Control, and Quality." Specifically, the effectiveness of the fire protection features under review must be evaluated and found acceptable in relation to their ability to detect, control, suppress, and extinguish a fire and provide passive protection to achieve the performance criteria and not exceed the damage threshold for the plant being analyzed. The associated evaluations must

also meet the documentation content (as outlined by NFPA 805, Section 2.7.1, "Content") and quality requirements (as outlined by NFPA 805, Section 2.7.3, "Quality") of the standard in order to be considered adequate. The NRC staff's review of the licensee's compliance with NFPA 805, Sections 2.7.1 and 2.7.3, is provided in SE Section 3.8.

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

The NRC staff also concludes that the fire risk evaluation methods used to model the cause and effect relationship of associated changes as a means of assessing the risk of plant changes during transition to NFPA 805 may continue to be used after implementation of the RI/PB fire protection program, based on the licensee's administrative controls to ensure that the models remain current and to provide assurance of continued quality (see SE Section 3.4.1, "Quality of the Fire Probabilistic Risk Assessment"). Accordingly, these cause-and-effect relationship models may be used after transition to NFPA 805 as a part of the PCEs conducted to determine the change in risk associated with proposed plant changes.

2.7 Modification and Implementation Items

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

The list of modifications and implementation items originally submitted in the LAR have been updated by the licensee in the final version of LAR Attachment S, "Plant Modifications and Items to be Completed during Implementation," provided in the licensee's letter dated November 20, 2014 (Reference 21).

2.7.1 Modifications

The NRC staff reviewed LAR Attachment S, "Modifications and Implementation Items," Table S-1, "Plant Modifications Committed," which describes the plant modifications necessary to implement the NFPA 805 licensing basis, as proposed. These modifications are identified in the LAR as necessary to bring Brunswick into compliance with either the deterministic or PB requirements of NFPA 805. As described below, LAR Attachment S, Table S-1, provides a description of each of the proposed plant modifications, presents the problem statement explaining why the modification is needed, and identifies the compensatory actions required to be in place pending completion of the modification.

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

LAR Attachment S, Table S-1 provides a detailed listing of the plant modifications that must be completed in order for Brunswick to be fully in accordance with NFPA 805, implement many of the attributes upon which this SE is based, and thereby meet the requirements of 10 CFR 50.48(c). The modifications will be completed in accordance with the schedule provided in the proposed NFPA 805 license condition, which states that the modifications will be completed by the startup of the second refueling outage for each unit after issuance of the SE and that appropriate compensatory measures will be maintained until the modifications are complete.

2.7.2 Implementation Items

Implementation items are items that the licensee has not fully completed or implemented as of the issuance date of the license amendment, but which will be completed during implementation of the license amendment to transition to NFPA 805 (e.g., procedure changes that are still in process, or NFPA 805 programs that have not been fully implemented). The licensee identified the implementation items in LAR Attachment S, Table S-2. For each implementation item, the licensee and the NRC staff have reached a satisfactory resolution involving the level of detail and main attributes that each remaining change will incorporate upon completion. Completion of these items in accordance with the schedule discussed in SE Section 2.7.3 does not change or impact the bases for the safety conclusions made by the NRC staff in the SE.

Each implementation item will be completed prior to the deadline for implementation of the RI/PB fire protection program based on NFPA 805, as specified in the license condition and the letter transmitting the amended license (i.e., implementation period) which states that implementation items, except item 9, described in LAR Attachment S, Table S-2, will be completed within 180 days after NRC approval unless the 180th day falls within an outage window; then in that case, completion of implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The license condition further states that the licensee shall complete LAR Attachment S, Table S-2, Implementation Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the SE.

The NRC staff, through an onsite audit or during a future fire protection inspection, may choose to examine the closure of the implementation items, with the expectation that any variations discovered during this review, or concerns with regard to adequate completion of the implementation item, would be tracked and dispositioned appropriately under the licensee's corrective action program. Any discrepancies identified during onsite audits or fire protection inspections examining dispositioning of the implementation items could be subject to appropriate NRC enforcement action as completion of the implementation items is required by the proposed license conditions.

2.7.3 Schedule

LAR (Reference 8) Section 5.5, as supplemented, provides the overall schedule for completing the NFPA 805 transition at Brunswick. The licensee stated that it will complete the implementation of the new program, including any procedure changes, process updates, and training for affected plant personnel 180 days after NRC approval of the NFPA 805 license amendment, with one exception: unless the turnover is due to fall within an outage window then the changes will be implemented 60 days after startup from the scheduled outage. In a letter dated December 18, 2014 (Reference 22), the licensee stated that Implementation Item 9 will be completed within 180 days after the startup of the second refueling outage for each unit after issuance of the SE.

LAR Section 5.5 also states that modifications will be completed by the startup of the second refueling outage (for each unit) after issuance of the SE and that appropriate compensatory measures will be maintained until modifications are complete.

3.0 TECHNICAL EVALUATION

The following sections evaluate the technical aspects of the requested license amendment to transition the FPP at Brunswick to one based on NFPA 805 (Reference 3) in accordance with 10 CFR 50.48(c). While performing the technical evaluation of the licensee's submittal, the NRC staff utilized the guidance provided in NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection" (Reference 40), to determine whether the licensee had provided sufficient information in both scope and level of detail to adequately demonstrate compliance with the requirements of NFPA 805, as well as the other associated regulations and guidance documents discussed in Section 2.0 of this SE. Specifically:

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

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

- Section 3.3 provides the results of the NRC staff review of the FM methods used by the licensee to demonstrate the ability to meet the NSPC using an FM PB approach.

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

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

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

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

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

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

3.1 NFPA 805 Fundamental FPP and Design Elements

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

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

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

- 10 CFR 50.48(c)(2)(vii) – *Performance-based methods.* While Section 3.1 of NFPA 805 prohibits the use of PB methods to demonstrate compliance with the NFPA 805, Chapter 3 requirements, 10 CFR 50.48(c)(2)(vii) states that the FPP elements and minimum design requirements of NFPA 805 Chapter 3 may be subject to the PB methods permitted elsewhere in the standard, provided a license amendment is granted and the approach 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.

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

3.1.1 Compliance with NFPA 805, Chapter 3 Requirements

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

The methods used by Brunswick for achieving compliance with the NFPA 805, Chapter 3 fundamental FPP elements and minimum design requirements are as follows:

1. The existing FPP element directly complies with the requirement: noted in LAR Attachment A, "NEI 04-02 Table B-1, Transition of Fundamental Fire Protection Program and Design Elements" (LAR Table B-1), as "Complies." (See discussion in SE Section 3.1.1.1.)

2. The existing FPP element complies through the use of an explanation or clarification: noted in LAR Table B-1 as "Complies with clarification." (See discussion in SE Section 3.1.1.2.)

3. The existing FPP element complies through the use of existing engineering equivalency evaluations (EEEEs) whose bases remain valid and are of sufficient quality: noted in LAR Table B-1 as "Complies with Use of EEEEs." (See discussion in SE Section 3.1.1.3.)

4. The existing FPP element complies with the requirement based on prior NRC approval of an alternative to the fundamental FPP attribute and the bases for the NRC approval remain valid: noted in LAR Table B-1 as "Complies by previous NRC approval." (See discussion in SE Section 3.1.1.4.)

5. The existing FPP element does not comply with the requirement, but the licensee is requesting specific approval for a PB method in accordance with 10 CFR 50.48(c)(2)(vii): noted in LAR Table B-1 as "Submit for NRC Approval." (See discussion in SE Section 3.1.1.5.)

The NRC staff has determined that, taken together, these methods compose an acceptable approach for documenting compliance with the NFPA 805, Chapter 3 requirements, because

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

The licensee stated in LAR Section 4.2.2, "Existing Engineering Equivalency Evaluation Transition," that the EEEEs that support compliance with NFPA 805, Chapter 3 or Chapter 4, were reviewed using the methodology contained in NEI 04-02 and that the review determined that the EEEEs are not based solely on quantitative risk evaluations; are appropriate use of an engineering equivalency evaluation; are of appropriate quality; meet the standard license condition; are technically adequate; reflect the plant as-built condition; and, basis for acceptability remains valid. In addition, the licensee determined that none of the transitioning EEEEs require NRC approval.

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

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

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

3.1.1.1 Compliance Strategy – Complies

For 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 information provided by the licensee in the LAR, as supplemented, and during the onsite audit (that is, the documents reviewed, discussions held with the licensee and the plant tours performed), the NRC staff concludes that the licensee's statements of compliance are acceptable.

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

- 3.3.7.1
- 3.11.5
- 3.4.1.(c)

- NFPA 805, Section 3.3.7.1, requires that storage of flammable gas shall be located outdoors, or in separate detached buildings. In a letter dated May 15, 2013 (Reference 23), the NRC staff asked fire protection engineering (FPE) RAI 09. In the RAI, the NRC staff identified that LAR Attachment A, Table B-1, Element 3.3.7.1 stated that no flammable gases are stored in safety-related buildings. However, the NRC staff found that the same entry also stated that the bulk flammable gases stored in the reactor building (RB), the diesel generator rooms (DG), and the augmented off-gas/auxiliary off-gas (AOG) building, as approved in the SER (Reference 28), are still in use. In a letter dated July 31, 2013 (Reference 12), the licensee responded to FPE RAI 09 and indicated that the LAR mistakenly stated that there were bulk flammable gases stored in the RB, DG, and AOG building and that the bulk gas storage allowed in the buildings is not flammable. The licensee revised LAR, Attachment A, which included a compliance statement of "Complies" for Element 3.3.7.1, and removed the reference to the SER. The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee confirmed that there are no flammable gases stored in the RB, DG, or AOG buildings, which meets the requirements of NFPA 805, Section 3.3.7.1.

- NFPA 805, Section 3.11.5, requires that electrical raceway fire barrier systems (ERFBS) shall be tested in accordance with, and shall meet the acceptance criteria of, NRC Generic Letter 86-10, Supplement 1. In FPE RAI 15 (Reference 23), the NRC staff requested that the licensee clarify the compliance strategy regarding ERFBS. Specifically, the NRC staff requested that the licensee provide a detailed description of what portion of the requirement is satisfied by "Complies" and what portion of the requirement is satisfied by the "Complies via EEEE." In its response to FPE RAI 15 (Reference 12), the licensee indicated that the compliance strategy "Complies via EEEE" will be deleted and the compliance strategy "Complies" will be clarified with a statement in the "Compliance Basis" column to indicate that the required ERFBS identified in LAR Attachment S, Table S-1 will be installed in accordance with the requirements specified in NFPA 805, Section 3.11.5. The licensee provided a revised LAR Attachment A, Section 3.11.5 that reflected these changes. The NRC staff concludes that the licensee's response to the RAI and statement of compliance are acceptable because the licensee indicated that required ERFBS will be installed in accordance with the requirements of NFPA 805, Section 3.11.5, and because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, in the licensee's FPP and included the action as an implementation item in LAR Attachment S, which is required by the proposed license condition.

  Also in FPE RAI 15 (Reference 23), the NRC staff requested clarification regarding the use of Pyrocrete ERFBS in the DG building emergency diesel generator Cell #1. In its response to FPE RAI 15 (Reference 12), the licensee indicated that the Pyrocrete configuration is credited in the FPRA for risk reduction purposes and that the Pyrocrete is not credited in the deterministic analysis (i.e., not in the SSA or nuclear safety capability assessment). As a

result, in a letter dated January 14, 2014 (Reference 24), in FPE RAI 15.01, the NRC staff requested that the licensee provide a more detailed description of each credited configuration, its fire resistance rating, identification of which attribute of NFPA 805, Chapter 3, was being used, and an update of compliance for each configuration in LAR Attachment C, as appropriate. In a letter dated February 28, 2014 (Reference 15), the licensee responded to FPE RAI 15.01 and identified 7 conduits and a junction or pull box containing credited circuits routed through the north fire barrier wall separating fire areas DG-4 Cell 2 (23foot level) and Fire Area DG-5 Cell 1 (23foot level) and terminating in a ceiling mounted junction/pull box in Fire Area DG-5. The licensee stated that the conduits and the junction or pull box are encased with metal lath and a 2-inch thickness of Pyrocrete. The licensee further stated that the Pyrocrete is noted in the vendor evaluation and test report as providing fire resistance to limit the backside of a protected steel plate to 200° F at 180 minutes and that the testing was conducted using the standard fire exposure curve as defined in American Society for Testing and Materials E-119. The licensee further stated that it is reasonable to expect that potential cable damage within the conduits and the junction/pull box would be limited for at least a similar time period for these configurations and that the protection provided ensures a substantial margin prior to damage, up to at least 180 minutes. The licensee further stated that the encapsulation of the credited circuits within the conduits and junction/pull box is considered an extension of the fire barrier and as such, NFPA, Chapter 3, Section 3.11.2 requirements apply. The licensee further stated that since the Pyrocrete also acts as part of various penetration seal configurations where the conduits pass through rated fire barriers, Section 3.11.4 of NFPA, Chapter 3, also applies and that is why a "Complies via EEEE" compliance statement technical basis justification for both criteria was proposed. The licensee further stated that LAR Attachment A, Table B-1, Section 3.11.2 has been revised to add the evaluations that provide the technical justifications described above as reference documents.

In addition, as part of the response to FPE RAI 15.01 (Reference 15), the licensee revised LAR Attachment C, Table B-3 to include discussion of a new "Type 0" VFDR associated with the described credited circuits (i.e., Pyrocrete embedded) and that this feature is specifically described in the revised LAR Attachment C, Table B-3 for Fire Area DG-5 Diesel Cell 1 Overview.

The NRC staff concludes that the licensee's responses to the RAIs and the statements of compliance are acceptable because the licensee demonstrated that the design and qualification of the Pyrocrete installations meets the requirements of NFPA 805, Chapter 3.

- NFPA 805, Section 3.4.1(c), requires that during every shift, the brigade leader and at least two brigade members shall have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance criteria. An exception in the standard states that sufficient training and knowledge shall be permitted to be provided by an

operations advisor dedicated to industrial fire brigade support. In FPE RAI 21 (Reference 23), the NRC staff requested additional clarification on the use of the term "Fire Brigade Operations Advisor" as indicated in the compliance strategy to comply with NFPA 805, Section 3.4.1(c). Specifically, the NRC requested that the licensee describe the duties of the Fire Brigade Operations Advisor. In its response to FPE RAI 21 (Reference 12), the licensee indicated that a change to the compliance strategy was made and that they are no longer using a Fire Brigade Operations Advisor. The licensee further stated that they will utilize a fire brigade where, during every shift, the brigade leader and at least two brigade members shall have sufficient training and knowledge of nuclear safety systems to understand the effects of fire and fire suppressants on nuclear safety performance.

In FPE RAI 21.01 (Reference 24), the NRC staff requested that the licensee describe how it is assured that the brigade leader and additional members will possess the necessary training and knowledge to comply with the requirements of NFPA 805, Section 3.4.1(c). In its response to FPE RAI 21.01 (Reference 15), the licensee stated that an equivalent knowledge of plant systems is provided for under its fleet fire brigade training procedure (FIR-NGGC-0007, NFPA 805 Fire Brigade Training Program), that specifies the plant systems, for either a pressurized-water reactor (PWR) or boiling-water reactor (BWR), that represent the minimum plant knowledge for a Non-Licensed Operator (NLO) fire brigade member or leader to understand the effects of fire and fire suppressants on nuclear safety performance criteria equipment. The NRC staff determined that the licensee's fleet procedural requirements for a fire brigade leader and at least two other members of the brigade to meet training and knowledge requirements, may be satisfied by completing and maintaining a fire brigade qualification, and either being a licensed Senior Reactor Operator, being a licensed Reactor Operator, successfully completing a reactor operator certification program, or successfully completing training on the following plant systems:

- Reactor Coolant System
- High Pressure Core Injection
- Reactor Core Isolation Cooling
- Reactor Auxiliary Cooling System
- Station Auxiliary Cooling System
- Residual Heat Removal System
- Core Spray System
- Emergency Service Water System

- Filtration/recirculation Ventilation System
- Electrical System Overview – Alternating Current & Direct Current
- Emergency Core Cooling System

The NRC staff concludes that the licensee's responses to the RAIs and the statement of compliance are acceptable because the licensee's procedure requires that the brigade leader and at least two brigade members have sufficient training and knowledge of nuclear safety systems, and the licensee's procedures ensure the requirements of NFPA 805, Section 3.4.1(c) will be met.

3.1.1.2 Compliance Strategy – Complies with Clarification

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

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

- 3.2.3(1)
- 3.3.1.2(5)
- 3.3.1.3.1
- 3.3.5.3
- 3.3.9
- 3.6.2
- 3.10.7

- NFPA 805, Section 3.2.3(1), requires that procedures be established for inspection, testing, and maintenance for fire protection systems and features credited by the fire protection program. The licensee indicated that surveillance frequencies are outlined in plant procedures and may be modified in accordance with the methodology in EPRI Report TR1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide" (Reference 73). The licensee submitted its compliance strategy for this requirement as a 10 CFR 50.48(c)(2)(vii) approval request in LAR Attachment L. See SE Section 3.1.4.2 for the NRC staffs evaluation of this request.

- NFPA 805, Section 3.3.1.2(5), requires that controls on use and storage of flammable and combustible liquids shall be in accordance with NFPA 30, "Flammable and Combustible Liquids Code" (Reference 74). The licensee stated that station procedures for storage and use of flammable and combustible liquids use NFPA 30 as a developmental reference and that a code compliance evaluation demonstrates compliance with NFPA 30, which establishes a point-by-point evaluation of compliance with NFPA 30. The licensee identified the need to provide modifications to existing diesel fuel oil tanks to achieve compliance with NFPA 30 regarding tank venting and the protection of tank supports as identified in LAR Attachment S, Table S-1, Modification 8.

The NRC staff concludes that the licensee's statement of compliance is acceptable because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, and included the action as a modification in LAR Attachment S, which is required by the proposed license condition.

- NFPA 805, Section 3.3.1.3.1, requires that a hot work safety procedure be developed, implemented, and periodically updated, as necessary, in accordance with NFPA 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work" (Reference 75) and NFPA 241, "Standard for Safeguarding Construction, Alteration, and Demolition Operations" (Reference 76). The licensee stated that for hot work safety procedures, compliance with NFPA 241 is addressed through compliance with NFPA 51B and that NFPA 241, 2009 edition, as referenced by NFPA 805, Section 5.1.1, with respect to hot work, states that the responsibility for hot work operations and fire prevention precautions, including permits and fire watches, and shall be in accordance with NFPA 51B, "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work."

  In FPE RAI 20 (Reference 23), the NRC staff stated that the station hot work permit procedure indicates that 'roving hot work fire watches" are used during operating Modes 4 and 5 and that the roving fire watch is allowed to monitor multiple hot work jobs in relatively close proximity to each other, and that the procedure indicates that using a video camera and monitor is acceptable for viewing hot work activities. The NRC staff further stated that neither of these practices are recognized by NFPA 51B and requested that the licensee provide the bases for why it considers these practices acceptable for compliance with NFPA 805, Section 3.3.1.3.1. In its response to FPE RAI 20 (Reference 12), the licensee stated that the NFPA standard is silent with regard to the use of a single fire watch for multiple hot work jobs and that because their administrative controls ensure that hot work jobs are within close proximity of each other and all provisions of the fire watch responsibilities must be controlled in their entirety at all times for all areas covered, that the fire watch requirements of NFPA 51B are met.

  Specifically, the licensee stated that functional equivalency for the use of a single, non-roving fire watch to monitor multiple hot work activities is provided through the following limitations and controls:

  - The hot work activities are occurring in the same hot work area within a line of sight of each other and within an approximate 35-foot radius;

  - All hot work activities being monitored by the fire watch within the hot work zone must stop if any adverse conditions occur within the hot work area until the adverse conditions are remedied;

  - The hot work fire watch will remain until 30 minutes after the last activity is completed;

- The use of a single fire watch for multiple areas does not apply where the potential exists for slag or sparks to escape the specific elevation and area where the hot work is being completed or where additional fire watches may be necessary to observe the entire hot work area;

- A hot work permit will be required for each activity and the 35-foot radius around each activity will be inspected per the requirements stated above.

LAR Attachment S, Table S-2, Implementation Item 10 has been added to revise the procedure to employ these limitations and conditions on the use of a single fire watch for multiple hot work activities and the NRC staff considers this acceptable because the action is included in LAR Attachment S which is required by the proposed license condition.

In FPE RAI 20.01 (Reference 24) the NRC staff requested that the licensee provide additional information regarding the use of video cameras in lieu of a fire watch as functionally equivalent. In its response to FPE RAI 20.01 (Reference 21), the licensee provided additional clarification regarding the type, location of hot work activities, conditions, and limitations under which video-type cameras are allowed to be used to supplement the hot work fire watch. The licensee also provided the justification or controls in place to achieve equivalency with the fire watch function which include:

- Process is controlled by hot work plant procedure;

- The need for supplemental cameras is typically limited to those plant areas where radiological conditions create a condition where dose is a major factor (i.e., Drywell, Reactor Water Cleanup rooms, Residual Heat Removal Area);

- The hot work job is pre-planned with camera equipment pre-staged to ensure the equipment is readily available and easy to use;

- When a hot work activity is due to start, the fire watch, and/or welding lead conduct a walk down of the work area to ensure that the work area conditions are appropriate for the capability of the cameras;

- Procedural checklists are completed to document the conditions and approval process. This also includes assurance that the work area and travel path are clearly understood by the relevant personnel, including easy and fast access to the work location prior to the start of work;

- The camera(s) is (are) then installed in the work area, and the coverage area is verified to be satisfactory by the fire watch;

- Camera angles are adjusted to ensure both the safety of the welder and the line of sight to cover the 35-foot area surrounding the weld location. If areas are blocked from view, when using a single camera, more than one camera may be utilized to cover the weld activity;

- A fire extinguisher is staged at the job site so that the fire watch can respond immediately to the work area in the event of a fire or other incident;

- During the ongoing work evolution, the travel path/work area conditions are verified prior to each entry;

- The fire watch establishes communication with personnel performing the hot work activity;

- The fire watch is staged at the step-off pad (i.e., a low dose standby area), and is dressed in protective clothing for immediate response. When using cameras, there is one fire watch for each weld activity;

- If conditions change such that the fire watch cannot effectively observe the work or perform the required response, the fire watch relocates to the work location; and

- If the video camera or monitor becomes inoperable, the hot work is stopped by the fire watch and a fire watch is posted in the area of the hot work until the video camera or monitor is returned to service.

The NRC staff concludes that the licensee's responses to the RAIs and statement of compliance are acceptable because the licensee demonstrated a compliance strategy regarding hot work fire watches that will meet the requirements of NFPA 805, Section 3.3.1.3.1, and also identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, in the licensee's FPP and included the action as an implementation item in LAR Attachment S, which is required by the proposed license condition.

- NFPA 805, Section 3.3.5.3, requires that electric cable construction comply with a flame propagation test as acceptable to the AHJ. The licensee stated that FAQ 06-0022 (Reference 60) provides an appendix to evaluate currently recognized flame propagation tests to the Institute of Electrical and Electronics Engineers (IEEE) 383-1974 Standard, the NRC minimum test standard, and acceptance criteria for cable flame propagation tests. The licensee further stated

that the specifications applicable for the procurement of various cables are listed in the design basis document for cables and raceways, and that the cables procured during and following construction were qualified as being self-extinguishing and non-propagating, and they meet or exceed the IEEE 383 flame test.

In FPE RAI 07 (Reference 23), the NRC staff requested clarification on the statement of compliance for LAR Attachment A, Table B-1 Section 3.3.5.3. In its response to FPE RAI 07 (Reference 12), the licensee stated that "Complies via EEEE" was not necessary and also provided a revised LAR Attachment A, Table B-1, that removed the "Complies via EEEE" compliance strategy.

The NRC staff concludes that the licensee's response to the RAI and statement of compliance are acceptable because the licensee demonstrated that electric cable construction meets or exceeds IEEE 383, which complies with NFPA 805, Section 3.3.5.3.

- NFPA 805, Section 3.3.9, requires that transformer oil collection basins and drain paths be periodically inspected to ensure that they are free of debris and capable of performing their design function. In FPE RAI 10 (Reference 23), the NRC staff identified that LAR Attachment A, Table B-1, Section 3.3.9 had been omitted and requested that the licensee correct the discrepancy. In its response to FPE RAI 10 (Reference 12), the licensee provided a revised LAR Attachment A, Table B-1, that included Section 3.3.9, "Transformers" as "Complies with Clarification." The licensee indicated that compliance with this element requires completion of Modification 2, as described in LAR Attachment S, Table S-1.

  The NRC staff concludes that the licensee's response to the RAI and statement of compliance are acceptable because the licensee provided a revised LAR Attachment A that included the missing information, and because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, and included the action as a modification in LAR Attachment S, which is required by the proposed license condition.

- NFPA 805, Section 3.6.2, requires that adequate water flow rate and nozzle pressure be provided for all hose stations and includes the provision for hose station pressure reducers where necessary for the safety of fire brigade members and off-site fire department personnel. The licensee stated that they would add pressure reduction devices to applicable fire hose stations based on a code compliance calculation for NFPA 14 (Reference 58) and identified this action as Modification 4, as described in LAR Attachment S, Table S-1.

  The NRC staff concludes that the licensee's statement of compliance is acceptable because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, and included the action as a modification in LAR Attachment S, which is required by the proposed license condition.

- NFPA 805, Section 3.10.7, requires that the carbon dioxide system be provided with an odorizer. The licensee stated that compliance with this requirement requires completion of Modification 3, as described in LAR Attachment S, Table S-1.

  The NRC staff concludes that the licensee's statement of compliance is acceptable because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, and included the action as a modification in LAR Attachment S, which is required by the proposed license condition.

3.1.1.3 Compliance Strategy – Complies with Use of EEEEs

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

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

- 3.3.2 • 3.11.4

- NFPA 805, Section 3.3.2, requires that walls, floors, and components required to maintain structural integrity be of noncombustible construction. In FPE RAI 05 (Reference 23), the NRC staff requested that the two compliance strategies "Complies" and "Complies via EEEE" regarding the structural steel fireproofing be clarified. Specifically, the NRC staff requested that the licensee provide a description of what portion of this requirement "Complies via EEEE." In its response to FPE RAI 05 (Reference 12), the licensee identified that the referenced EEEE applies only with regard to Fire Area CB-23E and that all other fire areas fall under the "Complies" compliance strategy.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated which compliance strategy applied to different fire areas which demonstrated compliance via EEEE with NFPA 805, Section 3.3.2.

- NFPA 805, Section 3.11.4, requires that through penetration fire stops for penetrations such as pipes, conduits, bus ducts, cables, wires, pneumatic tubes and ducts, and similar building service equipment that pass through fire barriers, be protected. In FPE RAI 14 (Reference 23), the NRC staff requested clarification regarding compliance strategies in LAR Attachment A, Table B-1, Section 3.11.4. In its response to FPE RAI 14 (Reference 12), the licensee indicated that the "Complies via EEEE" compliance strategy refers to the referenced list of evaluations where the various seal configurations were evaluated to be acceptable for the hazards in which they are installed and that

these various identified penetration seal locations are evaluated in each of the referenced EEEE's. The licensee provided a revision to LAR Attachment A, Table B-1, Section 3.11.4 and also a new table, "Table from Section 3.11.4," to LAR Attachment A, to clarify the scope of "Complies via EEEEs."

The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance with NFPA 805, Section 3.11.4, via EEEE for various penetration seal configurations.

3.1.1.4 Compliance Strategy – Complies with Previous NRC Approval

Certain NFPA 805, Chapter 3 requirements were supplanted by an alternative that was previously approved by the NRC. The approvals in the current licensing basis described in LAR Section 2.1 were documented in the original SE Report, dated November 22, 1977 (Reference 28), as supplemented by SEs dated April 6, 1979 (Reference 71); June 11, 1980 (Reference 30); December 30, 1986 (Reference 31); December 6, 1989 (Reference 32); July 28, 1993 (Reference 33); and February 10, 1994 (Reference 34).

The licensee identified that the following Chapter 3 elements required clarification:

- 3.3.5.2 • 3.5.5

- NFPA 805, Section 3.3.5.2, requires that only metal tray and metal conduits be used for electrical raceways. In FPE RAI 06 (Reference 23), the NRC staff requested additional information regarding the compliance strategy "Complies via Previous NRC Approval" as described in LAR Attachment A, Table B-1, Section 3.3.5.2, which identifies the requirement that only metal tray and metal conduits be used for electrical raceways. In its response to FPE RAI 06 (Reference 12), the licensee stated that the compliance strategy "Complies via Previous NRC Approval" applies only to the following Control Building cable access way fire zones: CB-01A, CB-01B, CB-02A, CB-02B, CB-12A, CB-12B, CB-13A and CB-13B for existing electrical raceway construction details which are located in Fire Areas CB-1 and CB-2. The licensee further stated that the "Complies with Clarification" compliance statement applies to all other plant fire areas and zones relative to cable installation specification requirements and the guidance of FAQ 06-0021, "Cable Air Drops," (Reference 77). The licensee submitted a revision to LAR Attachment A, Table B-1, Section 3.3.5.2 to include the clarification.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance via previous approval with NFPA 805, Section 3.3.5.2, by describing the previously approved configuration, by describing the areas that were previously approved, and by describing what areas complied with clarification.

- NFPA 805, Section 3.5.5, requires that each pump and its driver and controls be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers. In FPE RAI 11 (Reference 23), the NRC staff identified that LAR Attachment A, Table B-1, Section 3.5.5 indicates "complies" with "no additional clarification." The NRC staff requested the licensee describe in greater detail, compliance with fire pump separation from other fire pumps and from the remainder of the plant by rated fire barriers. In its response to FPE RAI 11 (Reference 12), the licensee described the previous NRC approval of the separation and submitted a revised LAR, Attachment A, Table B-1, Section 3.5.5, to include the new compliance statement of "Complies via Previous NRC Approval."

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance via previous approval with NFPA 805, Section 3.5.5, by describing the previously approved configuration and discussing the previous approval justification.

In both cases, the licensee evaluated the basis for the original NRC approval and determined that the bases are still valid. The NRC staff reviewed the information provided by the licensee and concludes that previous NRC approval has been demonstrated using suitable documentation that meets the approved guidance contained in RG 1.205, Revision 1. Based on the licensee's justification for the continued validity of the previously approved alternatives to the NFPA 805, Chapter 3 requirements, the NRC staff concludes that the licensee's statements of compliance, as supplemented, are acceptable.

3.1.1.5 Compliance Strategy – Submit for NRC Approval

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

- NFPA 805, Section 3.5.16, which concerns the use of fire protection water for specific non-fire plant evolutions. Contrary to the requirements of NFPA Section 3.5.16, fire protection water is used in certain non-fire support applications. NRC approval is requested for the temporary use of the fire protection water supply, with restrictions. See SE Section 3.1.4.1 for the NRC staff's review of this request.

- NFPA 805, Section 3.2.3(1), which concerns the use of performance-based methods to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805. In FPE RAI 16 (Reference 23), the NRC staff identified the licensee's intent to use the performance-based frequencies from EPRI Technical Report TR-1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide"

(Reference 73). In its response to FPE RAI 16 (Reference 12), the licensee requested the ability to utilize performance-based methods to establish the appropriate inspection, testing, and maintenance frequencies for fire protection systems and features required by NFPA 805. The licensee stated that the performance-based inspection, testing, and maintenance frequencies will be established as described in the EPRI technical report. See SE Section 3.1.4.2 for the NRC staff's review of this request.

As discussed in SE Section 3.1.4, the NRC staff concluded that the use of performance-based methods to demonstrate compliance with these fundamental fire protection program elements is acceptable.

3.1.1.6 Compliance Strategy – Multiple Strategies

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

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

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

- 3.3(2)
- 3.3.2
- 3.3.5.2
- 3.3.5.3

- NFPA 805, Section 3.3(2), requires that a fire prevention program with the goal of preventing a fire from starting be established, documented, and implemented as part of the fire protection program including design controls that restrict the use of combustible materials. In FPE RAI 04 (Reference 23), the NRC staff requested clarification regarding the compliance strategy in LAR Attachment A, Table B-1, Section 3.3(2) for design controls that are used to restrict combustibles. The NRC staff found that the LAR indicates two compliance strategies: "Complies" and "Complies via EEEE." Specifically the NRC staff requested the licensee provide a description of what portion of this requirement "Complies via EEEE." In its response to FPE RAI 04 (Reference 12), the licensee stated that the evaluation was for the acceptability to abandon in place the Thermo-Lag fire barrier materials installed within the identified 20-foot separation zones of Fire Area RB2-1 located in the Unit 2 Reactor Building and as such, the "Complies via EEEE" compliance strategy applies only to Fire Area RB2-1 and that all other Fire Areas in Section 3.3(2) fall under the "Complies" compliance strategy.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance or compliance via EEEE for NFPA 805, Section 3.3(2), for the fire areas discussed above.

- NFPA 805, Section 3.3.2, requires that walls, floors, and components required to maintain structural integrity be of noncombustible construction. In FPE RAI 05 (Reference 23) the NRC staff requested clarification regarding the compliance strategy in LAR Attachment A, Table B-1, Section 3.3.2, which indicates two compliance strategies: "Complies" and "Complies via EEEE." Specifically, the NRC staff requested the licensee provide a description of what portion of this requirement "Complies via EEEE." In its response to FPE RAI 05 (Reference 12), the licensee indicated that the referenced evaluation for structural steel fireproofing evaluated the acceptability of not fire proofing exposed structural steel located in the Control Building elevator shaft and that this evaluation is limited to fire zones CB-6, CB-11, and CB-22, which are located within fire area CB-23E and as such, the "Complies via EEEE" compliance strategy applies only to Fire Area CB-23E and all other fire areas fall under the "Complies" compliance strategy. The licensee submitted a revision to Attachment A, Table B-1, Section 3.3.2 to reflect this change.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance or compliance via EEEE for NFPA 805, Section 3.3.2, for the fire areas discussed above.

- NFPA 805, Section 3.3.5.2, requires that only metal tray and metal conduits be used for electrical raceways. In FPE RAI 06 (Reference 23), the NRC staff requested clarification regarding the compliance strategy in LAR Attachment A, Table B-1, Section 3.3.5.2, which indicates "Complies via previous NRC approval." The NRC staff however, determined that the section of the 1977 SER (5.1) (Reference 28) cited in the LAR addressed only cable access ways in the control building for safety-related equipment, and therefore, this previous approval does not encompass the extent of the NFPA 805 requirement for all tray and conduit electrical raceways. The licensee was requested to provide additional detail sufficient to allow "'previous NRC approval" or to submit an alternative compliance strategy. In its response to FPE RAI 06 (Reference 12), the licensee stated that the compliance strategy "Complies via Previous NRC Approval" applies only to Control Building fire zones CB-01A, CB-01B, CB-02A, CB-02B, CB-12A, CB-12B, CB-13A and CB-13B for existing electrical raceway construction details, which are located in Fire Areas CB-1 and CB-2. The licensee further stated that the "Complies with Clarification" compliance statement applies to all other plant fire areas and zones relative to raceway installation specification requirements and the guidance in FAQ 06-0021 (Reference 77). The licensee provided a revision to LAR Attachment A, Table B-1, Section 3.3.5.2, to include this clarification.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance via previous NRC approval or compliance with clarification for NFPA 805, Section 3.3.5.2 for the fire areas discussed above.

- NFPA 805, Section 3.3.5.3, requires that electric cable construction comply with a flame propagation test as acceptable to the AHJ. In FPE RAI 07 (Reference 23), the NRC staff requested clarification regarding the compliance strategy in LAR Attachment A, Table B-1, Section 3.3.5.3. Specifically, the licensee was requested to provide a specific description of what portion of this requirement is satisfied by the EEEE. In its response to FPE RAI 07 (Reference 12), the licensee indicated that the compliance strategy "Complies via EEEE" is not necessary. A revision was made to LAR Attachment A, Table B-1, Section 3.3.5.3, to include this clarification.

  The NRC staff concludes that the licensee's response to the RAI and the statement of compliance are acceptable because the licensee demonstrated compliance via previous NRC approval or compliance with clarification for NFPA 805, Section 3.3.5.3, for electric cable construction.

3.1.1.7 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, Section 3.4.5 and Section 3.11).

- Sections that are not applicable because of the following:

  - The licensee stated that they do not have systems of this type installed (e.g., Section 3.6.5, which applies to seismic hose station cross-connected to non-fire protection systems and Section 3.10.4, which has single failure limitations for gaseous fire suppression systems in areas that are required by both primary and backup systems); and

  - The requirements are structured with an applicability statement (e.g., Section 3.3.12, which applies to reactor coolant pumps or Sections 3.4.1(a)(2) and 3.4.1(a)(3), which apply to the fire brigade standards used since they depend on the type of brigade specified in the fire protection program).

3.1.1.8 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:

- 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 the state of compliance with the NFPA 805, requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:

  - With the requirement directly or with the requirement directly after the completion of an implementation item;

  - With the intent of the requirement (or element) and provided adequate justification;

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

  - Through the use of EEEEs;

  - Through the use of a combination of the above methods; and

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

3.1.2 Identification of the Power Block

The NRC staff reviewed the structures identified in LAR Attachment I, Table I-1 "Definition of Power Block" as comprising the "power block." The plant structures listed are established as part of the power block for the purpose of denoting the structures and equipment included in the RI/PB fire protection program 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 that could affect plant operation for power generation; equipment important to safety; equipment that could affect the ability to maintain NSCA in the event of a fire; or structures containing radioactive materials that could potentially be released in the event of a fire. The NRC staff concludes that the licensee evaluated the structures and equipment to adequately document the list of those structures that fall under the definition of "power block" in NFPA 805.

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

GL 2006-03 requested that licensees evaluate their facilities to confirm compliance with existing applicable regulatory requirements in light of the results of NRC testing that determined that both Hemyc and MT fire barriers failed to provide the protective function intended for compliance with existing regulations, for the configurations tested using the NRC's thermal acceptance criteria. In a letter dated June 6, 2006 (Reference 78), the licensee stated that Hemyc™ or MT™ fire barrier material has not been installed. Since Hemyc or MT electrical raceway fire barrier systems (ERFBS) are not used, the NRC staff concludes that the generic issue, GL 2006-03 (Reference 55) related to the use of these ERFBS is not applicable.

3.1.4 Performance Based Methods for NFPA 805, Chapter 3 Elements

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

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

(B) Maintains safety margins; and

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

In LAR Attachment L, "NFPA 805 Chapter 3 Requirements for Approval (10 CFR 50.48(c)(2)(vii))," the licensee requested NRC staff review and approval of 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 evaluation of these proposed methods is provided below.

3.1.4.1 NFPA 805, Section 3.5.16, Dedicated Fire Protection Water Supply

In LAR Attachment L, Approval Request 1, the licensee requested review and approval because it uses fire protection water in several plant support applications. NFPA 805, Section 3.5.16, states that the fire protection water supply system shall be dedicated for fire protection use only. Contrary to the requirements of NFPA, Section 3.5.16, the licensee indicated that fire protection water is utilized in the following plant support applications:

- Containment Heat Removal. In the event that nuclear service water is lost to the residual heat removal (RHR) heat exchangers, the water-based fire suppression system may be used to provide backup cooling for containment heat removal;

- Coolant Injection. In the event of a failure of the normal reactor level control systems to maintain water level, the water-based fire suppression system may be used as an alternate coolant injection system. In addition, fire water may be used for alternate boron injection;

- Fuel Pool Cooling. Fire hoses on the Reactor Building 117' elevation may be used as a makeup water source, if the spent fuel pool level cannot by recovered by normal means;

- RHR Service Water Shutdown and wet layup process;

- Flushing/filling RHR service water and heat exchangers;

- RHR service water system operability test;

- Flushing radwaste radiation monitor;
- Seal water to storm drain collector basin pumps;
- Temporary cooling water supply to the service air compressor; and
- Refill of standby gas treatment drain trough.

In FPE RAI 18 (Reference 23), the NRC staff requested more detail regarding the use of the fire protection water supply system for non-fire purposes. In its response to FPE RAI 18 (Reference 12), the licensee provided, for each non-fire protection use listed above, a description of the operation being performed, a reference to the procedure being used, the operator response in the event of a fire during that operation, and the impact of use concurrent with fire suppression demand. For each of the uses listed above, the licensee stated that based on flow rates and volumes used from the fire protection water supply, the margin available in the fire protection water supply system is adequate. The licensee further stated that the largest fire water demand for a safety-related area is the Unit 2 South RHR area, and the largest fire water demand for a non-safety related area is the main transformer. Both were calculated to be within the capacity of a single fire pump, and these flow demands are discussed in the water-based fire suppression system design basis document.

The licensee stated that each normal evolution is performed under procedural controls with annunciators for low tank level or local monitoring to alert the operator so that the minimum tank water level will not be violated. The licensee further stated that alerts are provided by tank low and low-low alarms, along with fire pump running annunciators in the main control room, that level switches on the tank control automatic makeup, and an annunciator alert is provided in the main control room if the tank level is drawn down. The licensee further stated that routine surveillance checks by plant operators using a local tank level indicator verify that the tank level is kept above the minimum level, that a manual bypass valve may also be used by the operator to refill the tank, and that fill water is supplied from a 15,000 gallon onsite county water storage tank.

The licensee stated that the use of fire protection water for containment heat removal, coolant injection, alternate boron injection, and fuel pool cooling is strictly controlled by emergency operating and abnormal operating procedures, and that plant operators are trained regularly on these procedures and the equipment. The licensee further stated that communications are handled by the plant public address system, sound powered phones, and operations radio systems.

The licensee stated that the use of the fire protection water for plant evolutions other than fire protection is an infrequent, abnormal or emergency operational occurrence requiring the control room supervisor direction and concurrence. The licensee further stated that the ability to isolate the non-fire protection flows ensures there is no impact on manual fire suppression efforts, and therefore, there is no impact on the nuclear safety performance criteria. The licensee further stated that the use of the fire protection water for plant evolutions other than fire protection involves fire protection water flow into existing plant systems and that leakage from these

systems is not part of the fire protection system operation or firefighting evolutions and, as such, has no impact on the radiological release-performance criteria.

The licensee stated that because both the automatic and manual fire suppression functions are maintained, any reliance on these systems for DID is maintained and that the methods, input parameters, and acceptance criteria used in the analysis were reviewed against those used for NFPA 805, Chapter 3 acceptance. The licensee further stated that the methods, input parameters, and acceptance criteria used to calculate flow requirements for the automatic and manual suppressions systems were not altered, and therefore, the safety margin inherent in the analysis for fire events has been preserved.

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

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

As discussed in SE Section 3.1.1.2, the NRC staff requested information regarding the compliance strategy for NFPA 805, Section 3.2.3(1). In FPE RAI 16 (Reference 23), the NRC staff identified, in LAR Attachment A, Element 3.2.3 and in LAR Attachment S, Implementation Item 5, the licensee's intent to use the performance-based frequencies for inspection, testing, and maintenance procedures, as described in EPRI Technical Report (TR) 1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide" (Reference 73). The adoption of the EPRI method as a performance-based alternative to the deterministic Chapter 3 element requires approval in accordance with 10 CFR 50.48(c)(2)(vii). Specifically, the NRC staff requested that the licensee address whether EPRI TR-1006756 is intended as an alternative, and, if so, provide the appropriate supporting information consistent with Section 50.48(c)(2)(vii). In its response to FPE RAI 16 (Reference 12), the licensee submitted an approval request in accordance with 10 CFR 50.48(c)(2)(vii).

The licensee stated that there will be no impact on the NFPA 805 nuclear safety performance criteria because the use of performance-based test frequencies established using TR-1006756 methods, combined with the requirements in NFPA 805, Section 2.6, "Monitoring Program," will provide assurance that the availability and reliability of the fire protection systems and features are maintained to the levels assumed in the NFPA 805 engineering analyses. The licensee further stated that the radiological release performance criteria are satisfied based on the determination of limiting radioactive release as discussed in LAR Attachment E. The licensee further stated that some fire protection systems and features are credited as part of the evaluation and the use of performance-based test frequencies established using TR-1006756 methods, with the new monitoring program, should ensure that the availability and reliability of the systems and features are maintained to the levels assumed in the analyses credited for meeting the radioactive release performance criteria.

The licensee further stated that the proposed alternative maintains the safety margins of the analyses because it provides assurance that the availability and reliability of the systems and features are maintained to the levels assumed in the original NFPA 805 engineering analyses, which includes those assumptions credited in the risk evaluation safety margin discussions and that the use of these methods should in no way invalidate the inherent safety margins contained in the codes used for design and maintenance of fire protection systems and features, and therefore, the safety margin inherent and credited in the analyses should be preserved.

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

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

3.2 Nuclear Safety Capability Assessment (NSCA) 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 requirements of 10 CFR 50.48(c).

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

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

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

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

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

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

> In the event of a fire during any operational mode and plant configuration, the plant shall be as follows:
>
> (1) *Reactivity Control.* Capable of rapidly achieving and maintaining subcritical conditions.
>
> (2) *Fuel Cooling.* Capable of achieving and maintaining decay heat removal and inventory control functions.
>
> (3) *Fission Product Boundary.* Capable of preventing fuel clad damage so that the primary containment boundary is not challenged.

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

> Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met:
>
> (a) *Reactivity Control.* Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.
>
> (b) *Inventory and Pressure Control.* With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained for a PWR and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a BWR such that fuel clad damage as a result of a fire is prevented.
>
> (c) *Decay Heat Removal.* Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.
>
> (d) *Vital Auxiliaries.* Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.

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

3.2.1 Compliance with NFPA 805 NSCA Methods

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

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

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

RG 1.205, Revision 1 (Reference 4) endorses NEI 04-02, Revision 2 (Reference 7), and Chapter 3 of NEI 00-01, Revision 2 (Reference 35), and promulgates the method outlined in NEI 04-02 for conducting a nuclear safety capability assessment. This NRC-endorsed guidance (i.e., NEI 04-02 Table B-2, "NFPA 805 Chapter 2 – Nuclear Safety Transition – Methodology Review" and NEI 00-01, Chapter 3) has been determined to address the related requirements of NFPA 805, Section 2.4.2. The NRC staff reviewed LAR Section 4.2.1, "Nuclear Safety Capability Assessment Methodology," and Attachment B, "NEI 04-02 Table B-2 – Nuclear Safety Capability Assessment – Methodology Review," against these guidelines.

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

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

The licensee developed the LAR based on the three guidance documents cited above. Although RG 1.205, Revision 1, endorses NEI 00-01, Revision 2, the licensee's review was based on the guidance in NEI 00-01, Revision 1. In addition, a review of NEI 00-01, Revision 2, Chapter 3, was conducted by the licensee to identify the substantive changes from NEI 00-01, Revision 1, that are applicable to the fire protection program. The NRC staff concludes that based on the information provided in the licensee's submittal, as supplemented, a systematic process to evaluate the post-fire SSA against the requirements of NFPA 805, Section 2.4.2; and the licensee used subsections (1), (2), and (3), which meets the methodology outlined in the latest NRC-endorsed industry guidance.

FAQ 07-0039, "Incorporation of Pilot Plant Lessons Learned – Table B-2" (Reference 63), provides one acceptable method for documenting the comparison of the safe shutdown analysis 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.

The licensee performed this evaluation by comparing its SSA against the NFPA 805 NSCA requirements using the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 1, and documenting the results of the review in LAR Attachment B, Table B-2, "NEI 04-02 Table B-2 Nuclear Safety Capability Assessment Methodology Review," in accordance with the guidance of NEI 04-02, Revision 2.

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

(1) The SSA directly aligns with the attribute: noted in the LAR Table B-2 as "Aligns." (See discussion in SE Section 3.2.1.1.)

(2) The SSA aligns with the intent of the attribute: noted in the LAR Table B-2 as "Aligns with intent." (See discussion in SE Section 3.2.1.2.)

(3) The SSA does not align with the attribute, but there is a prior NRC approval of an alternative to the attribute, and the bases for the NRC approval remains valid: noted in LAR Table B-2 as "Not in Alignment, but Prior NRC Approval." (See discussion in SE Section 3.2.1.3.)

(4) The SSA does not align with the attribute, but there are no adverse consequences because of the non-alignment: noted in LAR Table B-2 as "Not in Alignment, but No Adverse Consequences." (See discussion in SE Section 3.2.1.4.)

(5) The SSA does not align with the attribute: noted in LAR Table B-2 as "Not in Alignment." (See discussion in SE Section 3.2.1.5.)

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

As stated above, the licensee performed the review of the NSCA based on the guidance contained in NEI 00-01, Revision 1, instead of Revision 2. The licensee stated that an additional review was performed of NEI 00-01, Revision 2, Chapter 3 for specific substantive changes in the guidance from NEI 00-01, Revision 1, that are applicable to an NFPA 805 transition. The results of this review are summarized below:

- Post fire manual operation of rising stem valves in the fire area of concern (NEI 00-01 Section 3.2.1.2). The licensee conducted a review of the NSCA results and indicated that there are no recovery actions or DID recovery actions that require manual operation of a rising stem valve in the fire area of concern.

- Analysis of open circuits on high voltage (e.g., 4.16 kV) ammeter current transformers (NEI 00-01 Section 3.5.2.1). The licensee conducted an evaluation and concluded that this failure mode is unlikely for current transformers that could pose a threat to safe shutdown equipment.

- Analysis of control power for switchgear with respect to breaker coordination (NEI 00-01 Section 3.5.2.4). The licensee stated that control power is modeled in the SSD fault tree used to develop the NSCA and that a loss of control power results in an assumed loss of the switchgear, and there are no cases where a bus is credited to remain operable without control power.

3.2.1.1 Attribute Alignment – Aligns

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 information provided by the licensee in the LAR, as supplemented, and the information provided during the NFPA 805 site audit (that is, the documents reviewed, discussions held with the licensee and the plant tours performed), the NRC staff concludes that the licensee's statements of alignment are acceptable because the analyses are consistent with regulatory guidance for selecting the systems and equipment and their interrelationships necessary to achieve the NSPC, selection of the cables necessary to achieve the NSPC, and the identification of the location of nuclear safety equipment and cables.

3.2.1.2 Attribute Alignment – Aligns with Intent

In several of the NEI 00-01, Chapter 3, attributes, the licensee determined that the post-fire SSA aligns with the intent of the attribute, and provided additional clarification when describing its means of alignment. The attributes identified in LAR Attachment B, Table B-2, as having this condition are as follows:

- 3.1.1.9 72 Hour Coping;

- 3.1.3.4 Assign Shutdown Paths to Each Combination of Systems; and

- 3.3.3.2 Identify Interlocked Circuits and Cables Whose Spurious Operation or Mal-operation Could Affect Shutdown

The NEI 00-01 attributes 3.1.1.9, 3.1.3.4, and 3.3.3.2 for which the licensee stated aligns with intent listed above, describe similar means or methods that were applied to achieve the intended result of the NEI 00-01 guidance. The NRC staff concludes that the methods, as described by the licensee, are acceptable because they are similar to the specific methods in NEI 00-01, and therefore, align with the intent of NRC endorsed guidance.

LAR Attachment B, Table B-2, Section 3.1.1.9, "72-Hour Coping," indicates that the alternate shutdown methodology ensures cold shutdown can be achieved in 72 hours, including repairs. However, the NRC staff found that the cold shutdown actions including repairs are not identified as variances from deterministic requirements. In SSA RAI 4 (Reference 23), the NRC staff requested additional information regarding the need to enter cold shutdown to establish safe and stable conditions. In its response to SSA RAI 4 (Reference 10), the licensee indicated that the NSCA has shown that there are no fire scenarios that would require that the plant achieve cold shutdown to meet any nuclear safety performance goals, so the continuation of Reactor Coolant System (RCS) cooldown beyond hot shutdown conditions is not a prerequisite for establishing safe and stable conditions. The licensee submitted a revised LAR Attachment B, Table B-2, Section 3.1.1.9, to clarify their position concerning the safe and stable conditions required by NFPA 805. The NRC staff concludes that the licensee's response to the RAI and the corresponding attribute alignment are acceptable because the licensee confirmed that there are no fire scenarios that would require the plant to achieve cold shutdown to meet any nuclear safety performance goals and that the continuation of RCS cooldown beyond hot shutdown conditions is not a prerequisite for establishing a safe and stable condition which meets the intent of NEI 00-01 attribute 3.1.1.9.

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

The licensee did not identify any attributes in this category.

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

The licensee did not identify any attributes in this category.

3.2.1.5 Attribute Alignment – Not in Alignment

The licensee did not identify any attributes in this category.

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 nuclear safety capability assessment requirements using NEI 00-01, Revision 1, with a gap analysis to the NRC-endorsed process in Chapter 3 of NEI 00-01, Revision 2. The licensee documented the results of the review in LAR Attachment B, Table B-2, in accordance with NEI 04-02, Revision 2, and the gap analysis of NEI 00-01, Revision 2.

Based on the information provided in the licensee's submittal, as supplemented, the NRC staff accepts the method the licensee used to perform the NSCA with respect to the selection of systems and equipment, selection of cables, and identification of the location of nuclear safety equipment and cables, as required by NFPA 805, Section 2.4.2. The NRC staff concludes that the licensee's method is acceptable because it either meets the NRC-endorsed guidance directly or meets the intent of the endorsed guidance and adequate justification was provided.

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 FPP requirements based on Appendix R to 10 CFR 50 and NUREG-0800, Section 9.5.1 (Reference 79), 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 hot shutdown conditions following any fire occurring prior to establishing cold shutdown.

For the most limiting fire scenarios in every fire area, the licensee has documented the availability of long term cooling using the RHR system, in either the normal shutdown cooling mode or alternate shutdown cooling mode, or the Core Spray System, all of which are characterized by low pressure injection and at least one safety relief valve available to provide core flow. The RHR Service Water system rejects decay heat to the ultimate heat sink.

The licensee stated that initiation of RHR in the suppression pool cooling mode does not imply that the plant would proceed to cold shutdown and that following stabilization at hot shutdown, a long term strategy for reactivity control, decay heat removal, and inventory/pressure control would be determined based on the extent of equipment damage. The licensee further stated that if an assessment of the post-fire conditions indicated that placing the RHR system in the shutdown cooling mode would be advisable, then repair activities would commence in a safe and controlled manner to restore plant equipment necessary for reactor cooldown.

In SSA RAI 05 (Reference 23), the NRC staff requested a more detailed description of the systems, evolutions, and resources, required to maintain safe and stable conditions in an effort to understand the risk, given that safe and stable conditions were not included as an end state in the Fire Probabilistic Risk Assessment (FPRA). In its response to SSA RAI 05 (Reference 12), the licensee stated that they have qualitatively evaluated the actions and activities required to maintain those conditions and concluded that they represent relatively low risk evolutions and that the factors considered were related to simplicity, ensured equipment availability, and the routine nature of replenishing commodities. The licensee further stated that following the establishment of safe and stable conditions, the ability to control reactor pressure, inventory, and temperature requires limited operator involvement as the actions are characterized by simple manipulations of valves and/or pump controls and process instrumentation is readily available at appropriate locations. The licensee further stated that actions required to maintain safe and stable conditions are limited to simple control activities such as adjusting service water and/or RHR flow and that in the longer term, no additional operator interventions are required other than the replenishment of onsite commodities, such as nitrogen and fuel oil, and therefore, the risk associated with these activities is very low.

In its response to SSA RAI 05 (Reference 12), the licensee also stated that there are no fire scenarios in which "off-shift" personnel are required to perform any actions in order to achieve and maintain safe and stable conditions within 24 hours. The licensee further stated that for fire initiated events where the Emergency Response Organization is activated, personnel are available to ensure that the longer term steps required to replenish onsite commodities can be reliably accomplished and that supplemental resources are available for actions required to replenish commodities for the long term.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that safe and stable conditions can be maintained through relatively low risk evolutions that are related to simplicity, equipment availability, and the routine nature of replenishing commodities.

Based on a review of the licensee's analysis as described in the LAR, as supplemented, the NRC staff concludes that the licensee has provided 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

The limitations of 10 CFR 50.48(c)(2)(iii), "Use of feed-and-bleed," are not applicable to BWRs.

3.2.4 Assessment of Multiple Spurious Operations

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

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

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

The NRC staff reviewed LAR Section 4.2.1.4, "Evaluation of Multiple Spurious Operations," and LAR Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," to determine whether the licensee has adequately addressed MSO concerns. As described in the LAR, the licensee's process for identification and evaluation of MSOs used an expert panel and followed the guidance of NEI 04-02 (Reference 7), RG 1.205 (Reference 4), and FAQ 07-0038, "Lessons

Learned on Multiple Spurious Operations," Revision 3 (Reference 62). The licensee stated that the expert panel consisted of individuals from the site and corporate engineering departments, plant operations department, corporate PRA department, as well as industry consultants.

Attachment F to the LAR stated that the licensee conducted an initial expert panel review in 2010 and a second review in 2012. The licensee further stated that prior to the initial review, the panel was provided with training that included PRA and SSA discussion, as well as key points of the analysis. The licensee further stated that the expert panel's sources for identifying MSOs included the SSA, generic lists (e.g., from the BWR Owners Group), self-assessment results, PRA insights, results of other sites expert panels (H.B. Robinson and Crystal River) and operating experience, and that the results of the initial review were integrated into the NSCA and the FPRA. The licensee further stated that the second review panel dispositioned open items from the initial expert panel review and addressed new MSOs identified since the initial review.

Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," of the LAR describes the process the licensee used to address MSOs. That process includes 5 steps:

1. Identify potential MSOs of concern;

2. Conduct an expert panel to assess plant specific vulnerabilities;

3. Update the Fire PRA model and nuclear safety capability assessment to include the MSOs of concern;

4. Evaluate for NFPA 805 Compliance; and

5. Document Results.

In LAR Attachment F, under the results for Steps 3, 4, and 5, the licensee stated that MSOs identified in Steps 1 and 2 were incorporated into the FPRA model and evaluated for inclusion in the NSCA. The licensee further stated that variances from deterministic requirements (VFDRs) were created where MSO combinations did not meet the deterministic requirements of NFPA 805, Section 4.2.3, and these VFDRs were addressed using the performance-based approach of NFPA 805, Section 4.2.4. The licensee further stated that based on the evaluations, components associated with the MSOs were added to the NSCA equipment list and logics, and cable tracing and circuit analysis was performed. The Brunswick FPRA quantified the fire-induced risk model containing the MSO pathways, and the MSO contribution is included in the FPRA results, including those associated with VFDRs in the FREs.

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

3.2.5 Establishing Recovery Actions

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

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

NFPA 805, Section 4.2.3.1 states that:

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

NFPA 805, Section 4.2.4, "Performance-Based Approach," states, in part, that:

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

The NRC staff reviewed LAR (Reference 8) Section 4.2.1.3, "Establishing Recovery Actions," and LAR Attachment G, "Operator Manual Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of recovery actions per NFPA 805.

The licensee used the endorsed guidance provided in NEI 04-02, Section 4.6 (Reference 7) and the guidance in FAQ 07-0030, "Establishing Recovery Action" (Reference 61), to establish the population of recovery actions being carried forward in the RI/PB fire protection program. Recovery actions addressed during the NFPA 805 transition process included the consideration of existing operator manual actions (OMAs) in the deterministic fire protection program, as well as those being added based on the VFDRs identified in the individual fire area assessments.

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

The licensee stated in LAR Attachment G, "Recovery Actions Transition" (Reference 8) that the primary control stations, as defined in RG 1.205, include the Remote Shutdown Panels located in the southeast corner of the 20-foot level of the Unit 1 and Unit 2 Reactor Buildings. The licensee described the activities necessary to enable the primary control station(s) following

Control Room abandonment in LAR Attachment G, Tables G-1 (Unit 1) and G-2 (Unit 2) as primary control station(s) activities.

In SSA RAI 14 (Reference 23), the NRC staff requested that the licensee describe the extent of previous approvals for OMAs for alternate shutdown. In its response to SSA RAI 14 (Reference 12), the licensee stated that because there is insufficient detail provided in historical submittals, or in subsequent safety evaluations, no previous approval was assumed for manual actions. The licensee further stated that those OMAs designated as "RA" in LAR Attachment G, Tables G-1 and G-2, were subject to a risk evaluation and included in the LAR for approval under the NFPA 805 transition process. The licensee further stated that the actions designated in LAR Attachment G as primary control station (PCS), make use of the controls installed on the remote shutdown panel of the Unit 1 and Unit 2 Reactor Buildings that have complete isolation capability from the main control room (MCR). Based on its review of the licensee RAI response, the NRC staff determined that the licensee omitted these actions from the risk evaluation because they are taken at the primary control station, as defined in Section 2.4(b) of RG 1.205, Revision 1 (Reference 4). The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee provided a description of the previous approvals for OMAs for alternate shutdown.

OMAs meeting the definition of an RA are required to comply with the NFPA 805 requirements outlined above. Some OMAs may not be required to demonstrate the "availability of a success path," in accordance with NFPA 805, Section 4.2.3.1, but may still be required to be retained in the RI/PB FPP because of the DID considerations, as described in Section 1.2 of NFPA 805. In each instance, the licensee determined whether a transitioning OMA was an RA or not necessary for the post-transition RI/PB FPP.

The licensee stated that all credited RAs, as listed in LAR Attachment G (including DID RAs), were subjected to a feasibility review. The NRC staff determined that in accordance with the NRC-endorsed guidance in NEI 04-02, the feasibility criteria used in the licensee's assessment process were based on the 11 criteria described in FAQ 07-0030 (Reference 61), and each of the individual feasibility attributes were addressed. LAR Attachment G, Table G-1, "Unit 1 Recovery Actions and Activities Occurring at the Primary Control Station(s)," and Table G-2, "Unit 2 Recovery Actions and Activities Occurring at the Primary Control Station(s)" describe each recovery action associated with disposition of a VFDR from the fire area assessments as documented in LAR Attachment C, "Fire Area Transition."

The licensee included modifications in the LAR to provide improved panel accessibility to reduce the time for the alternate SSD operator to perform recovery actions on panels. This plant change is described in LAR Attachment S, "Modifications and Implementation Items," Table S-1, as modification 9. The NRC staff concludes that the licensee's action to "provide improved panel accessibility to improve the time for alternate safe shutdown operator to perform RAs; modify the panels for battery charger 2B-1 and 2B-2 AC power transfer switch panels (2-L6A and 2-L6B) and battery charger 1B-1 and 1B-2 AC power transfer switch panels (1-L6A and 1-L6B) to allow access to the switches without having to remove 10 bolts for each panel," is acceptable because it will incorporate the provisions of NFPA 805, Chapter 3, and because the action is required by the proposed license condition.

Based on the above considerations, 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, thereby meeting the requirements of 10 CFR 50.48(c). The NRC staff concludes that the feasibility criteria applied to RAs are acceptable based on conformance with the endorsed guidance contained in NEI 04-02 and successful completion of modification 9, as described in LAR Attachment S, Table S-1.

3.2.6 Plant Specific Treatments or Technologies

3.2.6.1 Very Early Warning Fire Detection System

The licensee proposed the installation of several very early warning fire detection systems (VEWFDS) (also referred to as "incipient detection") to monitor conditions inside certain Control Room electrical cabinets and for area wide incipient detection in fire areas CB-1 (Cable Access Way Unit 1) and CB-2 (Cable Access Way Unit 2), and fire zones CB-05 (CSR Unit 1), and CB-06 (CSR Unit 2) in fire area CB-23E (Control Room Extended). The following discussion is based on the information provided in LAR Attachment S, Table S-1, modification items 1 and 12; LAR Attachment C, Table B-3 (fire areas CB-1, CB-2, and CB-23E); LAR Section 4.8.3.6 "Incipient Detection in Main Control Boards," and the responses to FPE RAI 01 and SSA RAI 01 (Reference 12).

3.2.6.1.1 Area Wide Installation

The licensee indicated that these area wide detection systems would be treated as standard spot smoke detectors, as covered by NFPA 805, Section 3.8.2. The licensee further stated in LAR Attachment S, Table S-1, modification item 12 that "No additional PRA credit given beyond that for standard smoke type spot fire detection." The NRC staff concludes that this is acceptable because of the licensee's compliance with this section of the standard, as documented in LAR Attachment A, as discussed in SE Section 3.1.

3.2.6.1.2 Main Control Room Cabinet Installation

VEWFDS is being installed in the Main Control Room (fire area CB-23E) cabinets indicated in LAR Attachment S, Table S-1, modification item 1. The decision of acceptability of the current application was not based on PRA credit for this installation. Detailed discussion of PRA credit for this installation can be found in SE Section 3.4.

In FPE RAI 01 (Reference 23), the NRC staff requested that the licensee provide more details regarding NFPA code(s) of record, proposed installation configuration, acceptance testing, sensitivity and setpoint control(s), alarm response procedures and training, and routine inspection, testing, and maintenance that will be implemented to credit the VEWFDS. In its response to FPE RAI 01 (Reference 12), the licensee stated that installation and testing will be performed in accordance with the NFPA code of record and the original equipment manufacturer requirements. The licensee further stated that training and qualification of installation technicians associated with the installation of VEWFDS will be in accordance with applicable fleet and site procedures for the conduct of maintenance and construction activities and that training requirements will be finalized and documented as a part of the engineering

change process. The licensee further stated that final installation and commissioning of the system, including acceptance testing, sensitivity and setpoint control(s), will be performed by the original equipment manufacturer with assistance and support of site personnel and that regular and preventative maintenance will also be in accordance with the original equipment manufacturer requirements, NFPA 72, NFPA 76, and the plant's preventative maintenance program. The licensee stated that the risk reduction credited for the use of in-cabinet VEWFDS is in accordance with method elements, limitations and criteria of NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," Chapter 13 (Reference 45) and FAQ 08-0046, "Incipient Fire Detection Systems" (Reference 65), including the closeout memo.

In FPE RAI 01.01 (Reference 24), the NRC staff requested more detail regarding the code(s) of record for the VEWFDS. In its response to FPE RAI 1.01 (Reference 15), the licensee stated that the code of record for design, installation, and maintenance of the new VEWFDS detection system is NFPA 72, "National Fire Alarm and Signaling Code," 2010 Edition (Reference 80). The licensee further stated that NFPA 76, "Standard for the Fire Protection of Telecommunications Facilities," 2009 Edition (Reference 83), is being used as a part of the design basis with respect to transport time in order to ensure that the VEWFDS meets the performance goals for proper credit in the FPRA. The licensee also stated that from Section 8.5.3.1 of NFPA 76, the maximum transport time of 60 seconds is being used as a design basis for the VEWFDS.

The licensee stated that it plans to use a multi-zone, single detector with aspiration tubing provided for one or more cabinets, as this is the most feasible design. The licensee further stated that the VEWFDS will include continuously monitored trouble annunciation consisting of a circuit supervisory signal for faults in the detector(s) or a failure of one of the system modules and that any detector or system fault condition would be annunciated, investigated immediately, and appropriate compensatory measures implemented until the fault condition is corrected. The licensee further stated that in addition to the continuously monitored supervisory trouble indication, the VEWFDS will receive quarterly surveillance testing and annual maintenance as recommended by the original equipment manufacturer and that the VEWFDS will be integrated into the existing fire detection system with Main Control Room annunciations.

The licensee stated that control room operators will respond to the alarms in accordance with plant operating procedures and that qualified on-shift operations and/or maintenance personnel will respond to investigate all alarms without delay and with the same response urgency, and will ensure that there is continuous attendance of the affected area until the condition is resolved. The licensee further stated that responding personnel will have basic training sufficient to initiate early firefighting activities (i.e., use of portable fire extinguisher equipment) such that the expectation will be satisfied that if a developing fire is discovered, there will be an immediate intervention to suppress/control the fire.

The NRC staff concludes that the licensee's responses to the RAIs are acceptable and also that the fire protection aspects related to the proposed installation, testing, and operation of the VEWFDS at Brunswick is acceptable because:

- The installation will be performed in accordance with the appropriate NFPA codes and standards and the equipment manufacturers' requirements.

- The systems will be properly tested during commissioning such that the alert and alarm triggers will be set to provide an appropriate level of sensitivity without unnecessary nuisance or spurious alarms.

- The licensee's configuration and design control process will control and maintain the setpoints for both alert and alarm functions.

- The equipment will be periodically tested and maintained in accordance with the original equipment manufacturer, NFPA 72, "National Fire Alarm and Signaling Code," NFPA 76, "Standard for the Fire Protection of Telecommunications Facilities," and the licensee's preventative maintenance program.

- First responders will be trained in the use of fire extinguishers and instructed to suppress or control a fire that breaks out in the alarming cabinet.

- The licensee's procedure will require the first responders to provide continuous attendance of the affected area until the condition is resolved. .

- The installation of the incipient detection system is included in LAR Attachment S, Table S-1, modification item 1, which is required by the proposed license condition.

3.2.7 Conclusion for Section 3.2

The NRC staff reviewed the licensee's LAR, as supplemented, for conformity with the requirements contained in NFPA 805, Section 2.4.2, regarding the process used to perform the nuclear safety capability assessment. 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 nuclear safety performance criteria.

The NRC staff reviewed the licensee's process to identify and analyze MSOs. Based on the information provided in the LAR, as supplemented, the process used to identify and analyze MSOs is considered comprehensive and thorough. The NRC staff determined that 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 into the NSCA, as well as the applicable FREs. The NRC staff considers the licensee's approach for assessing the potential for MSO combinations to be acceptable because it was performed in accordance with NRC-endorsed guidance.

The NRC staff concludes that the process used by the licensee to review, categorize, and address recovery actions during the transition is consistent with the NRC-endorsed guidance contained in NEI 04-02 and RG 1.205, and 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.

The NRC staff reviewed the proposed installation of a VEWFDS to monitor conditions in certain key electrical cabinets and certain fire areas as area wide detection. Based on the information provided in the LAR, as supplemented, the NRC staff concludes that the fire protection aspects regarding installation, testing, and operation of the proposed VEWFDS installation are acceptable because the installation, testing, and operation of the system will be done in accordance with appropriate NFPA codes, original equipment manufacturer requirements, and the licensees preventative maintenance program requirements.

3.3 Fire Modeling Performance-Based Approach

NFPA 805 (Reference 3) allows both fire modeling and fire risk evaluations (FREs) as performance-based alternatives to the deterministic approach outlined in the standard. These two performance-based approaches are described in NFPA 805, Sections 4.2.4.1 and 4.2.4.2, respectively. Although fire modeling and FREs are presented as two different approaches for performance-based compliance, the FRE approach generally involves some degree of fire modeling to support engineering analyses and fire scenario development. NFPA 805, Section 1.6.18, defines a fire model as a "mathematical prediction of fire growth, environmental conditions, and potential effects on structures, systems, or components based on the conservation equations or empirical data."

The NRC staff reviewed LAR (Reference 8) Section 4.5.2, "Performance-Based Approaches," which describes how the licensee used fire modeling as part of the transition to NFPA 805, and LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," which describes how the licensee performed fire modeling calculations in compliance with the NFPA 805 performance-based evaluation quality requirements for fire protection systems and features, to determine whether the fire modeling used to support transition to NFPA 805 is acceptable.

In LAR Section 4.5.2, the licensee stated that the fire modeling performance-based approach was not used for the NFPA 805 transition. The licensee used the FRE performance-based approach (i.e., FPRA) with input from fire modeling analyses. Therefore, the NRC staff reviewed the technical adequacy of the FREs, including the supporting fire modeling analyses, as documented in Section 3.4.2 of this SE, to evaluate compliance with the NSPC.

The licensee did not propose any fire modeling methods to support performance-based evaluations in accordance with NFPA 805, Section 4.2.4.1, as the sole means for demonstrating compliance with the nuclear safety performance criteria. There are no plant-specific fire modeling methods acceptable for use to support compliance with NFPA 805, Section 4.2.4.1, as part of this licensing action supporting the transition to NFPA 805 at Brunswick.

3.4 Fire Risk Assessments

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

method in accordance with NFPA 805, Section 4.2.4.2. The fire modeling performance-based method of NFPA 805, Section 4.2.4.1, was not used for this application.

NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluations," states, in part, that:

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

3.4.1 Maintaining Defense in Depth and Safety Margins

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

3.4.1.1 Defense-in-Depth (DID)

NFPA 805, Section 1.2, states the following:

> Protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations is paramount to this standard. The fire protection standard shall be based on the concept of defense in depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:
>
> (1) Preventing fires from starting
>
> (2) Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage
>
> (3) Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

The NRC staff reviewed LAR Section 4.5.2.2, "Fire Risk Approach," LAR Section 4.8.1, LAR Attachment C section titled "Defense in Depth," LAR Table C-2, "Considerations for Defense-in-Depth Determination," and LAR Table B-3, "Attachment C – NEI 04-02 Table B-3-Fire Area Transition," which included information on "Required Regulatory Systems," in order to determine whether the principles of DID were maintained in regard to the planned transition to NFPA 805 at Brunswick.

When implementing the performance-based approach, the licensee followed the guidance contained in Section 5.3, "Plant Change Process," of NEI 04-02 (Reference 7), which includes a detailed consideration of defense-in-depth and safety margins as part of the change process. The licensee documented the method used to meet the DID requirements of NFPA 805 in LAR Attachment C, Table B-3. LAR Attachment C, Table B-3 and documents the results of the licensee's review of fire suppression and fire detection systems at Brunswick.

The licensee developed a methodology for evaluating DID which defines each of the three DID elements identified in NFPA 805, Section 1.2, referred to as echelons 1, 2, and 3. This method is described in LAR Section 4.5.2.2, and LAR Attachment C, which was updated to provide more information about the licensee's approach in response to SSA RAI 9 (References 23 and 12). The method for addressing DID was implemented in the FREs performed on each performance-based fire area. The fire risk evaluations evaluate variances from deterministic requirements (VFDRs) using an integrated assessment of risk, DID, and safety margins. Accordingly, each performance-based FRE documents review of DID. This includes: (1) the fire protection systems/features required to either meet the deterministic criteria of NFPA 805, Section 4.2.3, or to support the FPRA; (2) indication of whether changes or improvements are necessary for each fire protection system/feature to maintain a balance among the DID echelons; and 3) justification or basis for why the required fire protection systems/features are adequate for DID. As such, this portion of the FRE documentation is the licensee's internal record of the systems required to meet the NSPC and DID requirements of NFPA 805.

Based on its review of the response to SSA RAI 9, and the FREs during its audit of the Brunswick NFPA 805 transition to an RI/PB fire protection program (RI/PB FPP), the NRC staff concludes that the licensee has systematically and comprehensively evaluated fire hazards, area configuration, detection and suppression features, and administrative controls in each fire area and concludes that the methodology as proposed in its LAR adequately evaluates DID against fires as required by NFPA 805, and therefore, the proposed RI/PB FPP adequately maintains DID.

3.4.1.2 Safety Margins

NFPA 805, Section 2.4.4.3 states, in part, that:

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

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

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

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

LAR Section 4.5.2.2, "Fire Risk Approach," discusses how safety margins are addressed as part of the FRE process and that this process is based on the requirements of NFPA 805, industry guidance in NEI 04-02, and RG 1.205. An FRE was performed for each fire area containing VFDRs. The FREs contain the details of the licensee's review of safety margins for each performance-based fire area. In response to SSA RAI 9 (References 23 and 12), the licensee further described the methodology used to evaluate safety margins in the FREs to include the following evaluations and determinations:

(1) Fire modeling for the fire PRA was specifically reviewed for adequate safety margin and, in general, was developed utilizing industry, NRC, and National Institute of Standards and Technology accepted codes, supported by guidance that includes NUREG/CR-6850 (References 43- 45), NEI 04-02, and associated Frequently Asked Questions resolutions as described in LAR Section 3.4 and specifically identified throughout the LAR.

(2) Plant system performance was evaluated given the specific demands associated with postulated fire events. The methods, input parameters, and acceptance criteria utilized in the risk-informed, performance-based analysis were reviewed against the plant design basis events. This evaluation determined the safety margin established in the plant design basis events was preserved.

(3) In general, development of the PRA logic model and the fire PRA application was performed in accordance with industry accepted codes and standards including 10 CFR 50.48(c), NFPA 805 (2001 edition) (Reference 3), ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," (Reference 38) and RG 1.200, Revision 2 (Reference 37).

The safety margin criteria described in NEI 04-02, Section 5.3.5.3 and the LAR, as supplemented, are consistent with the criteria as described in RG 1.174 (Reference 36),and therefore, acceptable. Brunswick used appropriate codes and standards (or NRC guidance), and met the safety analyses acceptance criteria in the licensing basis, or, through the application of its FPRA in its FREs, provided sufficient margin to account for analysis and data uncertainty. Also, Brunswick has removed the unacceptable methods or has committed to updating the fire PRA. Based on its review of the LAR, the response to SSA RAI 9, and the FREs during its audit of the Brunswick NFPA 805 transition RI/PB FPP, the NRC staff concludes that the licensee's approach has adequately addressed the issue of safety margins in the implementation of the fire risk evaluation process.

3.4.2 Quality of the Fire Probabilistic Risk Assessment

The objective of the PRA quality review was to determine whether the plant-specific PRA used in evaluating the proposed LAR was 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."

The licensee developed its internal events PRA during the Individual Plant Examination process and continued to maintain and improve the PRA as Regulatory Guide 1.200, "An Approach For Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities" (Reference 37) and supporting industry standards have evolved. The licensee developed its fire PRA model using the guidance of NUREG/CR-6850, "EPRI/NRC-RES, Fire PRA Methodology for Nuclear Power Facilities" (Reference 43-45). The licensee developed its Fire PRA model for both Level 1 (core damage) and partial Level 2 (large early release) PRA during at-power conditions. For the development of the fire PRA, the licensee modified its internal events PRA model to capture the effects of fire.

In LAR Section 4.8.2, the licensee stated that no significant plant changes (beyond those identified and scheduled to be implemented as part of the transition to a fire protection program based on NFPA 805) are outstanding with respect to their inclusion in the FPRA model. Based on this information, the NRC staff concludes that upon completion of the modifications listed in LAR Attachment S, Table S-1 that are credited in the FPRA, the FPRA model meets the above criteria; will represent the current, as-built, as-operated configuration; and is, therefore, capable of being adapted to model both the post-transition and compliant plant as needed.

The licensee identified administrative controls and processes used to maintain the Fire PRA model current with plant changes and to evaluate any outstanding changes not yet incorporated into the PRA model for potential risk impact as a part of the routine change evaluation process. Further, as described in SE Section 3.8.3, the licensee has a program for ensuring that developers and users of these models are appropriately trained and qualified. Therefore, the NRC staff concludes that the PRA has the capability to support post-transition plant change evaluations to support, for example, the self-approval process, after any changes required during implementation are completed.

3.4.2.1 Internal Events PRA Model

The licensee's evaluation of the technical adequacy of the portions of its internal events PRA model used to support development of the Fire PRA model consisted of full scope peer review performed in April 2010 using the NEI 05-04 process (Reference 82), and the combined PRA standard, ASME/ANS-RA-Sa-2009 (Reference 38), as clarified by RG 1.200, Revision 2 (Reference 37). The internal events PRA model that was reviewed for the focused scope peer review serves as the basis for the Fire PRA used in performing PRA evaluations for the LAR.

For many supporting requirements (SRs), there are three degrees of "satisfaction" referred to as Capability Categories (CC) (i.e., I, II, and III), with I being the minimum, II considered widely acceptable, and III indicating the maximum achievable scope/level of detail, plant specificity, and realism. For other SRs, the CCs may be combined (e.g., the requirement for meeting CCI may be combined with II), or the requirement may be the same across all CCs so that the requirement is simply met or not met. For each SR, the PRA reviewer from the peer review team designates one of the CCs or indicates that the SR is met or not met.

LAR Attachment U, Table U-1 provides the licensee's dispositions to all 44 facts and observations (F&Os) defined as findings per peer review guidelines (Reference 82). Twenty two of these F&Os were strictly related to internal flooding. In general, an F&O is written for any SR that is judged not to be met or does not fully satisfy Capability Category II of the ASME standard, consistent with RG 1.200, Revision 2.

As described in LAR Attachment U, the licensee dispositioned each F&O by assessing the impact of the F&O on the Fire PRA and the results for the NFPA 805 application. The NRC staff requested additional information to assess the adequacy of some of the F&O dispositions. The NRC staff evaluated each F&O and the licensee's disposition in LAR Attachment U to determine whether the F&O had any significant impact on the FPRA. The NRC staff's conclusions regarding Brunswick's resolution of each of F&O are summarized in the NRC's Record of Review dated October 17, 2014 (Reference 83). F&Os requiring additional information for the staff to complete its review are discussed below.

In PRA RAI 17 (Reference 23), the NRC staff requested that the licensee identify any changes made to the Internal Events PRA subsequent to the last full-scope peer review that met the definition of a "PRA upgrade" in the ASME/ANS PRA Standard (Reference 38). In its response to PRA RAI 17 (Reference 11), the licensee stated that no changes meeting the definition of a PRA upgrade were made to the internal events modeling for Fire PRA since the peer reviews. The NRC staff finds this response to be acceptable because it is based on the ASME/ANS PRA Standard and RG 1.200 guidance on PRA upgrades.

The NRC staff reviewed the licensee's dispositions of Internal Flooding F&Os to gain potential insights on the technical adequacy of the fire events PRA or the NFPA 805 application. Many of the dispositions provided for the internal flooding PRA F&Os, presented in LAR Attachment U, Table U-1, explained that internal flooding F&Os have no impact on the Fire PRA or NFPA 805 application. In light of this, the NRC staff requested that the licensee provide additional information in PRA RAIs 18.g and RAI 18.h (Reference 23) about how the potential for fire-induced flooding or spray impacts were addressed in the Fire PRA. In its response to PRA RAIs 18.g and 18.h (Reference 11), the licensee did not discuss spray effects in detail but stated that no additional risk due to internal flooding was identified.

In RAI 18.g.01 (Reference 25), the NRC staff requested that the licensee provide additional information on potential fire-induced effects such as spray and interfacing system LOCAs. In its response to PRA RAI 18.g.01 (Reference 16), the licensee explained that fire induced pipe-break flooding events were addressed in the Fire PRA, specifically: (1) ISLOCA for Shutdown Cooling (SDC) suction line, and (2) water hammers in the pump discharge lines for Core Spray and Residual Heat Removal. The licensee further explained that the SDC flooding scenario is not a

major contributor to risk because the inboard and outboard containment isolation valves are powered from diverse sources and are closed while the reactor is at power with the inboard valve de-energized. The licensee also explained that water hammer events are conservatively modeled in the Fire PRA by postulating failed equipment and spray effects in the same room as the assumed pipe break. The NRC staff considers this issue to be resolved because the licensee addressed the potential impacts of flooding – including fire-induced spray effects and ISLOCAs - on the Fire PRA.

As a result of the review of the LAR, as supplemented (Reference 9) and modified in response to the RAIs (References 11 and 16), the NRC staff concludes that the Brunswick internal events PRA is technically adequate because its quantitative results demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines of RG 1.174. To reach this conclusion, the NRC staff reviewed all F&Os provided by the peer reviewers and determined that the resolutions of the F&Os support the determination that the quantitative results are adequate or had no significant impact on the FPRA. Accordingly, the NRC staff concludes that the licensee has demonstrated that the internal event PRA meets the guidance in RG 1.200, Revision 2; it is reviewed against the applicable SRs in ASME/ANS-RA-Sa 2009; and it is technically adequate to support the FREs and other risk calculations required for the LAR.

3.4.2.2 Fire PRA Model

The licensee evaluated the technical adequacy of the Brunswick Fire PRA model by conducting a peer review using the NEI 07-12 process (Reference 84), and the combined PRA standard, ASME/ANS-RA-Sa-2009 (Reference 38), as clarified by RG 1.200, Revision 2 (Reference 37), The full scope peer review of the FPRA was performed in December 2011 and serves as the basis for the quantitative risk evaluations for the LAR.

LAR Attachment V, Table V-1, provides the licensee's dispositions to all 55 F&Os that were characterized as Findings per peer review guidelines (Reference 84) which included F&Os against SRs that were met, not met, achieved Capability Category I, II, or III (or some combination thereof if so grouped), or were not applicable. The licensee did not provide a separate table of SRs meeting only Capability Category I or were not met, but stated in the LAR that SRs not assessed as meeting CC II were included in LAR Attachment V, Table V-1.

In PRA RAI 17 (Reference 23) NRC staff requested that the licensee identify any changes made to the Fire PRA that are consistent with the definition of a "PRA upgrade" since the last full-scope peer review of PRA models as defined by the ASME/ANS PRA Standard (Reference 38). In its response to PRA RAI 17 (Reference 11), the licensee stated that no changes that met the definition of a PRA upgrade were made following the peer review. The NRC staff finds this response to be acceptable because it is based on the ASME/ANS PRA Standard and RG 1.200 guidance on PRA upgrades.

As described in LAR Attachment V, the licensee dispositioned each F&O by assessing the impact of the F&O on the Fire PRA. The NRC staff evaluated each F&O and the licensee's respective disposition in LAR Attachment V to determine whether the F&O had any significant impact on the LAR. The NRC staff's conclusions on the resolution of each F&O are

summarized in the NRC's Records of Review dated October 17, 2014 (Reference 82). F&Os requiring additional information for the staff to complete its review are discussed below.

The NRC staff determined that the disposition of F&O 1-19 presented in LAR Attachment V, Table V-1, stated that the zones of influence (ZOIs) associated with a 143 kW heat release rate (HRR) (75th percentile) fire was used (instead of the nominal 317 kW HRR) in all fire areas except the turbine building. In PRA RAI 1.d (Reference 23), the NRC staff requested that the licensee provide a description of administrative controls; the results of a plant experience review; records of violations of transient combustible controls; and a description of other key factors used to support the reduced transient fire HRR in the FPRA. This RAI also asked the licensee to identify any other HRRs used in the FPRA.

In its response to PRA RAI 1.d (Reference 11), the licensee clarified that the only transient HRRs used were a 317 kW HRR in the turbine building, and a 143 kW HRR in all other plant areas. The NRC staff reviewed the licensee's response and concluded that it did not provide sufficient information on the new procedures and results of reviews of plant experience and records of violations of transient combustible controls. In PRA RAI 1d.01 (Reference 25) the NRC staff requested this additional information from the licensee. In its response to PRA RAI 1d.01 (Reference 16), the licensee clarified that the new controls will ensure that all fire loads will be reviewed prior to being allowed into the critical areas and provided the results of a review of plant records related to transient combustible controls including an evaluation of the last 3 years of violations documented in its Fire Protection Health Reporting system. The NRC staff reviewed this information and found that there were violations presented in this summary that appeared to indicate that a transient fire exceeding the reduced HRR used in the Fire PRA could have occurred. Consequently, the NRC staff issued PRA RAI 1d.02 (Reference 26), which requested that the licensee describe how these violations were considered in setting the reduced HRR used in the Fire PRA. In its response to PRA RAI 1d.02 (Reference 18), the licensee explained that none of the reported violations could have resulted in a transient fire exceeding the reduced HRRs credited in the Fire PRA. The NRC staff finds that the licensee's use of the lower HRR acceptable because the licensee provided a characterization of the past plant-specific transient combustible violations indicating that the lower HRR is achievable and has established an improved procedure that will provide confidence that all fire loads will be limited to the lower HRR.

In PRA RAI 1.e (Reference 23), regarding the treatment of sensitive electronics (cited in the disposition of F&O 1-20 presented in Table V-1 of the LAR), the NRC staff requested that the licensee provide a description of how the FPRA addressed fire impact on sensitive electronics and requested an assessment of the expected impact on Fire PRA results if fire impact on sensitive electronics was not evaluated. In its response to PRA RAI 1.e (Reference 14), the licensee stated that the fire impact on sensitive electronics was not originally assessed but that a sensitivity study showed that the effect on transition risk was negligible. In response to PRA RAI 24 (Reference 25), the licensee added Implementation Item 13 to LAR Attachment S, Table S-2 (Reference 16), to incorporate the treatment of sensitive electronics consistent with FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (Reference 96) into the Fire PRA before self-approval of post-transition changes. Based on the results of a sensitivity study showing that the impact of modeling the effects of fire on sensitive electronics is negligible for the transition change in risk and the addition of Implementation Item 13, the NRC staff finds the

licensee's evaluation of sensitive electronics consistent with accepted guidance, and therefore, acceptable.

F&O 1-26 noted that no basis was provided for values assigned to the human error probabilities (HEPs) following main control room (MCR) abandonment. The NRC staff asked questions (RAIs 1.f.i – 1.f.vii) (References 23, 25, 26, and 27) and the licensee provided clarifications (References 11, 16, 18, and 19). In its response to PRA RAI 1.f, the licensee explained that the feasibility of operator actions was established by walk-throughs of alternate shutdown actions performed by operators and is supported by operator training. The licensee also explained that the challenges of timing, coordination, and communication are addressed in the shutdown procedure and is addressed in the human reliability analysis (HRA) used to develop the HEP. The NRC staff finds that the licensee has developed a basis for the values assigned to the HEPs.

In its response to PRA RAI 1.f.ii.02 (Reference 18), and RAI 1.f.ii.03 (Reference 19), the licensee clarified how CDF and LERF associated with MCR abandonment are developed. The licensee stated that all core damage scenarios following MCR abandonment are attributable to either loss of control or loss of habitability. Loss of control is considered to occur whenever fire induced equipment failures (from fires within or outside of the MCR) preclude bringing the plant to a SSD state from the MCR. The licensee stated that a loss of control leads directly to core damage which is quantified by the FPRA. If there is no loss of control, but the fire causes the control room to be abandoned due to loss of habitability, core damage scenarios are caused by failure to shutdown using the alternate SSD procedure.

The licensee developed an event tree and fault tree model to evaluate the likelihood of failure to shutdown using the alternate SSD procedure. The model includes the alternative SSD equipment, as well as 36 required operator actions, both at and away from Primary Control Stations (PCSs). The licensee clarified that no SSD equipment is failed by fire in the MCR so the likelihood of the scenarios in the event tree are independent of the original fire. The sum of sequence probabilities leading to unsuccessful shutdown following MCR abandonment is reported as a CCDP of 2.5E-1 for each unit. The CCDPs for loss of control and loss of habitability are combined with the fire ignition frequencies to estimate the CDF values. LERF is estimated as 10 percent of CDF. In its response to PRA RAI 1.f.iii.02 (Reference 18), the licensee explained that containment bypass scenarios constitute an important part of the Internal Events PRA results, but do not contribute to the MCR abandonment scenarios, and therefore, the estimated LERF contribution for internal events was considered conservative for fire scenarios.

The NRC staff finds the MCR abandonment method acceptable because complex scenarios (i.e., those that preclude shutting down the plant from the control room) are assigned a CCDP of 1.0 and less complex scenarios are assigned a CCDP based on a standard and acceptable PRA evaluation of the failure of equipment that is not affected by the fire or by failure of required operator actions.

In PRA RAI 1.g (Reference 23), based on F&O 1-30 in LAR Attachment V, Table V-1, on fire propagation associated with a Motor Control Center (MCC), the NRC staff asked the licensee to describe the approach and assumptions used to model MCC fires, and about how the sensitivity

study on MCC fires was performed. In its response to PRA RAI 1.g (Reference 11), the licensee explained that determination of "open" cabinets, in which fire propagation was considered, and "closed" (i.e., "well-sealed") cabinets, in which fire propagation was not considered, was established based on walkdowns and equipment qualification documents. The NRC staff requested further clarification of the assumption described in LAR Section 4.8.3.1 that one out of ten MCC fires in a closed cabinet may result in an open cabinet configuration (i.e., the fire propagates outside of the cabinet). In its response to PRA RAI 24 (Reference 16), the licensee clarified that it uses 0.1 as the probability that a closed MCC becomes open or an arcing fault has enough energy to open the MCC and damage cables above the cabinet.

Industry guidance (Reference 85) provides a proposed generic approach to develop the likelihood that a fire propagates outside of a 440V AC or higher electric cabinet and damages nearby targets. The analysis in the industry guidance yields a likelihood that a fire in a cabinet damages cables six inches above the cabinet and provides an approach for evaluating different physical configurations. The analysis supports a probability of 0.1 for cabinet breach given a fire and damaging nearby targets outside the cabinet. The NRC staff finds the use of this 0.1 probability acceptable, but only for the well-sealed MCC and not all electrical cabinets. The use of 0.1 probability has an adequate and acceptable technical basis for well-sealed MCC breach and fire damage because it is consistent with the available operating experience and it is systematically applied to a representative physical configuration. The Brunswick evaluation also uses a 0.1 for the probability of breaching a well-sealed MCC and damaging nearby targets outside the MCC, and therefore, the NRC staff finds the Brunswick analysis acceptable.

In its response to PRA RAI 1.h (Reference 14) regarding use of a reduced HRR for pump motor (electrical) fires as cited in F&O 1-32 presented in LAR Attachment V, Table V-1, the licensee provided a sensitivity study that removed credit for using this reduced HRR. The HRR of 69 kW used in the Fire PRA for the 98th percentile electrical pump fires is lower than the NUREG/CR-6850 (References 43-45) recommended 98th percentile HRR of 211 kW. In the sensitivity study, the NUREG/CR-6850 recommended HRR value of 211 kW was used, and the results demonstrate that the impact to the risk estimates is negligible. In response to PRA RAI 24 (Reference 16), the licensee added Implementation Item 13 to LAR Attachment S, Table S-2, to incorporate this analysis into the Fire PRA before self-approval of post-transition changes because it could have a greater impact on future plant change evaluations. The NRC staff finds the licensee's response to this RAI to be acceptable because the lower HRR is unimportant for transition, and the licensee added an implementation item to incorporate the higher HRR prior to implementing self-approval.

Based on the disposition of F&O 4-18 presented in LAR Attachment V, Table V-1, the NRC staff asked PRA RAI 1.i (Reference 23) and requested that the licensee describe how parametric data uncertainty was propagated and whether the licensee accounted for the state of knowledge correlation (SOKC) for fire event-specific parameters. In its response to PRA RAI 1.i (Reference 11), the licensee stated that only SOKCs for component failure probabilities based on component type were performed. In response to RAI 1.i.01 (Reference 17), the license evaluated the impact of SOKC among fire ignition frequency, circuit failure probabilities, and non-suppression probabilities. The licensee reported that, "the effect of SOKC on the Brunswick Fire PRA are minimal with regard to the impact on the risk estimates." The NRC staff finds that the licensee's SOKC evaluation is acceptable because it has evaluated the effect

SOKC on the transition change in risk, and demonstrated that it has the capability and input values to include SOKC evaluations when appropriate.

Based on the disposition of F&O 4-13 presented in LAR Attachment V, Table V-1, related to treatment of fire modeling uncertainty, the NRC staff issued PRA RAI 1.n (Reference 23), which asked about fire modeling parameters that were not treated quantitatively. In its response to PRA RAI 1.n (Reference 11), the licensee provided a qualitative assessment of the impact on fire modeling uncertainties on the Fire PRA. The response provides a table listing 28 sources of modeling uncertainties and assumptions and qualitatively describes the impact of each on the Fire PRA. This table included a qualitative indication of the degree of conservatism introduced to the Fire PRA and rationale for why the fire modeling or related assumptions would not have a significant impact on the Fire PRA. Based on the table and associated evaluation, the NRC staff finds that the licensee has appropriately addressed qualitative modeling uncertainties.

In PRA RAI 2 (Reference 23), the NRC staff asked whether the licensee used any deviations from NUREG/CR-6850 guidance. In its response to PRA RAI 2 (Reference 12), the licensee explained that possible deviations from NUREG/CR 6850 guidance are addressed in Section 4.8.3 of the original LAR supplement (Reference 9), or in response to other PRA RAIs provided in the May 15, 2013 (Reference 23), RAI letter from the NRC.

In its response to PRA RAI 3 (Reference 12), the licensee clarified that guidance from Section 6.5.7 of NUREG/CR-6850 was used to calculate hot work and transient fire frequencies; that administrative controls were not used to reduce the HRR in modeling transient fires; and that radiation levels preclude at-power access for the locations assigned an Influence Factor of "0." Given that a hot work and transient fire Influence Factor of "50" was not used in any fire area, the NRC staff requested that the licensee perform a sensitivity study using Influence Factors of "50" consistent with the guidance in FAQ 12-0064 (Reference 86). The results of the sensitivity study, provided in an update to LAR Section 4.8.3.11 (Reference 14), show that the impact on the transition risk estimates is negligible. In its response to PRA RAI 24 (Reference 16), the licensee added Implementation Item 13 to LAR Attachment S, Table S-2 which incorporates appropriate use of the factor of 50 into the Fire PRA before self-approval of post-transition changes. Based on the results of the sensitivity study and Implementation Item 13, the NRC staff finds that transients and hot work fires are included consistent with the guidance in FAQ 12-0064.

In PRA RAI 4 (Reference 23), the NRC staff requested that the licensee explain how transient fires were placed to include pinch points where CCDPs are highest for a given physical analysis unit (PAU). In its response to PRA RAI 4 (Reference 11), the licensee explained that its method places transient ignition sources such that their ZOIs include targets located in the PAU in a manner that encompasses pinch points as defined in Section 11.5.1.6 of NUREG/CR-6850. Based on the review of the licensee's analysis, the NRC staff determined that vertical and horizontal ZOIs for possible transient and hot work fires are placed in a fire compartment in a way to maximize the number of potential targets that can result in high CCDP values. In its response, the licensee also explained that ignition sources are not postulated on top of equipment or wedged between cable trays, and that fires were normally modeled from the floor level unless specific situations dictated otherwise. The NRC staff considers the licensee's

method of placing transient fires within a fire compartment acceptable because it is consistent with the guidance of NUREG/CR-6850.

In PRA RAI 6 (Reference 23) the NRC staff requested that the licensee describe how propagation of fires was addressed for the MCR. In its response to PRA RAI 6 (Reference 12), the licensee explained that fires in the main control board (MCB) panels and cabinets (including panels and cabinets in the back panel area) caused either a fire that damaged only equipment within the panel or cabinet (self-contained fires) or a fire that escaped the panel or cabinet and also damaged equipment within a ZOI. An air gap separating cabinets precluded propagation to neighboring cabinets unless the walkdowns identified significant communicative space between cabinets. The licensee also clarified that transient fires were not important contributors to cabinet fires because the MCR does not contain any open-backed cabinets. The response to PRA RAI 6 explained that an (as yet uninstalled) incipient fire detection system was assumed to eliminate fire damage from the self-contained fires and that no credit was taken for the installed ion smoke detection system.

In PRA RAI 6.01 (Reference 25) the NRC staff indicated that the ion smoke detection system could be credited consistent with NUREG/CR-6850, but credit for incipient fire detection systems in the MCR deviated from NUREG/CR-6850 and FAQ 08-0046 (Reference 65) guidance. In the RAI, the NRC staff also indicated that the licensee's approach subdivided the MCB fire ignition by the number of MCB fire scenarios, a step also not consistent with guidance in NUREG/CR-6850. In its response to PRA RAI 6.01 (Reference 17), and as part of the final aggregate evaluation in its response to PRA RAI 23 (Reference 17), the licensee stated that it re-performed the MCR analysis consistent with the guidance in NUREG/CR-6850. The response to PRA RAI 6.01 included Implementation Item 15 that was also included in LAR Attachment S, Table S-2, which states that a focused scope peer review will be performed on these new methods before self-approval of non-minimal post transition changes to the MCR. The NRC staff finds the PRA evaluation of fires in the MCR acceptable because the unacceptable methods have been replaced with acceptable methods, and a focused scope peer review will be performed on these new methods before the PRA is used to support post-transition changes to the MCR.

In PRA RAI 7 (Reference 23), the NRC staff requested that the licensee describe how fire-induced instrument failures were modeled in the Fire PRA. In its response to PRA RAI 7 (Reference 12), the licensee explained that instruments that are required for operator response and are assumed to be available in the HRA were explicitly modeled in the fault trees along with their related power supplies. The licensee further explained that cables associated with credited instrumentation were traced and that failure of instrumentation resulting from fire impact on cables was addressed. The licensee explains that any instrument cable impacted by fire is assumed to fail associated instruments, and that in scenarios where the minimum instrumentation required to support the action fails, operator action is not credited. The NRC staff considers this method of modeling the failure of instrumentation needed in response to fire events directly in the Fire PRA fault trees, and not crediting operator actions when minimum instrumentation fails, to be acceptable because fire-induced instrument failures important for operator response are treated consistent with the NUREG/CR-6850 guidance.

In PRA RAI 8 (Reference 23), the NRC staff requested that the licensee provide a discussion of whether heat loads from fires can fail additional equipment in rooms that do not credit heating,

ventilation and air conditioning (HVAC). In its response to PRA RAI 8 (Reference 13), the licensee provided a qualitative explanation for why heat loads from fires cannot fail additional equipment in rooms that do not credit HVAC. In PRA RAI 8.01 (Reference 25), the NRC staff requested that the licensee describe whether temperature-limited components can be affected by fire, and, if so, what are the impacts on the risk estimates. In its response to PRA RAI 8.01 (Reference 17), the licensee re-evaluated the potential ambient operating temperatures for temperature-limited components using Fire Dynamics Tools (FDTs) modeling. For those fire scenarios determined to release sufficient heat, a sensitivity analysis was performed to account for additional failures of temperature-limited components and reported little or no measureable impact from those failures on the fire risk estimates. The licensee also included this updated treatment of temperature limited components in the integrated analysis performed in response to PRA RAI 23 (Reference 25). The NRC staff finds that the licensee's evaluation of fire heat loads acceptable because the licensee evaluated the potential ambient operating temperatures for temperature-limited components and included the results of this analysis in the integrated analysis performed in response to PRA RAI 23.

In PRA RAI 9 (Reference 23), the NRC staff requested that the licensee describe how wrapped or embedded ("protected") cables were modeled in the Fire PRA. In its response to PRA RAI 9 (Reference 11), the licensee stated that some cables in Fire Area DG-05 are currently protected with Pyrocrete and that one modification (Modification 5) will protect some cables in Unit Substation 1L. The basis for the credit taken for protection of these cables and the acceptability of that credit is described in the response to FPE RAI 15.01 (Reference 15) discussed in SE Section 3.1.1.1.

In PRA RAI 11.a (Reference 23), the NRC staff indicated that no transient fires were postulated in the MCB area where operators manipulate controls and requested that the licensee provide the results of assessing transient fires in the MCB area. In response to PRA RAI 11.a (Reference 12), the licensee explained that transient fires were not placed in that area because, in their approach, transient fires are only placed near targets. In the MCB area, there are no targets, as there are no exposed cables and the electrical cabinets do not have open backs. The NRC staff concludes that the licensee's evaluation is acceptable because of the licensee's assessment that there are no targets in the MCB area where operators manipulate controls.

In PRA RAI 11.b (Reference 23), the NRC staff requested that the licensee provide justification of the modeling of MCR electrical cabinets as containing only single bundle cables. In its response to PRA RAI 11.b (Reference 12), the licensee explained that walkdowns were performed to identify whether cabinets in the MCR contained multiple or single bundle cables. However, as a simplification, fires in the MCB panels were modeled as single-bundle fires. In response to the RAI, the licensee provided an update to LAR Section 4.8.3 (Reference 14), that included the results of a sensitivity study in which fires in MCB panels were all treated as multi-bundle fires, but in which the licensee also added credit for the currently installed in-cabinet ion smoke detectors in the MCBs (to reduce the time to suppression by 5 minutes). The results of the sensitivity show that the impact of modeling MCBs as containing multiple versus single cable bundles - coupled with adding the ion detectors - has a negligible impact of the transition risk. In the response to PRA RAI 24 (Reference 16), the licensee added Implementation Item 13 to LAR Attachment S, Table S-2 (Reference 16), to incorporate this analysis into the Fire PRA before self-approval of post-transition. The NRC staff concludes that the licensee's

modeling of MCB is acceptable because the single bundle assumption has no impact on the transition risk estimates and because the licensee included an implementation item that will modify their analysis to model multiple cable bundles before self-approval.

In PRA RAI 13 (Reference 23), the NRC staff requested that the licensee provide an explanation of the asymmetry in CDF and LERF results between the two units. In its response to PRA RAI 13 (Reference 12), the licensee explained that the asymmetry in CDF and LERF results between units result from the fact that cable routing in the two units is not symmetrical, and the Fire PRA is very sensitive to spatial relationships between ignition sources and targets. The NRC staff concluded that the licensee's response adequately explained the difference in CDF and LERF results between the two units.

In PRA RAI 25 (Reference 27), the NRC staff requested that the licensee explain how potentially unanalyzed conditions reported in a recent License Event Report (LER) 14-004-00, "Fire Related Unanalyzed Condition that Could Impact Equipment Credited in Safe Shutdown Analysis" (Reference 87), are reflected in the Fire PRA and if they have any impact on the risk estimates. In its response to PRA RAI 25 (Reference 19), the licensee evaluated each of the conditions to determine the impact on the Fire PRA. This evaluation showed that each condition was either already modeled in the Fire PRA, did not impact the Fire PRA, or was bounded by an existing scenario. The NRC staff finds that licensee's evaluation supports the conclusion that no changes to the Fire PRA were needed from the conditions identified in the LER.

As a result of its review of the LAR, as supplemented, the NRC staff concludes that the Brunswick Fire PRA has sufficient technical adequacy and that its quantitative results, considered together with the sensitivity studies, can be used to demonstrate that the change in risk due to the transition to NFPA 805 meets the acceptance guidelines in RG 1.174 and that, after changes described in Implementation Items in LAR Attachment S, Table S-2, the Fire PRA will be acceptable to support post-transition self-approval evaluations.

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

The NRC staff performed detailed reviews of the fire modeling used to support the fire risk evaluations 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 fire modeling used in support of the development of the fire risk evaluations:

NFPA 805, Section 2.4.3.3, states, in part that:

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

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

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

The ZOI around transient ignition sources and liquid fuel spill fires was determined based on tables used in the Generic Fire Modeling Treatments (GFMTs) approach. The tables provide the horizontal and vertical dimensions of the ZOI for various ignition sources (transient fuel packages, small liquid fuel fires, open cabinets, and cable trays) and different types of targets (i.e., thermoplastic and thermoset cables as defined in NUREG/CR-6850, Volume 2 (Reference 44), and class A combustibles). The GFMTs approach was used as a basis for the scoping or screening evaluation in support of the fire PRA.

The ZOI tables used in the GFMTs approach were obtained by using a collection of algebraic models and correlations. The following algebraic fire models and correlations were used for this purpose.

- Heskestad Flame Height Correlation (Reference 49, Chapter 3)
- Heskestad Plume Temperature Correlation (Reference 49, Chapter 9)
- Modak Point Source Radiation Model (Reference 49, Chapter 5)

These algebraic models are described in NUREG-1805, "Fire Dynamics Tools (FDTs): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program" (Reference 48). Validation and Verification (V&V) of these algebraic models is documented in NUREG-1824, "Verification and Validation of Selected Fire

Models for Nuclear Power Plant Applications," Volume 3 (Reference 49). The V&V of the fire models that were used to support the fire PRA is discussed in Section 3.8.3.2 of this SE.

The following empirical models that are not addressed in NUREG-1824 were used in the development of the GFMTs approach:

- Shokri and Beyler flame radiation model (Reference 88)
- Mudan flame radiation model (Reference 89)
- Plume heat flux correlation by Wakamatsu et al. (Reference 90)
- Hydrocarbon spill fire size correlation (Reference 91)
- Babrauskas method for estimating the heat release rate of pool fires (Reference 92)
- Flame extension correlation (Reference 93)

Heskestad's flame height and plume temperature correlations and Modak's point source model were also used to determine the ZOI of cabinet fires. A refined approach based on the method by Shokri and Beyler was used to determine the radial ZOI for cabinet fires in the electronic equipment rooms (MCR back panel areas).

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 fire PRA used these approaches to further screen ignition sources, scenarios, and compartments that would not be expected to generate an HGL, and to identify the ignition sources that have the potential to generate an HGL for further analysis. The following correlations were used to determine the potential for the development of an HGL:

- Method of McCaffrey, Quintiere and Harkleroad for naturally ventilated compartments (Reference 49, Chapter 2)
- Method of Beyler for closed compartments (Reference 49, Chapter 2)

These HGL correlations are also described in NUREG-1805 and their V&V is documented in NUREG-1824, Volume 3. The licensee's approach to determine the potential for the development of an HGL includes a damage time adjustment to account for the thermal inertia of cable targets. This adjustment, referred to as the "heat soak" method, is based in part on Table H-5 in NUREG/CR-6850, Volume 2. A similar adjustment was used in assessing the damage time of cables located above a burning electrical cabinet.

The licensee's ZOI approach was used as a screening tool to distinguish between fire scenarios that required further evaluation and those that did not require further evaluation. The licensee stated that qualified personnel performed a plant walkdown to identify ignition sources and surrounding targets or SSCs in compartments, and assess whether these targets and SSCs

were within the ZOI of the ignition source. Based on the fire hazard present, these generalized ZOIs were used to screen from further consideration those specific ignition sources that did not adversely affect the operation of credited SSCs, or targets, following a fire. The licensee's screening was based on the 98th percentile fire HRR from the NUREG/CR-6850 methodology.

The Consolidated Model of Fire Growth and Smoke Transport (CFAST), Version 6 (Reference 94) was used for the MCR abandonment time calculations. Finally, Fire Dynamics Simulator (FDS), Version 5 (Reference 93) was used to analyze current transformer fire scenarios in the diesel generator basement and, initially, to assess fire induced flows in motor control centers (MCCs). The latter was not used in the final fire PRA. V&V of CFAST and FDS is documented in NUREG-1824, Volume 5 and Volume 7, respectively.

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

3.4.2.3.2 RAIs Pertaining to Fire Modeling in Support of the Brunswick Fire PRA

In letters dated May 15, 2013 (Reference 23), and January 14, 2014 (Reference 24), the NRC staff requested additional information. In letters dated June 28, 2013 (Reference 10); July 15, 2013 (Reference 11); July 31, 2013 (Reference 12); August 29, 2013 (Reference 13); September 30, 2013 (Reference 14); and February 28, 2014 (Reference 15), the licensee responded to these RAIs. The following paragraphs describe RAI responses related to the acceptability of the fire models used.

- In FM RAI 01.a (Reference 23), the NRC staff requested that the licensee describe the uncertainty associated with assuming a 15-minute fire brigade arrival time and the adverse effects of not meeting this assumption on the results of the fire PRA.

  In its response to FM RAI 01.a (Reference 10), the licensee explained that the fire brigade response time is used to determine when the door to the MCR is opened in a subset of the scenarios considered in the MCR abandonment time calculations. The licensee indicated that the 15-minute arrival time was derived based on historical fire brigade drill times and a 50 percent correction factor. The licensee contended that several aspects of the drills, such as compliance with security, administrative controls, radiological controls, and other barriers, cause drill times to be greater than response times during actual fire events. Hence, the licensee used a 50 percent reduction factor to modify the average time obtained from drill data.

  However, the NRC staff concluded that the use of a 50 percent reduction factor to modify fire drill data was not acceptable because of insufficient technical justification. The NRC staff determined that the calculated MCR abandonment times for the scenarios that are affected by the assumed fire brigade response time (i.e., the scenarios in which the door is assumed to open at 15 min) are not used in the fire PRA, and that the probability for control room abandonment is calculated based solely on the results of the CFAST calculations for the

scenarios with a closed door. Based on its review of the licensee's RAI response, the NRC staff also determined that the licensee used FAQ 08-0050, "Manual Non-Suppression Probability," (Reference 67) to calculate non-suppression probability, which indicates that the brigade response time for the MCR is not explicitly used in the PRA, and should not be used in future PRA updates, including post-transition plant change evaluations. The NRC staff concludes that the licensee's response to the RAI is acceptable, with the caveat described above regarding the reduction factor, because the calculated MCR abandonment times for the scenarios that are affected by the assumed fire response time are not applicable for the fire PRA.

- In FM RAI 01.b (Reference 23), the NRC staff requested that the licensee justify the use of the maximum compartment height in the MCR abandonment time calculations, as the compartment height varies in different parts of the MCR.

  In its response to FM RAI 01.b (Reference 12), the licensee indicated that the actual ceiling height was used for large portions of the control room and adjacent spaces, however, obstructions such as ducts and beams, that project below the ceiling in specific locations, were not considered in the CFAST model.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee provided adequate justification for using the maximum compartment height in the MCR abandonment time calculations.

- The NRC staff reviewed the sensitivity analysis in the MCR abandonment time study and found that poorly ventilated burning conditions result in a significant reduction in the abandonment times. In FM RAI 01.c (Reference 23), the NRC staff requested that the licensee explain why the base case that was used to determine the probability for MCR abandonment was not adjusted to include poorly ventilated burning.

  In its response to FM RAI 01.c (Reference 13), the licensee indicated that the sensitivity study is based on several conservative assumptions (e.g., the fire is placed on the floor, electrical cabinets contain multiple cable bundle fires, etc.). The licensee recalculated the abandonment time using the baseline input for fire location and HRR, and demonstrated that poorly ventilated conditions do not impact the fire PRA calculations.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that poorly ventilated cabinet fires do not adversely affect the probability for MCR abandonment.

- In FM RAI 01.e (Reference 23), the NRC staff requested that the licensee explain how the effect of the increased HRR from intervening combustibles (cable trays) on the ZOI was accounted for in the fire modeling calculations.

In its response to FM RAI 01.e (Reference 10), the licensee explained that additional screening using a bounding approach identified several fire scenarios involving propagating fires in cable trays that may result in a significant contribution to the risk. However, the licensee also indicated that fires in those areas were already recognized as significant contributors to risk, and that planned modifications along with existing solid cable tray bottoms in most locations are used to mitigate the risk.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that it adequately accounted for the effect of the increased HRR from intervening combustibles in the fire risk calculations.

- In FM RAI 01.f (Reference 23), the NRC staff requested that the licensee explain how wall and corner effects were accounted for in the HGL calculations.

  In its response to FM RAI 01.f (Reference 10), the licensee indicated that wall and corner effects were not explicitly accounted for in the HGL calculations. However, the licensee contended that the HGL calculations were conservative as the room size was reduced by 6 feet and the McCaffrey, Quintiere, and Harkleroad and Beyler models (NUREG-1934, Reference 54) over-estimate the temperature. The licensee demonstrated this conservatism by performing HGL calculations for a fire area that has ignition sources along the wall and corner.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that the HGL temperature calculations for fires with ignition sources along a wall or at a corner have sufficient conservatism to offset fire location effects.

- In FM RAI 01.g (Reference 23), the NRC staff requested that the licensee justify the use of a 3' x 3' footprint and zero elevation for transient combustibles.

  In its response to FM RAI 01.g (Reference 10), the licensee indicated that the transient floor area used in the model was based on walkdown data, with a 3' x 3' area as a minimum footprint, but that a larger footprint was used if needed. The licensee also stated that the floor area recorded during walkdowns was used for determining the floor area ratio (i.e., geometry factor), which apportions the transient fire ignition frequency assigned to specific fire scenarios in a physical analysis unit. The licensee further stated that transient fires are assumed to be at floor level to represent small trash receptacles and combustible materials brought into the zone on a temporary basis.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee provided adequate justification for using a 3' x 3' footprint and zero fire elevation for transient combustible fires.

- In FM RAI 01.h (Reference 23), the NRC staff requested that the licensee provide assurance that all cables that are not PRA targets and that all non-cable

intervening combustibles were considered and to explain how intervening combustibles were accounted for in the fire modeling analysis.

In its response to FM RAI 01.h (Reference 10), the licensee referred to two supporting documents related to fire PRA walkdown instructions and scoping fire modeling to explain how fires involving non-cable intervening combustibles and cables that are not targets in the fire PRA were accounted for.

The NRC staff determined that the only type of intervening combustibles that were considered were cable trays. In FM RAI 01.h.01 (Reference 24), the NRC staff requested that the licensee explain whether non-cable intervening combustibles were identified during the walkdowns and how the contribution from non-cable intervening combustibles was accounted for in the fire modeling and fire PRA analysis.

In its response to FM RAI 01.h.01 (Reference 15), the licensee explained that the walk down instructions considered the possibility of non-cable intervening combustibles and included provisions for recording any such observations. The licensee further stated that no non-cable intervening combustibles were observed during the actual walk downs, and therefore, cables were the only intervening combustibles that were accounted for in the analysis.

The NRC staff concludes that the licensee's responses to the RAIs are acceptable because the licensee demonstrated that there are no non-cable intervening combustibles that need to be considered in the fire modeling analysis and that have any potential to contribute to plant risk.

- In FM RAI 02.a (Reference 23), the NRC staff requested that the licensee describe the characterization of installed cabling in the power block, specifically with regard to critical damage thresholds.

  In the response to FM RAI 02.a (Reference 10), the licensee stated that slightly over 96 percent of the cables in the power block are characterized as thermoset, and that thermoset damage thresholds were, therefore, assigned to all cable targets in the fire PRA. The licensee provided a list of the types of cables (insulation and jacket material) that are considered in the fire PRA, and referred to NUREG/CR-6850, Appendix H, to further justify the use of thermoset damage criteria.

  The NRC staff concludes that, because the amount of thermoplastic cable is minimal, the credible fire scenarios in the power block that would involve a thermoplastic cable and significantly contribute to plant risk, or otherwise threaten SSD capability, are limited, and therefore, any difference in the extent of damage is negligible. Therefore, the NRC staff concludes that the licensee's approach to characterizing the damage threshold for cables in the power block, as described in the response to this RAI, is acceptable.

- In FM RAI 02.c (Reference 23), the NRC staff requested that the licensee describe the damage thresholds that were used for non-cable components such as valves, electrical cabinets, etc.

  In its response to FM RAI 02.c (Reference 12), the licensee indicated that the vulnerability of non-cable components is determined by the failure of the associated cable, and therefore, the damage threshold of thermoset cables for non-cable components was used. The licensee indicated that active components such as electrical cabinets, valves, pumps, etc. are assumed to fail consistent with the damage criteria for thermoset cables in accordance with the guidance provided in NUREG/CR-6850, Appendix H. The licensee also indicated that passive components such as check values and tanks were assumed not to be subject to damage by fire in accordance with the guidance provided in NUREG/CR-6850, Appendix H.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee adequately demonstrated why the damage threshold of thermoset cables was used for non-cable components such as valves and electrical cabinets.

- In FM RAI 02.d (Reference 23), the NRC staff requested that the licensee describe the damage criteria used for exposed, temperature-sensitive equipment.

  In its response to FM RAI 02.d (Reference 13), the licensee indicated that such devices are generally contained in cabinets, which provide protection. The licensee further stated that incipient detectors mitigate damage to electronic equipment in the MCBs.

- In FM RAI 02.d.01 (Reference 24), the NRC staff requested that the licensee provide clarification on how the Very Early Warning Fire Detection System (VEWFDS) was credited in the MCB analysis.

  In its response to FM RAI 02.d.01, the licensee stated that incipient detection in the main control boards was not credited in the fire PRA with mitigating the failure of any sensitive electronics. The licensee further stated that a quantitative sensitivity study was performed using the damage criteria as described in FAQ 13-0004 (Reference 94). The licensee further stated that the sensitivity study did not find any instances of sensitive electronics not contained in enclosures. The licensee indicated that the results of the sensitivity study were incorporated into the fire PRA.

  The NRC staff concludes the licensee's responses to the RAIs are acceptable because the licensee demonstrated that all sensitive electronics are contained in enclosures, and that potential damage to sensitive electronics is properly accounted for in the fire PRA.

3.4.2.3.3 Conclusion for Section 3.4.2.3

Based on the licensee's description in the LAR, as supplemented, of the process for performing fire modeling in support of the fire risk evaluations, and clarifications provided in response to the RAIs, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.4.3.3, is acceptable.

3.4.2.4 Conclusions Regarding Fire PRA Quality

Based on NUREG-0800, Section 19.2, Section III.2.2.4.1 (Reference 42), which summarizes the NRC staff's review of PRA quality required for an application, the NRC staff concludes that the licensee's PRA satisfies the guidance in RG 1.174, Section 2.3, and RG 1.205, Section 4.3 regarding the technical adequacy of the PRA used to support risk assessment for transitioning to NFPA 805.

The Fire PRA methods used to support the LAR were evaluated by the NRC staff in SE Section 3.4.2.2, and the NRC staff did not accept some of the methods proposed by the licensee. Fire PRA methods that are not accepted by the NRC are not considered alternatives to NRC accepted codes and standards. In each case, the licensee removed the method from the PRA or demonstrated that the method did not impact its ability to meet the risk acceptance guidelines of RG 1.174, and provided an implementation item in LAR Attachment S to replace the method prior to self-approval, as described in responses to PRA RAI 23 (Reference 17) and PRA RAI 24 (Reference 16).

The NRC staff concludes that the PRA approach, methods and data are acceptable, and therefore, that NFPA 805, Section 2.4.3.3, is satisfied for the request to transition to NFPA 805. The NRC staff based this conclusion on the findings that: (1) the PRA model for Brunswick meets the criteria in that, upon completion of the modifications discussed in LAR Attachment S, Table S-1, it would adequately represent the current, as-built, as-operated configuration, and is, therefore, capable of being adapted to model both the post-transition and compliant plant as needed; (2) the PRA models conform sufficiently to the applicable PRA standards for internal events and fires at an appropriate CC after considering the disposition of the peer review F&Os and NRC staff review findings; and (3) the fire modeling used to support the development of the Brunswick Fire PRA has been confirmed as appropriate and acceptable.

In PRA RAI 16 (Reference 23), the NRC staff requested that the licensee propose a method to verify the validity of the reported change-in-risk following completion of proposed modifications and to include a plan of action to notify the NRC if the risk guidelines are exceeded. In its response to PRA RAI16 (Reference 12), the licensee added Implementation Item 9 to LAR Attachment S, Table S-2, to verify the validity of the change in risk estimates after completion of modifications identified in LAR Attachment S, Table S-1. In PRA RAI 16.01 (Reference 25), the NRC staff requested an additional implementation item that would evaluate the change in risk estimates following completion of implementation items (e.g., procedure changes) in LAR Attachment S, Table S-2. In its response to PRA RAI 16.01 (Reference 16), the licensee added Implementation Item 14 to LAR Attachment S, Table S-2, that will provide for the review of the change in risk due to completion of other implementation items in LAR Attachment S, Table S-2. Both implementation items state that after completion of modifications, the as-built change-in-risk

will be evaluated using the post transition change process described in LAR Section 4.7.2 and any necessary actions will be identified. The NRC staff finds this acceptable because it provides confidence that the change-in-risk measured after completion of the as yet incomplete modification will meet the acceptance guidelines in RG 1.174.

Finally, based on the licensee's administrative controls to maintain the PRA models current and assure continued quality and using only qualified staff and contractors (as described in SE Section 3.8.3), the NRC staff concludes that the PRA maintenance process is adequate to maintain the quality of the Brunswick PRA to support self-approval of future RI changes to the fire protection program under the NFPA 805 license condition following completion of all implementation items described in the updated LAR Attachment S, Table S-2 (Reference 21).

3.4.3 Fire Risk Evaluations

For those fire areas for which the licensee used an RI approach to meet the nuclear safety performance criteria, the licensee used fire risk evaluations in accordance with NFPA 805, Section 4.2.4.2, and RG 1.205, Section C.2.2.4, to justify acceptable alternatives to complying with NFPA 805 deterministic criteria. The NRC staff reviewed the following information during its evaluation of Brunswick's fire risk evaluations: LAR Section 4.5.2, "Performance Based Approaches;" LAR Attachment C, "NEI 04-02 Table B-3 -Fire Area Transition;" and LAR Attachment W, "Fire PRA Insights," as well as associated supplemental information.

Plant configurations that did not meet the deterministic requirements of NFPA 805, Section 4.2.3.1, are considered variances from deterministic requirements (VFDRs) and are summarized in LAR Attachment C. Each VFDR was categorized into one of four types: (1) unprotected cables where cable separation did not meet the deterministic requirements of NFPA 805; (2) control room abandonment recovery actions; (3) area wide automatic suppression where the requirement of NFPA 805, Section 4.2.3.3(b) is not met for redundant success paths that are located in the same Fire Area; and (4) cable and equipment zone separation where the requirement of NFPA 805, Section 4.2.3.3(b) is not met due to lack of minimum separation of required cables and equipment in redundant success paths.

For each fire area containing VFDRs, a fire risk evaluation was performed in accordance with NFPA 805, Section 4.2.4.2 to estimate the change in risk associated with transition to NFPA 805. In LAR Attachment W, Section W.2.1, and in its response to PRA RAI 12 (Reference 12), the licensee described how the change-in-risk is determined. The change-in-risk is the difference between the post-transition plant PRA model (with VFDRs retained and with risk-reduction modifications) and the compliant plant PRA model (with the VFDRs removed and no risk-reduction modifications). The VFDRs were removed from the compliant plant PRA model by (1) removing cables from the fire damage sets for VFDRs associated with cables, (2) adding suppression to fire areas for VFDRs associated with the lack of automatic area wide suppression, (3) assuming one train is not failed by fire for VFDRs associated with lack of train separation (the risk effect of either train is equal), and (4) assuming that recovery actions mitigating VFDRs are 100 percent successful. The licensee also clarified in response to PRA RAI 12 (Reference 12) that method 4, "assuming recovery actions to be 100% successful," was only used for scenarios associated with MCR abandonment because only those scenarios included recovery actions. The total

change-in-risk associated with transition to NFPA 805 is the sum of the change-in-risk in all fire areas.

LAR Attachment S, Table S-1, identified modifications that will be completed before transitioning to NFPA 805. As part of its RAI response (Reference 19), the licensee provided an updated LAR Attachment S, Table S-1, and explained that modifications 6, 7, 10, and 11 were deleted because they were not needed to meet the risk acceptance guidelines. The licensee explained that modifications 6 and 7 would have minimal impact on the risk estimates, and the NRC staff found that modifications 10 and 11 were not credited in the Fire PRA. Because removal of each modification was evaluated and determined to have minimal effect on risk, the NRC staff concludes that the removal of these 4 modifications would not adversely impact the risk estimates and compliance with the acceptance guidelines in RG 1.174.

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 RG 1.205, Section 2.2.4.1, and FAQ 08-0054 (Reference 68). The NRC staff further concludes that the results of these calculations for each fire area demonstrate that the difference between the risk associated with implementation of the deterministic requirements and that of the VFDRs meets the risk acceptance criteria described in NFPA 805, Section 2.4.4.1.

3.4.4 Additional Risk Presented by Recovery Actions

The NRC staff reviewed LAR Attachment C, "NEI 04-02 Table B-3 -Fire Area Transition," LAR Attachment G, "Recovery Actions Transition," and LAR Attachment K, "Existing Licensing Action Transition," during its evaluation of the additional risk presented by the NFPA 805 recovery actions at Brunswick. SE Section 3.2.5 describes the identification and evaluation of recovery actions.

The licensee used the guidance in RG 1.205, Revision 1 (Reference 4) to define the primary control station and to identify recovery actions. Accordingly, any actions required to transfer control to, or operate equipment from, the primary control station, while required as part of the RI/PB fire protection program, were not considered recovery actions per the RG 1.205 guidance and in accordance with NFPA 805. Likewise, any OMAs required to be performed outside the control room and not at the primary control station were considered recovery actions.

The licensee identified the recovery actions in LAR Attachment G, Table G-1, and indicated which recovery actions were modeled in the Fire PRA and credited for risk reduction and which recovery actions are required for DID only. Operator actions performed at the primary control station following MCR abandonment are identified in LAR Attachment G, Table G-1, but as explained above, are not considered recovery actions. In LAR Attachment W, Section W.2.1 and in its response to PRA RAI 11 (Reference 12), the licensee explained that all actions credited in the Fire PRA for risk reduction are associated with abandonment of the MCR on loss of habitability. Accordingly, the only fire area for which the additional risk of recovery actions was determined and reported in LAR Attachment W, Tables W-4-1 and W-4-2, was Fire Area CB-23E (Control Room Extended).

In PRA RAI 14 (Reference 23), the NRC staff requested that the licensee clarify how the additional risk of recovery actions was calculated. In its response to PRA RAI 14 (Reference 12), the licensee explained that the additional risk of recovery actions was estimated by assuming total success of recovery actions for the compliant case, and using the nominal value for recovery actions in the post transition case. FAQ 07-0030, "Close-out of National Fire Protection Association Frequently Asked Questions 07-0030 on Establishing Recovery Actions," (Reference 61) guidance provides several alternatives on estimating the additional risk of recovery actions by eliminating the VFDRs or other more bounding estimates. Assuming total success of recovery actions for the compliant plant is only consistent with eliminating a VFDR as described in FAQ 07-0030 if all cutsets associated with a VFDR have recovery actions, which may not always be true. However, the proposed method does measure the additional risk of recovery action in the general sense that the risk of the action is removed from the compliant PRA model and retained in the post-transition PRA model. NFPA 805, Section 4.2.4, requires that, "the additional risk presented by [recovery actions] use shall be evaluated." The NRC staff finds that the licensee's approach of assuming total success of recovery actions for the compliant plant is acceptable because it provides an evaluation that provides a well-defined and meaningful measure of additional risk of recovery actions that can be compared to acceptance guidelines.

LAR Attachment G contained a review of all of the recovery actions for adverse impact and dispositioned each action. The updated LAR Attachment W (Reference 16) provides the additional risk from recovery actions as 9.06E-07/year for CDF and 9.06E-8/year for LERF. The estimates are the same values for Unit 1 and Unit 2. None of the recovery actions listed in LAR Attachment G, Table G-1, were found to have an adverse impact on the Fire PRA. All recovery actions listed in LAR Attachment G were evaluated against the feasibility criteria provided in NEI 04-02, FAQ 07-0030, and RG 1.205. As described in Implementation Item 14 in LAR Attachment S, Table S-2, the licensee will re-evaluate recovery actions as the plant modifications and procedures are finalized before the transition to NFPA 805 is completed.

The NRC staff concludes that the licensee's determination of the additional risk of recovery actions acceptable because the licensee evaluated the additional risk as required by NFPA 805, Section 4.2.4. Furthermore, the estimated risk values associated with recovery actions meets the acceptance guidelines in RG 1.174, and therefore, the NRC staff concludes that the additional risk of recovery actions meet the requirements of NFPA 805, Sections 4.2.4 and 2.4.4.1.

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

The licensee did not use any RI or PB alternatives to compliance with NFPA 805.

3.4.6 Cumulative Risk and Combined Changes

In LAR Attachment S, Table S-1, the licensee identified planned NFPA 805 transition modifications that decrease risk and for which the licensee took credit in the assessment of the cumulative risk impact of the transition to NFPA 805. The licensee included modifications which were not needed to bring the facility into compliance with the deterministic requirements of NFPA 805. The licensee credited the risk reduction from these modifications by including them

in the post-transition risk, but not the compliant plant risk. Similarly, the licensee incorporated the risk increase from the retained VFDRs by including them in the post-transition case but not in the compliant case. Therefore, the licensee's LAR to transition RI/PB fire protection program is a combined change request per RG 1.174, Revision 2, Section 2.1.1.

The total CDF and total LERF are estimated by adding the risk assessment results for internal events, fire, and external hazard events. The CDF and LERF results are summarized in SE Table 3.4.6-1. In PRA RAI 10 (Reference 23), the NRC staff requested that the licensee provide the basis for LAR Attachment W, Table W-1, internal events CDF and LERF and seismic CDF values. In its response to PRA RAI 10 (Reference 11), the licensee explained that several updates to the internal events PRA models have occurred since 2006 and provided a summary of those changes. The calculated CDF and LERF values decreased incrementally since 2006 down to values reported in LAR Attachment W, Table W-1. The licensee also explained that the LAR Attachment W, Table W-1, CDF value was developed from a seismic margins analysis using a 0.16 g earthquake. Given that the licensee has worked to improve its external events risk estimates and the fact that the total CDF and LERF reported in LAR Attachment W, Table W-1, are well below RG 1.174 guidelines for Region II changes of 1E-4/year and 1E-5/year CDF and LERF respectively, the NRC staff finds the reported values to be an acceptable basis for the combined change.

Table 3.4.6-1, CDF and LERF for Brunswick after Transition to NFPA 805

Hazard Group	Unit 1		Unit 2	
	CDF (/year)	LERF (/year)	CDF (/year)	LERF (/year)
Internal Events (including internal flooding)	7.9E-6	5.7E-7	7.9E-6	5.7E-7
External Flood	4.3E-7	3.0E-9	4.5E-7	3.0E-9
High Wind	5.8E-6	5.0E-8	6.5E-6	8.7E-8
Seismic	6.2E-8	8.7E-10	6.5E-8	8.7E-10
Fire Events	2.1E-5	4.4E-6	2.1E-5	4.1E-6
TOTAL	3.5E-5	5.0E-6	3.5E-5	4.8E-6

In PRA RAI 23 (Reference 25), the NRC staff requested that the licensee provide the results of an aggregate transition change in risk analysis that uses only PRA methods, approaches, and data acceptable to the NRC staff. In its response to PRA RAI 23 (Reference 17), the licensee made changes to the Fire PRA and provided updated risk information. The licensee also provided the ΔCDF and ΔLERF results estimated for each fire area at Brunswick that is not deterministically compliant, in accordance with NFPA 805, Section 4.2.3, "Deterministic Approach." The final total estimated change-in-risk associated with transition to NFPA 805 is reported as Δ CDF of 2.1E-06/year and Δ LERF of 1.1E-07/year for Unit 1, and Δ CDF of 3.9E-06/year and Δ LERF of 1.2E-07/year for Unit 2. The change-in-risk estimates credit the completed and planned modifications, recovery actions, and administrative controls that will be implemented to reduce risk as part of the transition to NFPA 805. From SE Table 3.4.6-1, the total CDF is less than 1E-4/year, and the total LERF is less than 1 E-5/year and reported change-in-risk is less that the applicable guidelines of 1E-5/year and 1E-6/year for CDF and

LERF, respectively. Each of the individual fire area changes-in-risk values for CDF and LERF reported also met the RG 1.174 acceptance guidelines.

Therefore, the NRC staff concludes that the risk associated with the proposed alternatives to compliance with the deterministic criteria of NFPA 805 is acceptable for the purpose of this application, in accordance with NFPA 805, Section 2.4.4.1. Additionally, the NRC staff concludes that the licensee satisfied RG 1.174, Section 2.4, and NUREG-0800, Section 19.2 regarding acceptable risk.

3.4.7 Uncertainty and Sensitivity Analyses

The licensee evaluated key sources of uncertainty and sensitivity in response to a number of RAIs and provided the following sensitivity study to address one key source.

The licensee used updated fire bin frequencies provided in NUREG/CR-6850, Supplement 1 (Reference 45) (i.e., FAQ-08-0048). The guidance in FAQ-08-0048 (Reference 66) states that a sensitivity study must be performed using the mean of the fire frequency bins contained in Section 6 of NUREG/CR-6850 for those bins with an alpha value less than or equal to one. The licensee provided this sensitivity in LAR Supplement (Reference 9), Section 4.8.3.3, and updated it in response to RAIs to remove methods that have not been accepted (Reference 14). However, the sensitivity study did not encompass all adjustments to the method used in the integrated analysis that was performed in response to PRA RAI 23 (Reference 17). Despite this fact, the sensitivity study indicated total increases in Δ CDF and Δ LERF and CDF and LERF values that were well within Region II of RG 1.174 acceptance guidelines. Inspection of the results indicate that even if the risk increases resulting from the integrated analysis in response to PRA RAI 23 were added to the risk increases from the sensitivity study, the total values would still remain within the RG 1.174 acceptance guidelines.

No other key source of uncertainty requiring a sensitivity analysis was identified by the licensee or by NRC staff.

3.4.8 Conclusion for Section 3.4

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

- The licensee's PRA used to perform the risk assessments in accordance with NFPA 805, Section 2.4.4 (plant change evaluations), and Section 4.2.4.2 (fire risk evaluations), is of sufficient quality to support the application to transition the Brunswick fire protection program to NFPA 805. The PRA approach, methods, tools and data are acceptable in accordance with NFPA 805, Section 2.4.3.3.

- The changes to the PRA model replaced certain approaches, data, and methods identified during the LAR review with ones that are acceptable to the NRC staff. The PRA models may be used to support post-transition self-approval of

changes because the identified acceptable methods will be used until they are replaced by other acceptable methods.

Implementation Item 13 in LAR Attachment S, Table S-2 (Reference 21), identifies four specific PRA approaches and methods that will be changed to acceptable approaches and methods before the use of the fire PRA to support self-approval of post transition changes. Additionally, Implementation Item 14 in LAR Attachment S, Table S-2, states that the licensee will re-evaluate recovery actions as the plant modifications and procedures are completed. The licensee will follow the post transition change impact review process if any of these changes result in changes in risk that would require NRC staff review and approval as specified in the proposed NFPA 805 license condition.

- The licensee's PRA maintenance process is adequate to support self-approval of future risk informed changes to the fire protection program following completion of the PRA related Implementation Items 9, 13, and 14, as described in LAR Attachment S, Table S-2.

- The transition process included a detailed review of fire protection DID and safety margins, as required by NFPA 805. The NRC staff finds the licensee's documentation on 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 guidance in RG 1.205, Revision 1, which provides an acceptable approach for meeting the requirements of 10 CFR 50.48(c).

- The changes in risk (i.e., ΔCDF and ΔLERF) associated with the proposed alternatives to compliance with the deterministic criteria of NFPA 805 (fire risk evaluations) are acceptable and the licensee has satisfied the guidance contained in RG 1.205, Revision 1, RG 1.174, and NUREG-0800, Section 19.2, regarding acceptable risk, which meets the requirements of NFPA 805.

- The risk associated with the use of recovery actions was evaluated and provided in accordance with the guidance in RG 1.205, Revision 1, and NFPA 805, Section 4.2.4. The NRC staff concludes that the additional risk associated with each NFPA 805 recovery action is acceptable because the risk for each fire area that relies on a recovery action is below the acceptance guidelines in RG 1.174, and therefore, meets the acceptance criteria in RG 1.205, Revision 1.

- The licensee did not utilize any RI or 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

NFPA 805 (Reference 3), Section 2.2.3, "Evaluating Performance Criteria," states that:

> To determine whether plant design will satisfy the appropriate performance criteria, an analysis shall be performed on a fire area basis, given the potential

fire exposures and damage thresholds, using either a deterministic or performance-based approach.

NFPA 805, Section 2.2.4, "Performance Criteria," states that:

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

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

3.5.1 Nuclear Safety Capability Assessment Results by Fire Area

NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment," states, in part, that:

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

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

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

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

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

NFPA 805, Section 4.2.2, "Selection of Approach," states that:

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

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

The NRC staff reviewed LAR (Reference 8), Section 4.2.4, "Fire Area Transition," LAR Section 4.8.1, "Results of the Fire Area Review," LAR Attachment C, "NEI 04-02 Table B3-Fire Area Transition," LAR Attachment G, "Recovery Actions Transition," LAR Attachment S, "Modifications and Implementation Items" and LAR Attachment W, "Fire PRA Insights," during its evaluation of the ability of each fire area to meet the nuclear safety performance criteria of NFPA 805.

Brunswick is a two-unit BWR with 43 individual fire areas, including the Yard, and each fire area is composed of one or more fire zones. Based on the information provided in the LAR, as supplemented, the licensee performed the nuclear safety capability assessment on a fire area basis. LAR Attachment C provides the results of these analyses on a fire area basis and also identifies the individual fire zones within the fire areas.

Table 3.5.1 of this SE identifies those fire areas that were analyzed using either the deterministic or performance-based approach in accordance with NFPA 805, Chapter 4, based on the information provided in LAR Attachment C Table C-1, "Fire Area Summary for the Brunswick Steam Electric Plant."

Table 3.5-1, Brunswick Fire Areas and Compliance Strategy

Fire Area	Area Description	NFPA 805 Compliance Basis Unit 1	NFPA 805 Compliance Basis Unit 2
AOG-1	Augmented Off-Gas Building	Deterministic	Deterministic
CB-1	Unit 1 Cable Access Ways	Performance Based	Performance Based
CB-2	Unit 2 Cable Access Ways	Performance Based	Performance Based
CB-7	Unit 1 Division I Battery Room	Deterministic	Deterministic
CB-8	Unit 1 Division II Battery Room	Deterministic	Deterministic
CB-9	Unit 2 Division I Battery Room	Deterministic	Deterministic

Fire Area	Area Description	NFPA 805 Compliance Basis Unit 1	NFPA 805 Compliance Basis Unit 2
CB-10	Unit 2 Division II Battery Room	Deterministic	Deterministic
CB-23E	Control Room Extended	Performance Based	Performance Based
DG-1	Diesel Generator Basement	Performance Based	Performance Based
DG-2	Diesel Cell 4	Deterministic	Deterministic
DG-3	Diesel Cell 3	Deterministic	Deterministic
DG-4	Diesel Cell 2	Performance Based	Deterministic
DG-5	Diesel Cell 1	Performance Based	Performance Based
DG-6	E5 Switchgear	Deterministic	Deterministic
DG-7	E6 Switchgear	Performance Based	Performance Based
DG-8	E7 Switchgear	Performance Based	Performance Based
DG-9	E8 Switchgear	Deterministic	Deterministic
DG-10	Loading Dock	Deterministic	Deterministic
DG-11	E1 Switchgear	Performance Based	Performance Based
DG-12	E2 Switchgear	Performance Based	Performance Based
DG-13	E3 Switchgear	Deterministic	Performance Based
DG-14	E4 Switchgear	Performance Based	Performance Based
DG-16E	Fan Room Extended	Performance Based	Performance Based
DG-19	Fuel Oil Tank Cell 1	Deterministic	Deterministic
DG-20	Fuel Oil Tank Cell 2	Deterministic	Deterministic
DG-21	Fuel Oil Tank Cell 3	Deterministic	Deterministic
DG-22	Fuel Oil Tank Cell 4	Deterministic	Deterministic
DUCTBANK	Ductbank	Performance Based	Performance Based
ISB	ISFSI Storage Building	Deterministic	Deterministic
MWT-1	Makeup Water Treatment	Deterministic	Deterministic
RB1-1	Unit 1 Reactor Building General Areas	Performance Based	Performance Based
RB1-6	Mini Steam Tunnel	Deterministic	Deterministic
RB2-1	Unit 2 Reactor Building	Performance Based	Performance Based
RB2-6	Mini Steam Tunnel	Deterministic	Deterministic
RMCSB	Radioactive Material Container Storage	Deterministic	Deterministic
RPDC1	Unit 1 Recirc Power Distribution Center	Deterministic	Deterministic
RPDC2	Unit 2 Recirc Power Distribution Center	Deterministic	Deterministic
RW-1	Radwaste Building	Deterministic	Deterministic
SERV	Service Building	Deterministic	Deterministic
STORES	Hot Shop/Stores/Warehouse Building	Deterministic	Deterministic
STORM	Storm Drain Rad Monitor Building	Deterministic	Deterministic
SW1-1	Service Water Building	Performance Based	Performance Based
TB1	Turbine Building General Areas	Performance Based	Performance Based
YARD	Yard	Deterministic	Deterministic

LAR Attachment C provides the results of these analyses on a fire area basis. For each fire area, the licensee documented the following:

- The approach used in accordance with NFPA 805 (i.e., the deterministic approach in accordance with NFPA 805, Section 4.2.3, or the performance-based approach in accordance with NFPA 805, Section 4.2.4);

- The SSCs required in order to meet the nuclear safety performance criteria;
- Fire detection and suppression systems required to meet the nuclear safety performance criteria;
- An evaluation of the effects of fire suppression activities on the ability to achieve the nuclear safety performance criteria; and;
- The disposition of each VFDR using either modifications (completed or committed) or the performance of a fire risk evaluation in accordance with NFPA 805, Section 4.2.4.2.

3.5.1.1 Fire Detection and Suppression Systems Required to Meet the Nuclear Safety Performance Criteria

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

The licensee performed a detailed analysis of fire protection features and identified the fire suppression and detection systems required to meet the nuclear safety performance criteria for each fire area. The revised LAR Attachment C, includes a "Required Regulatory Systems" table for each fire area and fire zone, which identifies if the fire suppression and detection systems installed in these areas are required to meet criteria for separation, DID, risk, licensing actions, or existing engineering equivalency evaluations (EEEEs).

The NRC staff reviewed each fire area in LAR Attachment C, to ensure the fire detection and suppression systems met the principles of DID in regard to the planned transition to NFPA 805. The NRC staff concludes that the licensee used appropriate methods to evaluate nuclear safety, DID, and safety margins, and adequately identified the fire detection and suppression systems, as well as the fire protection features required to meet the NFPA 805 nuclear safety performance criteria on a fire area basis.

3.5.1.2 Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

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

The licensee stated that damage to plant areas and equipment from the accumulation of water discharged from manual and automatic fire protection systems is minimized by the provision of floor drainage systems (where provided) and pedestals, is managed by the fire brigade, and/or controlled by curbs or sumps (where available) to mitigate the consequences of suppression discharge, and therefore, fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

The NRC staff concludes that the licensee's evaluation of the suppression effects on the nuclear safety performance criteria is acceptable because the licensee evaluated the fire suppression effects on meeting the nuclear safety performance criteria and determined that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

3.5.1.3 Licensing Actions

In LAR Section 4.2.3, "Licensing Action Transition," the licensee indicated that none of the licensing actions will be transitioned into the NFPA 805 fire protection program. In LAR Attachment K, the licensee described the licensing actions that are no longer necessary and will not be transitioned into the NFPA 805 fire protection program.

3.5.1.4 Existing Engineering Equivalency Evaluations (EEEEs)

In LAR Section 4.2.2, the licensee stated that the EEEEs that support compliance with NFPA 805, Chapter 4, were reviewed using the methodology contained in NEI 04-02 (Reference 7). The licensee further stated that the methodology for performing the EEEE review included the following determinations:

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

In LAR Section 4.2.2, the licensee stated that the guidance in RG 1.205 (Reference 4), Regulatory Position 2.3.2, and FAQ 08-0054 (Reference 68) was followed and that EEEEs that demonstrate that a fire protection system or feature is "adequate for the hazard" are summarized in the LAR as follows:

- If not requesting specific approval for an "adequate for the hazard" EEEE, then the EEEE is referenced where required and a brief description of the evaluated condition is provided.

- If requesting specific NRC approval for "adequate for the hazard" EEEEs, then the EEEE is referenced where required to demonstrate compliance and is included in Attachment L for NRC review and approval.

The licensee identified and summarized the EEEEs for each fire area in LAR Attachment C, as applicable. The licensee did not request the NRC staff to review and approve any of these EEEEs.

Based on the NRC staff's review of the licensee's methodology for review of EEEE's and identification of the applicable EEEEs in LAR Attachment C, the NRC staff concludes that the use of EEEEs is acceptable because it meets the requirements of NFPA 805, and the guidance provided in RG 1.205 and FAQ 08-0054.

3.5.1.5 Variances from Deterministic Requirements

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

- A fire risk evaluation determined that applicable risk, DID, and safety margin criteria were satisfied without further action.

- A fire risk evaluation determined that applicable risk, DID, and safety margin criteria were satisfied with a credited recovery action.

- A fire risk evaluation determined that applicable risk, DID, and safety margin criteria were satisfied with a DID RA.

In LAR Attachment C, the licensee stated that no VFDRs were resolved using modifications and that all modifications identified in LAR Attachment S were credited for either risk reduction or NFPA Code compliance, but not as resolutions to VFDRs.

In a letter dated May 15, 2013 (Reference 23), the NRC staff asked SSA RAI 15 and requested that the licensee provide an updated LAR Attachment C, Table B-3. In its response to SSA RAI 15 (Reference 13), the licensee provided, by fire area, a list of each VFDR identified and dispositions for each VFDR in LAR Attachment C, Table B-3, that reflected changes to LAR Attachments C and G, and addressed clarifications. The VFDRs listed by fire area, provided the component affected, the disposition of the VFDR, the performance goal affected, and the failure impact.

The four types of VFDRs at Brunswick are (1) Unprotected Cables, (2) Control Room Abandonment Recovery Actions, (3) Area Wide Automatic Suppression, and (4) Cable and Equipment Separation Zones. VFDRs can be dispositioned either through (1) a performance-based approach if the VFDR meets the acceptance criteria of the fire risk evaluation, (2) the development of a recovery action, or (3) the completion of a modification. The dispositions of the VDFRs at Brunswick are either through a performance-based approach or the development of a recovery action. No VFDRs were dispositioned through the completion of a modification.

In SSA RAI 15.02 (Reference 24), the NRC staff requested that the licensee provide VFDR lists for Fire Areas CB-1 and CB-2. In its response to SSA RAI 15.02 (Reference 15), the licensee provided the VFDR lists for Fire Areas CB-1 and CB-2 and provided the appropriate revisions to LAR Attachment C, Table B-3. The NRC staff concludes that the licensee's responses to the RAIs are acceptable because the licensee provided a listing of each VFDR along with its disposition.

For all fire areas where the licensee used the performance-based approach to meet the nuclear safety performance criteria, each VFDR and the associated disposition has been described in LAR Attachment C. Based on the NRC staff review of the VFDRs and associated resolutions as described in LAR Attachment C, as supplemented, the NRC staff concludes that the licensee's identification and resolution of the VFDRs is acceptable because the licensee demonstrated that each VFDR has been appropriately dispositioned.

3.5.1.6 Recovery Actions

LAR Attachment G (Reference 8), Tables G-1 (Unit 1) and G-2 (Unit 2), list the recovery actions identified in the resolution of VFDRs in LAR Attachment C for each fire area. The recovery actions identified include actions considered necessary to meet risk acceptance criteria, as well as actions, relied upon as DID (see SE Section 3.5.1.7 below), and actions taken at the PCS.

In SSA RAI 15.01 (Reference 24), the NRC staff requested clarification of certain recovery action feasibility walkdown results in the service water building. In its response to SSA RAI 15.01 (Reference 15), the licensee identified the need for an additional implementation item to update documentation for feasibility evaluations. The licensee stated that Implementation Item 12 is being added to LAR Attachment S, Table S-2, to update the manual action feasibility calculation and alternate shutdown procedures to incorporate final results of the feasibility evaluations. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee identified a required action that will incorporate the provisions of NFPA 805, Chapter 3, and included the action as an implementation item in LAR Attachment S, which is required by the proposed license condition.

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions," and Attachment G, "Recovery Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of recovery actions per NFPA 805. The details of the NRC staff review for recovery actions are described in SE Section 3.2.5, "Establishing Recovery Actions." The NRC staff's evaluation of the additional risk of recovery actions credited to meet the risk acceptance guidelines is provided in SE Section 3.4.4.

3.5.1.7 Recovery Actions Credited for Defense-in-Depth

The licensee stated in the LAR that recovery actions required for DID are not credited in the risk determination for the fire area, but are credited in the fire safety analysis. The nuclear safety performance goals, objectives, and criteria of NFPA 805 are met without these actions. These recovery actions are required for DID and are part of the RI/PB fire protection program, which necessitates that these actions be subject to a plant change evaluation if subsequently modified or removed.

The NRC staff reviewed LAR Section 4.2.1.3, "Establishing Recovery Actions," and Attachment G, "Recovery Actions Transition," to evaluate whether the licensee meets the associated requirements for the use of recovery actions per NFPA 805. The NRC staff's evaluation of the licensee's process for identifying recovery actions and assessing their feasibility is provided in SE Section 3.2.5, "Establishing Recovery Actions."

3.5.1.8 Plant Fire Barriers and Separations

With the exception of Electrical Raceway Fire Barrier Systems (ERFBS), passive fire protection features include the fire barriers used to form fire area boundaries (and barriers separating SSD trains) that were established in accordance with Brunswick's pre-NFPA 805 deterministic fire protection program. For the transition to NFPA 805, the licensee retained previously established fire area boundaries as part of the RI/PB fire protection program.

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

3.5.1.9 Electrical Raceway Fire Barrier Systems (ERFBS)

The licensee stated that the ERFBS used at Brunswick met the deterministic requirements of NFPA 805, Chapter 3. Each fire area using ERFBS is identified in LAR Attachment C, Table B-3. In fire areas with deterministic compliance, the ERFBS met the requirements of NFPA 805, Section 4.2.3. In fire areas with performance-based compliance, the ERFBS were analyzed using the performance-based approach in accordance with NFPA 805, Section 4.2.4. Each performance-based fire area utilizing ERFBS, as identified in LAR Attachment C, included a discussion of any variance from deterministic requirement (VFDR) analysis used to evaluate the acceptability of this feature.

In SSA RAI 2.01 (Reference 24), the NRC staff requested clarification of the ERFBS barrier definition and justification for why certain barriers are not treated as ERFBS, specifically regarding modification items 5 and 7 listed in LAR Attachment S, Table S-1. In its response to SSA RAI 2.01 (Reference 15), the licensee indicated that they had determined two modifications -- Table S-1, Items 5 and 7-- are required to reduce risk in the specific areas, and see, therefore, are needed to meet the requirements of NFPA 805, Chapter 3, Section 3.11.5, for any ERFBS installation that may be used. The licensee stated that Modification 5 will use

ERFBS and has been noted in the LAR Attachment C, Table B-3, "Required Regulatory Systems Table." In a letter dated August 15, 2014 (Reference 19), the licensee stated that modification 7, as described in LAR Attachment S, Table S-1, would not be completed because the risk increase for not completing the modification is negligible. The licensee added new Implementation Item 11 to LAR Attachment S, Table S-2, to update the program documentation as appropriate for the selected method of protection. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee identified required actions that will incorporate the provisions of NFPA 805, Chapter 3, and included the actions as a modification and implementation item in LAR Attachment S, which is required by the proposed license condition. The NRC staff also concludes that not completing modification 7 is acceptable because the licensee determined that there is only a negligible increase in risk if the modification is not completed.

The licensee stated that in both deterministic and performance-based fire areas, the installed ERFBS configurations met the testing requirements required by NFPA 805, Section 3.11.5 and that compliance is described in LAR Attachment A, Table B-1, Section 3.11.5.

3.5.1.10 Conclusion for Section 3.5.1

As documented in LAR Attachment C, for those fire areas that used a deterministic approach in accordance with NFPA 805, Section 4.2.3, the NRC staff concludes that each of the fire areas analyzed using the deterministic approach meets the associated criteria of NFPA 805, Section 4.2.3. This conclusion is based on:

- The licensee's documented compliance with NFPA 805, Section 4.2.3;
- The licensee's assertion that the success path will be free of fire damage without reliance on recovery actions;
- The licensee's assessment that the suppression systems in the fire area will have no impact on the ability to meet the nuclear safety performance criteria; and
- The licensee's appropriate determination of the automatic fire suppression and detection systems required to meet the nuclear safety performance criteria.

For those fire areas that used the performance-based approach in accordance with NFPA 805, Section 4.2.4, the NRC staff concludes that each fire area has been properly analyzed, and compliance with the NFPA 805 requirements demonstrated as follows:

- No exemptions from the pre-NFPA 805 fire protection licensing basis were transitioned to the NFPA 805 licensing basis.
- VFDRs were evaluated and either found to be acceptable based on an integrated assessment of risk, DID, and safety margins or recovery actions were identified and actions planned or implemented to address the issue.

- Recovery actions used to demonstrate the availability of a success path to achieve the nuclear safety performance criteria were evaluated and the additional risk of their use determined, reported, and found to be acceptable.

- The licensee's analysis appropriately identified the fire protection SSCs required to meet the nuclear safety performance criteria, including fire suppression and detection systems.

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

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

Accordingly, the NRC staff concludes that each fire area utilizing the deterministic or performance-based approach meets the applicable requirements of NFPA 805, Section 4.2.

3.5.2 Clarification of Prior NRC Approvals

As stated in LAR Attachment T, there are no elements of the current fire protection program for which NRC clarification is needed.

3.5.3 Fire Protection during Non-Power Operational Modes

NFPA 805, Section 1.1, "Scope," states the following:

> This standard specifies the minimum fire protection requirements for existing light water nuclear power plants during all phases of plant operation, including shutdown, degraded conditions, and decommissioning.

NFPA 805, Section 1.3.1, "Nuclear Safety Goal," states the following:

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

The NRC staff reviewed LAR Section 4.3, "Non-Power Operational Modes," and LAR Attachment D, "NEI 04-02 Non-Power Operational Modes Transition," to evaluate the licensee's treatment of potential fire impacts during non-power operations (NPOs). Brunswick used the process provided in NEI 04-02, Revision 2 (Reference 7), for demonstrating that the nuclear safety performance criteria are met for higher risk evolutions during NPO modes.

3.5.3.1 NPO Strategy and Plant Operating States

In LAR Section 4.3 and LAR Attachment D, the licensee stated that the process used to demonstrate that the nuclear safety performance criteria are met during NPO modes is consistent with the guidance contained in FAQ 07-0040 (Reference 64). As described in LAR Attachment D, the licensee's outage risk management and shutdown risk management procedures implement Brunswick's outage risk management strategy. The licensee stated that the process to demonstrate that the nuclear safety performance criteria are met during NPO modes involved the following steps:

- Reviewing the existing Outage Management Processes;
- Identifying Equipment/Cables:
  - Reviewing plant systems to determine success paths that support each of the DID Key Safety Functions (KSFs), and
  - Identifying cables required for the selected components and determined their routing;
- Performing Fire Area Assessments (identify pinch points – plant locations where a single fire may damage all success paths of a KSF); and
- Managing pinch-points associated with fire-induced vulnerabilities during the outage.

As described in the LAR, the licensee identified equipment and cables necessary to support the KSFs success paths. The operational modes and functional requirements for the systems and components were reviewed. The KSF success path equipment and cables were incorporated in the NPO database model. Following identification of KSF equipment and cables, the licensee performed analysis on a fire area basis to identify areas where redundant equipment and cables credited for a given KSF might fail due to fire damage (i.e., pinch-points). The licensee used a deterministic approach to identify these pinch-points and mitigated these pinch-points through the use of recovery actions and/or fire prevention/protection controls. In LAR Section 4.3.2, the licensee stated that fire modeling was not used to eliminate any pinch-points.

LAR Attachment D stated that Nuclear Generation Group Standard Procedures for shutdown risk management will be used to implement the plant's philosophy of outage risk management for cold shutdown and refueling modes, as well as for when the reactor is defueled. The licensee stated that this was developed to define the risk management program and provide supplemental site instructions based on design for various shutdown conditions. The licensee further stated that in addition to clarifying definitions and responsibilities, the procedure provides general expectations and specific guidance for maintaining each of the KSFs.

3.5.3.2 NPO Analysis Process

The licensee stated that its goal is to ensure that contingency plans are established when the plant is in a high risk evolution and it is possible to lose a KSF due to fire. LAR Section 4.3 discusses these additional controls and measures, however, the licensee further stated that

during low risk periods, normal risk management controls, as well as fire prevention/protection processes and procedures, will be used.

3.5.3.3 NPO Key Safety Functions (KSF) and SSCs Used to Achieve Performance

LAR Attachment D defines the KSFs, the process for establishing the success paths to achieve the KSFs, and the components required for the success paths. In SSA RAI 11.a (Reference 23), the NRC staff requested that the licensee provide a list of components that were not included in the at-power analysis or identified as having a different function for NPO. In its response to SSA RAI 11.a (Reference 12), the licensee provided a list of the required components. The licensee indicated in LAR Attachment D that after the list of KSFs was finalized, the required system components, their power supplies, and their required functional states were identified. The licensee further stated that since the equipment and power supplies needed to maintain NPO KSFs are similar to those required to safely shut down the plant while at power, the SSA was used as the basis for equipment selection. The licensee further stated that the list of NPO equipment was refined since some SSD components are not required for NPO, whereas other non-SSD components were identified for inclusion. The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee identified the components that were not included in the at-power analysis or identified as having a different function for NPO.

Pinch points refer to a particular location in an area where the damage from a single fire scenario could result in failure of multiple components or trains of a system such that the maximum detriment on that system's performance would be realized from the single fire scenario. Typically, this involves close vertical proximity of cables which support redundant components or trains of a system such that all such cables can be damaged by just one fire scenario.

In SSA RAI 11.b (Reference 23), the NRC staff requested that the licensee provide a list of the KSF pinch points by fire area that were identified in the NPO fire area reviews, including a summary level identification of unavailable paths in each fire area. In its response to SSA RAI 11.b (Reference 12), the licensee provided the list of unavailable paths and pinch points for each system component by fire area. A summary level list was provided by the licensee as follows:

> For the RHR System suction valves, discharge valves, and flow-path valves that can potentially fail during a fire event resulting in loss of shutdown cooling or inventory, the pinch point areas are:
>
> - Battery Rooms
> - DG Switchgear Rooms
> - DG Engine Rooms
> - Reactor Buildings
> - Drywell
> - Turbine Buildings

For the Service Water System boundary valves or pumps that can potentially fail during a fire event resulting in a loss of service water flow or pump operation, the pinch point areas are:

- Service Water Building
- Reactor Buildings
- Turbine Buildings

For the Electrical System, power supply, control power, batteries, and switchgear ventilation that can potentially be lost or limited due to a fire event, pinch point areas are:

- Reactor Buildings
- Turbine Buildings
- Service Water Building

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee provided a list of unavailable paths and pinch points for each system component by fire area.

In SSA RAI 11.d (Reference 23), the NRC staff requested that the licensee identify locations where KSFs are achieved using recovery actions or for where instrumentation not already included in the at-power analysis is needed to support recovery actions required to maintain safe and stable conditions. In its response to SSA RAI 11.d (Reference 12), the licensee stated that no locations exist at Brunswick where NPO associated KSFs are achieved solely through recovery actions. The licensee further stated that there are certain instances where recovery actions are identified as possibly being needed if a KSF path is not available as identified in the analysis. The licensee further stated that validation of these variable in-plant procedures uses a procedure review and validation process for feasibility evaluation of any actions. The licensee further stated in LAR Attachment D (Revision 0) that recovery actions proposed for NPO are summarized as follows:

- Service Water System valves (e.g., 1(2)-SW-V4, -V36, -V102, -V105, or -V106) may require manual operation;

- Service Water Pump may require replacement of a control circuit fuse prior to operation;

- Residual Heat Removal Service Water Valve 1(2)-E11-PDV-F068A(B) may require manual operation;

- Residual Heat Removal System valves (e.g., 1(2)-E11-F015A(B) or -F017A(B)) or boundary valves (e.g., 1(2)-B32-F023A(B)), may require manual actions;

- Emergency buses may lose DC control power and may require transfer to the alternate source of control power;

- Ventilation to 4k (volt) V and 480V switchgear rooms may be lost and may require manual actions to open doors to supply outside air;

- Prior to loading Emergency Diesel Generator #1 or #3, the corresponding emergency bus load breakers and offsite power supply breaker may need to be locally tripped; and

- Motor Control Center 1XDB may lose its normal power supply and may require manual actions to transfer to the alternate power supply.

The licensee further stated that these were not yet finalized in procedures but that they will be through LAR Attachment S, Table S-2, Implementation Item 1.

The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee demonstrated that there are no locations that exist where NPO associated KSFs are achieved solely through recovery actions, but there are certain instances where recovery actions are identified as possibly being needed if a KSF path is not available. The licensee proposed several recovery actions for NPO and developed an action to include them in procedures. The NRC staff concludes that the licensee's action is acceptable because the action that will incorporate the provisions of NFPA 805 is included as an implementation item in LAR Attachment S, which is required by the proposed license condition.

Based on its review of the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee used acceptable methods consistent with the guidance provided in 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 Attachment S, Table S-2, which are required by the proposed license condition.

3.5.3.4 NPO Pinch Point Resolutions and Program Implementation

In LAR Section 4.3, the licensee discussed non-power operational modes and included a discussion of the process it used to demonstrate that the nuclear safety performance criteria are met during NPO modes. One of the steps in this process included the management of pinch-points associated with fire-induced vulnerabilities during an outage. In LAR Figure 4-6, the licensee depicted its process for managing pinch points. The licensee provided additional discussion regarding pinch points in LAR Attachment D, "NEI 04-02 Non-Power Operational Modes Transition."

In SSA RAI 11.c (Reference 23), the NRC staff requested that the licensee provide a description of any actions that are credited to minimize the impact of fire induced spurious actuations on power operated valves (e.g., air-operated valves and motor-operated valves) during NPO either as pre-fire plant configuring or as required during the fire response recovery. In its response to SSA RAI 11.c (Reference 12), the licensee stated that although the NPO analysis takes credit for the at-power pre-fire rack out of the RHR shutdown cooling suction valves prior to

establishing shutdown cooling, the proposed NPO procedures will not prescribe any actions to specifically mitigate the impact of fire induced spurious actuations, either pre or post fire. The licensee further stated that in order to allow for unforeseen shutdown alignments for which spurious operation of a valve may cause the loss of a key safety function, it may be allowable to de-energize one or more valves, subject to an engineering evaluation, as part of the strategy, to mitigate the potential loss of that key safety function. The NRC staff concludes that the licensee's response to the RAI is acceptable because even though the licensee does not have any specific actions to mitigate the impact of fire induced spurious actuations during NPO, the licensee indicated that there is a process to de-energize one or more valves through the use of an engineering evaluation as part of a strategy to mitigate the potential loss of a key safety function.

The analyses performed by the licensee identified vulnerabilities associated with loss of KSFs during the high risk evolutions. In LAR Attachment S, Table S-2, Implementation Item 1, the licensee has identified an action to incorporate the NPO analysis into plant procedures and documentation. The NRC staff concludes that the identified action is acceptable because it will incorporate the provisions of NFPA 805 and because the licensee included the action as an implementation item in LAR Attachment S, which is required by the proposed license condition.

NFPA 805 requires that the nuclear safety performance criteria be met during any operational mode or condition, including NPO. As described above, the licensee has performed engineering analyses to demonstrate that it meets this requirement. The licensee has:

- Identified the KSFs required to support the nuclear safety performance criteria during NPOs;
- Identified the plant operating states where further analysis is necessary during NPOs;
- Identified the SSCs required to meet the KSFs during the plant operating states analyzed;
- Identified the location of these SSCs and their associated cables;
- Performed analyses on a fire area basis to identify pinch points were one or more KSFs could be lost as a direct result of fire-induced damage; and
- Planned/implemented modifications to appropriate procedures in order to employ a fire protection strategy for reducing risk at these pinch points during high-risk evolutions (HREs).

Accordingly, based on the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee has provided reasonable assurance that the nuclear safety performance criteria are met during NPO modes and HREs.

3.5.4 Conclusion for Section 3.5

The NRC staff reviewed the licensee's RI/PB fire protection program, as described in the LAR and its supplements, to evaluate the nuclear safety capability assessment results. The licensee used a combination of the deterministic approach and the performance-based approach, in accordance with NFPA 805, Sections 4.2.3 and 4.2.4.

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

- Fire suppression effects were evaluated and found to have no adverse impact on the ability to achieve and maintain the nuclear safety performance criteria for each fire area; and

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

Accordingly, the NRC staff concludes that there is reasonable assurance that each fire area utilizing the deterministic approach does so in accordance with NFPA 805, Section 4.2.3.

For those fire areas that utilized a performance-based approach, the NRC staff confirmed the following:

- Fire suppression effects were evaluated and found to have no adverse impact on the ability to achieve and maintain the nuclear safety performance criteria for each fire area;

- All VFDRs were evaluated using the fire risk evaluation performance-based method (in accordance with NFPA 805, Section 4.2.4.2) to address risk impact, DID, and safety margin, and found to be acceptable;

- All recovery actions 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; and

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

Accordingly, the NRC staff concludes that there is reasonable assurance that each fire area utilizing the performance-based approach does so in accordance with NFPA 805, Section 4.2.4.

The NRC staff concludes that the licensee's analysis and outage management process during NPO provides reasonable assurance that the nuclear safety performance criteria will be met during NPO modes and HREs, and that the licensee used methods consistent with the guidance provided in FAQ 07-0040 and RG 1.205. The NRC staff also concludes that no recovery actions are required during NPO modes and that the licensee's overall approach for fire protection during NPO modes is acceptable.

3.6 Radioactive Release Performance Criteria

3.6.1 Introduction

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

NFPA 805, Section 1.3.2, "Radioactive Release Goal," states that:

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

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

> Either of the following objectives shall be met during all operational modes and plant configurations:
>
> (1) Containment integrity is capable of being maintained.
>
> (2) The source term is capable of being limited.

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

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

The NRC staff has endorsed (with certain exceptions), the methodology given in NEI 04-02 (Reference 7) as providing methods acceptable to the staff for establishing a fire protection program consistent with NFPA 805 (Reference 3), and 10 CFR 50.48(c) in RG 1.205 (Reference 4). Using these methods, the licensee has assessed the capability of the current fire protection program to meet the NFPA 805 performance criteria as contained in NEI 04-02 and FAQ 09-0056 (Reference 69). The results of the licensee's assessment are documented in the LAR (Reference 8).

The NRC reviewed the assessment provided in the LAR in order to determine if the existing FPP, with its planned modifications, would meet the radioactive release performance criteria requirements of an RI/PB fire protection program, in accordance with 10 CFR 50.48(a) and (c), using the guidance in RG 1.205 and NUREG-0800, Section 9.5.1.2 (Reference 40). The NRC staff also performed an audit of the licensee's evaluation to determine whether the Brunswick FPP and its planned modifications would be capable of meeting the NFPA 805 radioactive release goals, objectives, and performance criteria. The results of the NRC staff evaluation and audit are provided below.

3.6.2 Scope of Review

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

3.6.3 Identification of Plant Areas Containing Radioactive Materials

The licensee performed a screening of plant fire areas to determine where there was a potential for generating radioactive effluents during firefighting operations. The results of the screening review are documented in LAR Attachment E, Table E-1, "Radioactive Release Compartment Review." The fire areas where there was no possibility of radioactive materials being present were identified and eliminated from further review. Each fire area that had the potential for generation of radioactive effluents created by firefighting activities was identified (screened in) for further evaluation.

The screened in areas included those areas where most of the radioactive materials were present such as in the reactor building, the turbine building, and other miscellaneous buildings. The review found that these areas had adequate engineered controls for containment of liquid and gaseous effluent. The licensee's review identified the existing engineering controls that were present and sufficient to contain gaseous and liquid effluent. These engineering controls credited for containment of gaseous and liquid effluent are identified and documented in LAR Attachment E. The NRC staff concludes that the identified engineering controls were adequate because they provided sufficient capacity to contain the gaseous and liquid firefighting effluents.

The licensee's review also identified other plant areas such as the Radwaste and Augmented Off-Gas Buildings where there were some engineered controls for containment of effluents (e.g., drains and sumps for containment of liquid effluent from fire suppression activities and routing the exhaust to a monitored pathway). For these areas, administrative controls were credited for containment of potential radioactive effluent.

The licensee's review also identified other plant areas where radioactive materials were present where there we no engineered controls for containment of effluents (e.g., Yard-RCA) area. For these areas, the potential for radioactive release and radiation exposure to members of the public was evaluated in a quantitative assessment.

The NRC staff concludes that the licensee's screening of plant areas and identification of the potentially affected areas is an adequate assessment because the review incorporated all plant areas and identified potentially affected areas with and without engineering controls, in accordance with the guidance in NEI 04-02, as endorsed by RG 1.205.

3.6.4 Fire Pre-Plans

The licensee reviewed the existing fire protection program to determine if it is adequate to ensure that gaseous and liquid radioactive effluents generated as a direct result of fire suppression activities would be contained and monitored before release to unrestricted areas. The results of the licensee's review are documented in LAR Attachment E, Table E-1. This review included the following steps:

- Identification of applicable documentation, including the fire pre-plan, procedures and drawings.
- Review of current documentation to identify whether the current documents discuss the containment and monitoring of potential contamination involving fire suppression activities.
- Review of engineering controls for gaseous effluents to determine in which areas the gaseous effluents are contained.
- Review of engineering controls for liquid effluents to determine in which areas the liquid effluents are contained.
- An identification of those documents needing revision such as to provide for monitoring and containment of fire suppression agents and radioactive release.

The NRC staff concludes that the licensee's evaluation is adequate because the review was performed in accordance with the guidance in NEI 04-02, Appendix G, as endorsed by RG 1.205.

3.6.5 Gaseous Effluent Controls

In areas where engineering controls exist for containment, filtering, and monitoring of gaseous effluent, the engineering controls were determined to provide adequate containment because the effluent was either contained, or filtered to remove radioactive materials and subsequently monitored prior to discharge.

For plant areas where the effectiveness of the installed engineering controls was not adequate to contain the gaseous effluent, the licensee will modify the fire protection program to establish compensatory actions such that the Fire Brigade and Radiation Protection personnel will manually establish containment and perform monitoring of radioactive effluent. The licensee included an action to "implement the results of the radioactive release analysis and update fire pre-plans," in LAR Attachment S, Table S-2, Implementation Item 2. The NRC staff considers this action acceptable because it will result in compliance with NFPA 805 and because it is required by the proposed license condition. For these plant areas, the NRC staff concludes that NFPA 805 radioactive release goals, objectives, and performance criteria will be met because the radioactive release will be manually contained to within acceptable limits by a combination of the installed engineered controls and compensatory actions taken by the Fire Brigade and Radiation Protection personnel.

In other plant areas where adequate engineered controls were not sufficient for containment of radioactive effluents, the licensee evaluated potential releases using a bounding quantitative analysis of the potential impacts of radioactive gaseous effluents during a fire. The bounding case was the Yard area where the largest single radioactive source was in a Sealand storage container fully loaded with radioactive waste. The licensee performed a dose assessment based on the type of radionuclides that are stored, and the maximum amount of radioactive material that was allowed to be stored, and then assumed to be released during a fire.

During the NRC's audit of the licensee's LAR, the NRC reviewed the licensee's bounding analysis described in Calculation BNP-RAD-027, "Evaluation of Dose Consequences from a Fire in a Radwaste Container in an Outside Area." The review determined the assessment to be adequate because the bounding assessment was based on conservative assumptions and analytical methods that are recognized by the NRC as acceptable methods. The results of the analysis demonstrate that the maximum offsite dose at the Exclusion Area Boundary was less than the 10 CFR 20 dose limits for members of the public.

The NRC staff concludes that the licensee's approach is acceptable because it adequately quantified and limited the maximum amount of radioactive material that can be safely stored during a fire event. The staff also concludes that the public dose from radioactive material released as a gaseous effluent during a fire would not exceed of the radiological release performance criteria of NFPA 805 and the public dose limits of 10 CFR 20.

3.6.6 Liquid Effluent Controls

The licensee identified those areas where sufficient engineering controls exist for containment of liquid effluent (e.g., floor drains routed to sumps and tanks). The NRC staff reviewed those engineering controls and determined that those controls provided adequate containment because the effluent is collected, stored, processed and monitored prior to discharge.

The licensee's review also identified those areas where there were not sufficient engineered controls to adequately contain potential liquid effluents released during firefighting activities, such as in the outside Yard area. Release of liquids to any unrestricted area due to the direct effects of fire suppression activities will be controlled by a fire preplan, and the action to update the fire pre-plans with the results of the radioactive release analysis is included in LAR Attachment S, Table S-2, Implementation Item 2. The NRC staff considers the action acceptable because it will incorporate the provisions of NFPA 805 in the licensee's fire protection program and it is included as an implementation item in LAR Attachment S, which is required by the proposed license condition.

Based on the engineering controls that provide sufficient capacity to contain liquid effluents and the proposed administrative controls to control release of liquids to any unrestricted area, the NRC staff concludes that NFPA 805 radioactive release goals, objectives, and performance criteria will be met because the radioactive release will be manually contained to within acceptable limits by a combination of the installed engineered controls and compensatory actions taken by the Fire Brigade and Radiation Protection personnel.

3.6.7 Fire Brigade Training Materials

The licensee reviewed the fire brigade training materials to ensure they were consistent with the pre-fire plans in terms of containment and monitoring of potentially contaminated smoke and fire suppression water. The review is documented in LAR Attachment E, Table E-1. Each training module and lesson plan was evaluated, and those training materials needing improvements were identified and documented. The revision of the training materials to address radioactive release requirements of NFPA 805 is described in LAR Attachment S, Table S-2, Implementation Item 2. The NRC staff considers the action acceptable because it will result in compliance with NFPA 805 and because it is required by the proposed license condition. The training material revisions will describe the actions the Fire Brigade will take to ensure that the engineering controls are intact and capable of supporting containment of gaseous and liquid effluents, including the use of manual mitigation methods when necessary.

The NRC staff reviewed the licensee's evaluation of training materials and concludes that upon completion of the implementation item, the training materials will be adequate to instruct the Brunswick staff to implement the fire protection program because plant staff will be informed and capable of taking actions to limit the public dose to within the radiological release performance criteria of NFPA 805.

3.6.8 Conclusion for Section 3.6

Based on the information provided in the LAR, the licensee's use of fire pre-plans, the results of the NRC staff's evaluation of the identified engineered controls used to contain potential releases, and the development and implementation of fire brigade response and training procedures, the NRC staff concludes that the licensee's RI/PB fire protection program provides reasonable assurance that radiation releases to any unrestricted area resulting from the direct effects of fire suppression activities are as low as reasonably achievable and are not likely to exceed the radiological release performance criteria of NFPA 805 and the radiological dose limits in 10 CFR Part 20. The NRC staff concludes that upon completion of the implementation item, the licensee's fire protection program will comply 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 <u>NFPA 805 Monitoring Program</u>

3.7.1 Monitoring Program

For this SE section, the following requirements from NFPA 805 (Reference 3), Section 2.6, are applicable to the NRC staff's review of the LAR:

NFPA 805, Section 2.6, "Monitoring," 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.

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

> Acceptable levels of availability, reliability, and performance shall be established.

NFPA 805, Section 2.6.2, "Monitoring Availability, Reliability, and Performance," states that:

> Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience.

NFPA 805, Section 2.6.3, "Corrective Action," states that:

> If the established levels of availability, reliability, or performance are not met, appropriate corrective actions to return to the established levels shall be implemented. Monitoring shall be continued to ensure that the corrective actions are effective.

The NRC staff reviewed LAR (Reference 8), Section 4.6, "Monitoring Program," that the licensee developed to monitor availability, reliability, and performance of Brunswick fire protection program systems and features after transition to NFPA 805. The focus of the NRC staff review was on critical elements related to the monitoring program, including the selection of fire protection program 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 fire protection program implementation, which the NRC staff concludes is acceptable.

The licensee stated that it will develop an NFPA 805 monitoring program consistent with FAQ 10-0059 (Reference 70). Development of the monitoring program will include a review of existing surveillance, inspection, testing, compensatory measures, and oversight processes for adequacy. The review will examine adequacy of the scope of SSCs within the existing plant programs, performance criteria for availability and reliability of SSCs, and the adequacy of the plant corrective action program. The monitoring program will incorporate phases for scoping, screening using risk criteria, risk target value determination, and monitoring implementation. The scope of the program will include fire protection systems and features, nuclear safety capability assessment equipment, SSCs relied upon to meet radioactive release criteria, and fire protection programmatic elements.

As described above, NFPA 805, Section 2.6, requires that a monitoring program be established in order to ensure that the availability and reliability of fire protection systems and features are maintained, as well as to assess the overall effectiveness of the fire protection program in meeting the performance criteria. Monitoring should ensure that the assumptions in the associated engineering analysis remain valid.

Based on the information provided in the LAR, as supplemented, the NRC staff concludes that the licensee's NFPA 805 monitoring program 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 scope of SSCs to be monitored;
- Utilizes an acceptable screening process for determining the SSCs to be included in the program;
- Establishes availability, reliability, and performance criteria for the SSCs being monitored; and
- Requires corrective actions when SSC availability, reliability, or performance criteria targets are exceeded to bring performance back within the required range.

However, since the final values for availability and reliability, as well as the performance criteria for the SSCs being monitored, have not been established for the NFPA 805 monitoring program as of the date of this SE, completion of the licensee's NFPA 805 monitoring program is an implementation item, addressed in LAR Attachment S, Table S-2, Implementation Item 3.

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 LAR Attachment S, Table S-2, within 180 days after NRC approval (or 60 days after the startup from an refueling outage if the 180th day occurs during an outage), which is prior to completion of the modifications to achieve full compliance with 10 CFR 50.58(c) (i.e., by the startup of the second refueling outage for each unit after issuance of the SE).

3.7.2 Conclusion for Section 3.7

The NRC staff reviewed the licensee's RI/PB fire protection program and concludes that the licensee's approach for meeting the requirements of NFPA 805, Sections 2.6, regarding the monitoring program is acceptable and that there is reasonable assurance that the licensee will develop a monitoring program that meets the requirements specified in Sections 2.6.1, 2.6.2, and 2.6.3 of NFPA 805, because the licensee identified an action to revise plant documents to monitor and trend the fire protection program, and included that action as an implementation item, which is required by the proposed license condition.

3.8 Program Documentation, Configuration Control, and Quality Assurance

For this SE section, the requirements from NFPA 805 (Reference 3), Section 2.7, "Program Documentation, Configuration Control and Quality," are applicable to the NRC staff's review of the LAR in regard to the appropriate content, configuration control, and quality of the documentation used to support the Brunswick fire protection program transition to NFPA 805.

NFPA 805, Section 2.7.1.1, "General," states that:

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

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

> A fire protection program design basis document shall be established based on those documents, analyses, engineering evaluations, calculations, and so forth that define the fire protection design basis for the plant. As a minimum, this document shall include fire hazards identification and nuclear safety capability assessment, on a fire area basis, for all fire areas that could affect the nuclear safety or radioactive release performance criteria defined in Chapter 1.

NFPA 805, Section 2.7.1.3, "Supporting Documentation," states that:

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

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

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

NFPA 805, Section 2.7.2.2, "Supporting Documentation," states that:

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

NFPA 805, Section 2.7.3.1, "Review," states that:

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

NFPA 805, Section 2.7.3.2, "Verification and Validations" 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.

3.8.1 Documentation

The NRC staff reviewed LAR (Reference 8), Section 4.7.1, "Compliance with Documentation Requirements in Section 2.7.1 of NFPA 805," to evaluate the Brunswick fire protection program design basis document and supporting documentation.

The Brunswick fire protection program design basis is a compilation of multiple documents (i.e., fire safety analyses, calculations, engineering evaluations, nuclear safety capability assessments, etc.), databases, and drawings which are identified in LAR Figure 4-9, "NFPA 805 Transition – Planned Post-Transition Documentation Relationships." The licensee stated that the analyses conducted to support the NFPA 805 transition were performed in accordance with Brunswick processes which meet or exceed the requirements for documentation outlined in NFPA 805, Section 2.7.1.

Specifically, the licensee stated that they have documented analyses to support compliance with 10 CFR 50.48(c) and that the analyses are performed in accordance with their 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 analysis. The licensee further stated that the analyses will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy.

Based on the LAR description, as supplemented, of the content of the fire protection program design basis and supporting documentation, and taking into account the licensee's plans to maintain this documentation throughout the life of the plant, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Sections 2.7.1.1, 2.7.1.2, and 2.7.1.3, regarding adequate development and maintenance of the fire protection program design basis documentation, is acceptable.

3.8.2 Configuration Control

The NRC staff reviewed LAR Section 4.7.2, "Compliance with Configuration Control Requirements in Section 2.7.2 and 2.2.9 of NFPA 805," in order to evaluate the Brunswick configuration control process for the new NFPA 805 fire protection program.

To support the many other technical, engineering and licensing programs, the licensee has existing configuration control processes and procedures for establishing, revising, or utilizing program documentation. Accordingly, the licensee is integrating the new fire protection program design basis and supporting documentation into these existing configuration control processes and procedures. These processes and procedures require that all plant changes be reviewed for potential impact on the various Brunswick licensing programs, including the fire protection program.

The LAR stated that the configuration control process includes provisions for appropriate design, engineering reviews and approvals, and that approved analyses are considered controlled documents available through the document control system. The LAR also stated that analyses based on the PRA program, which includes the fire risk evaluations, are issued as formal analyses subject to these same configuration control processes, and are additionally subjected to the PRA peer review process specified in the ASME/ANS PRA standard (Reference 38).

Configuration control of the existing fire protection program during the transition period is maintained by the change evaluation process, as defined in existing configuration management and configuration control procedures. LAR Attachment S, Table S-2, includes Implementation Item 4 to update configuration control procedures to reflect the new NFPA 805 licensing bases requirements. The NRC staff concludes that this action is acceptable because it will incorporate the provisions of NFPA 805 in the licensee's fire protection program and because it is included as an implementation item in LAR Attachment S, Table S-2, which is required by the proposed license condition.

The NRC staff reviewed the licensee's process for updating and maintaining the fire PRA in order to reflect plant changes made after completion of the transition to NFPA 805 in SE Section 3.4.

Based on the description of the Brunswick configuration control process, which indicates that the new fire protection program design basis and supporting documentation will be controlled, plant changes will be reviewed for impact on the fire protection program, and upon completion of the implementation item, the NRC staff concludes that there is reasonable assurance that the requirements of NFPA 805, Sections 2.7.2.1 and 2.7.2.2, will be met.

3.8.3 Quality

The NRC staff reviewed LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," to evaluate the quality of the engineering analyses used to support transition of the fire protection program to NFPA 805 based on the requirements outlined above. The individual sections of this SE provide the NRC staff's evaluation of the application of the NFPA 805 quality requirements to the licensee's fire protection program, as appropriate.

3.8.3.1 Review

NFPA 805, Section 2.7.3.1 requires that each analysis, calculation, or evaluation performed be independently reviewed. The licensee stated that its procedures require independent review of analyses, calculations, and evaluations, including those performed in support of compliance with 10 CFR 50.48(c). The LAR also stated that the transition to NFPA 805 was independently reviewed, and that analyses, calculations, and evaluations to be performed post-transition will be independently reviewed, as required by existing procedures.

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

3.8.3.2 Verification and Validation

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

3.8.3.2.1 General

NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," Volumes 1-7 (Reference 50), documents the V&V of five selected fire models commonly used to support applications of RI/PB fire protection at nuclear power plants. The seven volumes of this NUREG series report provide technical documentation concerning the predictive capabilities of a specific set of fire dynamics calculation tools and fire phenomenological models that may be used for the analysis of fire hazards in postulated nuclear power plant scenarios. When used within the limitations of the fire models and considering the identified uncertainties, these models may be employed to demonstrate compliance with the requirements of 10 CFR 50.48(c) as part of an approved performance-based approach in accordance with NFPA 805, Chapter 4.

Accordingly, for those fire modeling elements performed by the licensee using the V&V applications contained in NUREG-1824 to support the transition to NFPA 805, the NRC

concludes that the use of these models is acceptable, provided that the intended application is within the appropriate limitations, as identified in NUREG-1824.

In LAR Section 4.5.2, the licensee also identified the use of several empirical correlations that are not addressed in NUREG-1824. 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 guidance.

Table 3.8-1, "V&V Basis for Fire Modeling Correlations Used at Brunswick," in SE Attachment A, and Table 3.8-2, "V&V Basis for Other Fire Models and Related Calculations Used at Brunswick," in SE Attachment B, identify these empirical correlations and algebraic models, respectively, as well as a staff disposition for each.

The NRC staff concludes that the theoretical bases of the models and empirical correlations used in the fire modeling calculations that were not addressed in NUREG-1824 were identified and described in authoritative publications (References 86-91). SE Table 3.8-1 summarizes the additional fire models, and the NRC staff's evaluation of the acceptability of each.

The fire modeling employed by the licensee in the development of the fire risk evaluations used empirical correlations that provide bounding solutions for the ZOI, and conservative input parameters, which produced conservative results for the fire modeling analysis. See SE Section 3.4.2.3 for further discussion of the licensee's fire modeling method.

3.8.3.2.2 Discussion of RAIs

In letters dated May 15, 2013 (Reference 23) and January 14, 2014 (Reference 24), the NRC staff requested additional information. In letters dated June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; and February 28, 2014 (References 10-15, respectively), the licensee responded to these RAIs.

- In FM RAIs 03.a and 03.b (Reference 23), the NRC staff requested that the licensee update LAR Attachment J to include all correlations and fire models used in the fire modeling. The NRC staff also requested that the licensee demonstrate that the correlations and models were used within the validated range of input parameters, or to provide justification for their use outside the validated range.

  In its response to FM RAIs 03(a) and 03(b) (Reference 13), the licensee provided a revised and expanded version of LAR Attachment J. The licensee also demonstrated that the correlations and models were generally used within their validated range, and provided justification for special cases where these methods have been used outside the validated range.

  The NRC staff concludes that the licensee's response to the RAIs is acceptable because the licensee has either used correlations and fire models within their validated range, or provided technical justification for their application outside the validated range.

3.8.3.2.3 Post-Transition

The licensee also stated that it will revise the appropriate processes and procedures to include NFPA 805 quality requirements for use during the performance of post-transition fire protection program changes, including those for V&V. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for V&V is identified in LAR Attachment S, Table S-2, as Implementation Item 4, and the NRC staff considers this action acceptable because it will achieve compliance with NFPA 805, and because it is required by the proposed license condition.

3.8.3.2.4 Conclusion for Section 3.8.3.2

Based on the licensee's description of the Brunswick process for V&V of calculational models and numerical methods and their commitment for continued use post-transition, the NRC staff concludes that the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.2, is acceptable because the models are consistent with approved uses in NRC guidance or other authoritative publications and the licensee has identified an action that will result in compliance with NFPA 805 and that action is required by the proposed license condition.

The NRC staff concludes that the licensee's approach provides reasonable assurance that the fire modeling used in the development of the fire scenarios for the Brunswick fire PRA is appropriate, and thus acceptable for use in transition to NFPA 805 because the V&V of the empirical correlations used by the licensee are consistent with either NUREG-1824 or the Society of Fire Protection Engineers (SFPE) Handbook of Fire Protection Engineering.

3.8.3.3 Limitations of Use

NFPA 805, Section 2.7.3.3 requires that only acceptable engineering methods and numerical models be used for applications to the extent that these methods have been subject to V&V; and that they are applied within the scope, limitations, and assumptions prescribed for that method. The LAR stated that the engineering methods and numerical models used in support of the transition to NFPA 805 were subject to the limitations of use outlined in NFPA 805, Section 2.7.3.3, and that the engineering methods and numerical models used post-transition will be subject to these same limitations of use. As an example, in LAR Section 4.5.1.2, "Fire PRA," the licensee stated that the Generic Fire Modeling Treatments (GFMTs) approach as used to support the NFPA 805 transition, was applied within its documented limitations.

3.8.3.3.1 General

The NRC staff assessed the acceptability of each empirical correlation or other fire model in terms of the limits of its use. SE Table 3.8-1 in SE Attachment A, and Table 3.8-2 in SE Attachment B, summarize the fire models used, how each was applied in the Brunswick fire risk evaluations, the V&V basis for each, and the NRC staff evaluation for each.

3.8.3.3.2 Discussion of RAIs

In letters dated May 15, 2013 (Reference 23) and January 14, 2014 (Reference 24), the NRC staff requested additional information. In letters dated June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; and February 28, 2014 (References 10-15), the licensee responded to these RAIs.

- In FM RAI 04 (Reference 23), the NRC staff requested that the licensee explain how it ensured that the GMFTs, FDS, and other correlations and fire models were applied within their limits of applicability.

  In its response to FM RAI 04 (Reference 13), the licensee provided documentation describing the V&V of fire models used that also included the technical justification for the use of fire models in the cases where they were applied outside their range of applicability.

  The NRC staff concludes that licensee's response to the RAI is acceptable because the licensee either used correlations and fire models within their range of applicability, or provided adequate justification for the use of the methods outside their range of applicability.

3.8.3.3.3 Post-Transition

The licensee also stated that it will revise the appropriate processes and procedures to include the NFPA 805 quality requirements for use during the performance of post-transition fire protection program changes, including those for limitations of use. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements for limitations of use is identified in LAR Attachment S, Table S-2, as Implementation Item 4 and the NRC staff considers this action acceptable because it will achieve compliance with NFPA 805, and because the action is required by the proposed license condition.

3.8.3.3.4 Conclusion for Section 3.8.3.3

Based on the licensee's statements that the fire models used to support development of the fire risk evaluations were used within their limitations, and the description of the Brunswick 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 because the models are consistent with approved uses in NRC guidance or other authoritative publications, and the licensee has identified an action that will result in compliance with NFPA 805 and that action is required by the proposed license condition.

3.8.3.4 Qualification of Users

NFPA 805, Section 2.7.3.4 requires that personnel performing engineering analyses and applying numerical methods (e.g., fire modeling) be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire

protection, and power plant operations. The licensee's procedures require that cognizant personnel who use and apply engineering analyses and numerical models be competent in the field of application and experienced in the application of the methods, including those personnel performing analyses in support of compliance with 10 CFR 50.48(c).

Specifically, these requirements are being addressed through the implementation of an engineering qualification process at Brunswick. The licensee has developed procedures that require that cognizant personnel who use and apply engineering analyses and numerical models be competent in the field of application and experienced in the application of the methods, including those personnel performing analyses in support of compliance with 10 CFR 50.48(c). These requirements are being addressed through the implementation of an engineering qualification process. Brunswick has developed qualification or training requirements for personnel performing engineering analyses and numerical methods.

The NRC staff requested that the licensee provide additional information pertaining to qualifications of the personnel who supported the Brunswick fire modeling. Applicable RAIs and responses are discussed below:

- In FM RAI 05.a (Reference 23), the NRC staff requested that the licensee explain what constitutes the necessary qualifications for the personnel performing and reviewing fire modeling analyses.

  In its response to FM RAI 05.a (Reference 10), the licensee explained that qualification to perform and review fire modeling analyses is based on successful completion of specific parts of the licensee's qualification program, completing specific reading assignments of project instructions, and meeting specific education requirements

- In FM RAI 05.b (Reference 23), the NRC staff requested that the licensee describe the process and procedures for ensuring that the personnel performing and reviewing fire modeling analyses have the necessary qualifications.

  In its response to FM RAI 05.b (Reference 10), the licensee explained that in the case of contractors, the contractor's quality assurance process ensures that the personnel performing the fire modeling are qualified and trained. The licensee further described a number of internal programs that are designed to provide the minimum training necessary to perform fire modeling as required in Section 2.7.3.4 of NFPA 805. The licensee further stated that project instructions are available to identify and document required training and mentoring and that during and following transition, the existing engineering staff will continue to be knowledgeable in fire modeling techniques and the credentials of new fire modeling personnel will be reviewed to ensure they have the necessary qualifications.

- In FM RAI 05.c (Reference 23), the NRC staff requested that the licensee explain the communication process between the fire modeling analysts and PRA personnel and any measures taken to ensure that the fire modeling was

performed adequately and will continue to be performed adequately during post-transition.

In its response to FM RAI 05.c (Reference 12), the licensee stated that throughout the NFPA 805 transition process, the fire protection engineers who conducted the fire modeling and the PRA engineers maintained frequent communications and worked closely together. The licensee provided a number of specific examples to illustrate this cooperation, and explained that the same process will be used during implementation and post-transition.

The NRC staff concludes that the licensee's responses to the RAIs are acceptable because the licensee demonstrated that competent and experienced personnel developed the fire risk evaluations, including the supporting fire modeling calculations and additional documentation for models and empirical correlations not identified in previous NRC-approved V&V documents.

Further, LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805 Fire Protection Quality," states, in part, that:

> Post-transition, for personnel performing fire modeling or Fire PRA development and evaluation, CP&L [Carolina Power & Light] has developed and maintains qualification requirements for individuals assigned various tasks. Position-Specific Guides have been developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per the requirements of NFPA 805, Section 2.7.3.4 to perform assigned work....

In addition, based on the licensee's description of the procedures for ensuring personnel who use and apply engineering analyses and numerical methods are competent and experienced, the NRC staff concludes that the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.4, is acceptable.

3.8.3.5 Uncertainty Analysis

NFPA 805, Section 2.7.3.5, requires that an uncertainty analysis be performed to provide reasonable assurance that the performance criteria have been met. (Note: 10 CFR 50.48(c)(2)(iv) states that an uncertainty analysis performed in accordance with NFPA 805, Section 2.7.3.5, is not required to support calculations used in conjunction with a deterministic approach.) The licensee stated that an uncertainty analysis was performed for the analyses used in support of the transition to NFPA 805, and that an uncertainty analysis will be performed for post-transition analyses.

3.8.3.5.1 General

The industry consensus standard for PRA development (i.e., the ASME/ANS PRA standard) (Reference 37), includes requirements to address uncertainty. Accordingly, the licensee addressed uncertainty as a part of the development of the Brunswick fire PRA. The NRC staff's evaluation of the licensee's treatment of these uncertainties is discussed in SE Section 3.4.7.

NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (Reference 51), discusses three types of uncertainty associated with fire modeling calculations as follows:

(1) Parameter Uncertainty: Input parameters are often chosen from statistical distributions or estimated from generic reference data. In either case, the uncertainty of these input parameters affects the uncertainty of the results of the fire modeling analysis.

(2) Model Uncertainty: Idealizations of physical phenomena lead to simplifying assumptions in the formulation of the model equations. In addition, the numerical solution of equations that have no analytical solution can lead to inexact results. Model uncertainty is estimated via the processes of V&V. An extensive discussion of quantifying model uncertainty can be found in NUREG-1934, "Nuclear Power Plant Fire Modeling Application Guide (NPP FIRE MAG)," (Reference 54).

(3) Completeness Uncertainty: This refers to the fact that a model is not a complete description of the phenomena it is designed to simulate. Some consider this a form of model uncertainty because most fire models neglect certain physical phenomena that are not considered important for a given application. Completeness uncertainty is addressed by the description of the algorithms found in the model documentation. It is addressed, indirectly, by the same process used to address the Model Uncertainty.

3.8.3.5.2 Discussion of RAIs

In letters dated May 15, 2013 (Reference 23) and January 14, 2014 (Reference 24), the NRC staff requested additional information. In letters dated June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; and February 28, 2014 (References 10-15) the licensee responded to these RAIs.

- In FM RAI 06 (Reference 23), the NRC staff requested that the licensee describe how the input parameter uncertainty, model uncertainty and completeness uncertainty were accounted for in the fire modeling analysis.

  In its response to FM RAI 06 (Reference 13), the licensee indicated that input parameter uncertainty was addressed by using conservative inputs, or by assessing the effect of varying inputs in sensitivity cases. The licensee explained that model uncertainty was addressed by using models within their validated range, or by providing justification for their use outside the validated range. The licensee addressed completeness uncertainty by using a conservative approach in the fire PRA and by using models specifically designed for a particular application.

  The NRC staff concludes that the licensee's response to the RAI is acceptable because the licensee adequately accounted for parametric, model, and

completeness uncertainty in the fire modeling analysis performed in support of the NFPA 805 transition.

3.8.3.5.3 Post-Transition

The licensee stated that it will revise the appropriate processes and procedures to include the NFPA 805 quality requirements for use during the performance of post-transition fire protection program changes, including those regarding uncertainty analysis. Revision of the applicable post-transition processes and procedures to include NFPA 805 requirements regarding uncertainty analysis is identified in LAR Attachment S, Table S-2, as Implementation Item 4, and the NRC staff considers this action acceptable because it will achieve compliance with NFPA 805, and because it is required by the proposed license condition.

3.8.3.5.4 Conclusion for Section 3.8.3.5

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

3.8.3.6 Conclusion for Section 3.8.3

Based on the above discussions, the NRC staff concludes that subject to completion of the implementation items, the RI/PB fire protection quality assurance program meets each of the requirements of NFPA 805, Section 2.7.3, which include conducting independent reviews, performing V&V, limiting the application of acceptable methods and models to within prescribed boundaries, ensuring that personnel applying acceptable methods and models are qualified, and performing uncertainty analyses.

3.8.4 Fire Protection Quality Assurance Program

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

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The licensee established its fire protection quality assurance (QA) program in accordance with the guidelines of NUREG-0800, Section 9.5.1, BTP CMEB Position C.4, "Quality Assurance Program" (Reference 78). The guidance in NEI 04-02, Appendix C (Reference 7) recommends that the LAR include a description of how the existing fire protection quality assurance program will be transitioned to the new NFPA 805 RI/PB fire protection program.

The LAR stated that the fire protection QA program is included within and implemented by the Brunswick nuclear QA program, although certain aspects of that program are not applicable to the fire protection program. Further, the licensee stated that it will maintain the existing fire protection quality assurance program. In LAR Attachment S, Table S-3, Implementation Item 4, the licensee included actions to revise the appropriate processes and procedures to include the

NFPA 805 quality requirements for use during the performance of post-transition fire protection program changes, including those for verification and validation, limitations of use, and uncertainty analysis. The NRC staff considers those actions acceptable because the licensee will incorporate the provisions of NFPA 805, Section 2.7.3, in the fire protection program and because they are required by the proposed license condition.

Based on its review, the NRC staff concludes that the licensee's fire protection QA program is acceptable because the existing program includes the fire protection systems that are required by NFPA 805, Chapter 4, and because the changes to the fire protection QA program will incorporate the provisions of NFPA 805, Section 2.7.3, in the licensee's fire protection program.

3.8.5 Conclusion for Section 3.8

The NRC staff reviewed the licensee's RI/PB fire protection program as described in the LAR, as supplemented, to evaluate the NFPA 805 program documentation content, the associated configuration control process, and the appropriate QA requirements. The NRC staff concludes that subject to completion of the implementation items described in LAR Attachment S, the licensee's approach for meeting the requirements specified in Section 2.7 of NFPA 805 is acceptable.

4.0 FIRE PROTECTION LICENSE CONDITION

The licensee proposed an FPP license condition regarding transition to an RI/PB FPP under NFPA 805, in accordance with 10 CFR 50.48(c)(3)(i). The new license condition adopts the guidelines of the standard fire protection license condition promulgated in RG 1.205, Revision 1, Regulatory Position C.3.1, as issued on December 18, 2009 (74 FR 67253). Plant-specific changes were made to the sample license condition; however, the proposed plant-specific FPP license condition is consistent with the standard fire protection license condition, incorporates all of the relevant features of the transition to NFPA 805 at Brunswick and is, therefore, acceptable.

The following license condition is included in the revised license for Brunswick, and will replace Renewed Facility Operating License No. DPR-71 Paragraph 2.B.(6) and Renewed Facility Operating License No. DPR-62, Paragraph 2.B.(6).

> Fire Protection
>
> Duke Energy Progress, Inc. 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 license amendment request dated September 25, 2012; as supplemented by letters dated December 17, 2012; June 28, 2013; July 15, 2013; July 31, 2013; August 29, 2013; September 30, 2013; February 28, 2014; March 14, 2014; April 10, 2014; June 26, 2014; August 15, 2014; August 29, 2014; November 20, 2014; and December 18, 2014, and as approved in the safety evaluation dated January 28, 2015. 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 Brunswick. 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 $1x10^{-7}$/year (yr) for CDF and less than $1x10^{-8}$/yr for LERF. The proposed change must also be consistent with the defense-in-depth philosophy and must maintain sufficient safety margins. The change may be implemented following completion of the plant change evaluation.

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

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

Prior NRC review and approval is not required for changes to the NFPA 805, Chapter 3 fundamental fire protection program elements and design requirements for which an engineering evaluation demonstrates that the alternative to the Chapter 3 element is functionally equivalent or adequate for the hazard. The licensee may use an engineering evaluation to demonstrate that a change to an NFPA 805 Chapter 3 element is functionally equivalent to the corresponding technical requirement. A qualified fire protection engineer shall perform the engineering evaluation

and conclude that the change has not affected the functionality of the component, system, procedure, or physical arrangement, using a relevant technical requirement or standard.

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

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

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

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

Prior NRC review and approval is not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC safety evaluation dated January 28, 2015, to determine that certain fire protection program changes meet the minimal criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

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

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

2. The licensee shall implement the modifications to its facility, as described in Table S-1, "Plant Modifications Committed," of Duke letter BSEP 14-0122, dated November 20, 2014, to complete the 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 safety evaluation. The licensee shall maintain appropriate compensatory measures in place until completion of these modifications.

3. The licensee shall complete all implementation items, except item 9, listed in LAR Attachment S, Table S-2, "Implementation Items," of Duke letter BSEP 14-0122, dated November 20, 2014, within 180 days after NRC approval unless the 180th day falls within an outage window, then in that case completion of the implementation items, except item 9, shall occur no later than 60 days after startup from that particular outage. The licensee shall complete implementation LAR Attachment S, Table S-2, Item 9, within 180 days after the startup of the second refueling outage for each unit after issuance of the safety evaluation.

5.0 SUMMARY

The NRC staff reviewed the licensee's application, as supplemented by various letters, to transition to an RI/PB FPP in accordance with the requirements established by NFPA 805. The NRC staff concludes that the applicant's approach, methods, and data are acceptable to establish, implement and maintain an RI/PB FPP in accordance with 10 CFR 50.48(c).

Implementation of the RI/PB fire protection program in accordance with 10 CFR 50.48(c) will include the application of a new fire protection license condition. The new license condition includes a list of implementation items that must be completed in order to support the conclusions made in this SE, as well as an established date by which full compliance with 10 CFR 50.48(c) will be achieved. Before the licensee is able to fully implement the transition to an FPP based on NFPA 805 and apply the new fire protection license condition, to its full extent, the implementation items must be completed within the timeframe specified.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the North Carolina official was notified on January 23, 2015, of the proposed issuance of the amendment. The state official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no

significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on August 13, 2013 (78 FR 49300). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations; and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

9.0 REFERENCES

1. Branch Technical Position (BTP) APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants" (ADAMS Accession No. ML070660461).

2. Appendix A to BTP APCSB 9.5-1, "Guidelines for Fire Protection for Nuclear Power Plants Docketed Prior to July 1, 1976" (ADAMS Accession No. ML070660458).

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

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

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

8. Annacone, Michael J., Carolina Power and Light Company, letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket

Nos. 50-325, 50-324, License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," September 25, 2012 (ADAMS Accession No. ML12285A428).

9. Annacone, Michael J., Carolina Power and Light Company, letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Additional Information Supporting License Amendment Request to Adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants (2001 Edition)," December 17, 2012 (ADAMS Accession No. ML12362A284).

10. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," June 28, 2013 (ADAMS Accession No. ML13191B271).

11. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," July 15, 2013 (ADAMS Accession No. ML13205A016).

12. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," July 31, 2013 (ADAMS Accession No. ML13220B041).

13. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," August 29, 2013 (ADAMS Accession No. ML13246A276).

14. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," September 30, 2013 (ADAMS Accession No. ML13277A040).

15. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Second Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," February 28, 2014 (ADAMS Accession No. ML14073A168).

16. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," March 14, 2014 (ADAMS Accession No. ML14079A233).

17. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," April 10, 2014 (ADAMS Accession No. ML14118A105).

18. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," June 26, 2014 (ADAMS Accession No. ML14191A672).

19. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," August 15, 2014 (ADAMS Accession No. ML14234A326).

20. Hamrick, George T., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Response to Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," August 29, 2014 (ADAMS Accession No. ML14254A188).

21. Gideon, William R., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Additional Information Regarding License Amendment Request to Adopt Voluntary Risk Initiative National Fire Protection Association Standard 805 (NRC TAC Nos. ME9623 and ME9624)," November 20, 2014 (ADAMS Accession No. ML14337A724).

22. Gideon, William R., Duke Energy Progress, Inc., letter to U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325, 50-324, Additional Information Regarding License Amendment Request to Adopt Voluntary Risk Initiative National Fire Protection Association Standard 805 (NRC TAC Nos. ME9623 and ME9624)," December 18, 2014 (ADAMS Accession No. ML15002A260).

23. Gratton, Christopher, U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Units 1 and 2, Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," May 15, 2013 (ADAMS Accession No. ML13123A231).

24. Saba, Farideh, U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Units 1 and 2, Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," January 14, 2014 (ADAMS Accession No. ML13365A320).

25. Saba, Farideh, U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Units 1 and 2, Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," February 12, 2014 (ADAMS Accession No. ML14028A178).

26. Hon, Andrew, U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Units 1 and 2, Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," June 4, 2014 (ADAMS Accession No. ML14155A209).

27. Hon, Andrew, U.S. Nuclear Regulatory Commission, "Brunswick Steam Electric Plant, Units 1 and 2, Request for Additional Information Regarding Voluntary Risk Initiative National Fire Protection Association Standard 805," July 24, 2014 (ADAMS Accession No. ML14205A592).

28. Safety Evaluation, Completion of Facility Modifications for Fire Protection, November 22, 1977 (ADAMS Accession No. 4008005691).

29. Safety Evaluation, Reviews Fire Protection Plan and Discusses Close Proximity of Conduit and Trays, Three Hour Fire Barrier for Cable Tray Crossings and Physical Inspection, April 2, 1979 (ADAMS Accession No. 7904160105).

30. Safety Evaluation Supporting Amendments 28 and 51 to Licenses DPR-71 and DPR-62, Supplement 2 to Fire Protection Safety Evaluation, June 11, 1980 (ADAMS Accession No. 8006270039).

31. Safety Evaluation Supporting Exemption from 10 CFR 50, Appendix R, Subsections III.G and J, RE: Fire Protection and Emergency Lighting for Safe Shutdown Capability, December 30, 1986 (ADAMS Accession No. 8701020210).

32. Safety Evaluation Clarifying and Revising 15 of 19 areas Identified in 861230 Safety Evaluation, RE: 10 CFR 50, Appendix R, December 6, 1989 (ADAMS Accession No. 8912110146).

33. Safety Evaluation Concluding that the Interior Masonry Walls may be Downgraded to Non-Fire Rated, July 28, 1993 (ADAMS Accession No. 9308170310).

34. Safety Evaluation Supporting Amendments 169 and 200 to Licenses DPR-71 and DPR-62, February 10, 1994 (ADAMS Accession No. 9402180097).

35. Nuclear Energy Institute, NEI 00-01, "Guidance for Post-Fire Safe Shutdown Circuit Analysis," Revision 2, Nuclear Energy Institute (NEI), Washington, DC, June 5, 2009 (ADAMS Accession No. ML091770266).

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

37. U.S. Nuclear Regulatory Commission, Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk- Informed Activities," Revision 2, March 2009 (ADAMS Accession No. ML090410014).

38. American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) standard ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," February 2, 2009.

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

40. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, December 2009 (ADAMS Accession No. ML092590527).

41. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed License Amendment Requests After Initial Fuel Load," Revision 3, September 2012 (ADAMS Accession No. ML12193A107).

42. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, June 2007 (ADAMS Accession No. ML071700658).

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

44. U.S. Nuclear Regulatory Commission, NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology," September 2005 (ADAMS Accession No. ML052580118).

45. U.S. Nuclear Regulatory Commission, NUREG/CR-6850, Supplement 1, "Fire Probabilistic Risk Assessment Methods Enhancements," September 2010 (ADAMS Accession No. ML103090242).

46. Correia, R. 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 Accession No. ML13165A194).

47. U.S. Nuclear Regulatory Commission, NUREG/CR-6931, "Cable Response to Live Fire (CAROLFIRE)," Volumes 1, 2, and 3, April 2008 (ADAMS Accession Nos. ML081190230, ML081190248, and ML081190261).:

48. U.S. Nuclear Regulatory Commission, NUREG/CR-7100, "Direct Current Electrical Shorting in Response to Exposure Fire (DESIREE-Fire): Test Results," April 2012 (ADAMS Accession No. ML121600316).

49. U.S. Nuclear Regulatory Commission, NUREG-1805, "Fire Dynamics Tools (FDT^s): Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," December 2004 (ADAMS Accession No. ML043290075).

50. U.S. Nuclear Regulatory Commission, NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications," May 2007. Volume 1: Main Report, Volume 2: Experimental Uncertainty, Volume 3: Fire Dynamics Tools (FDT^s), Volume 4: Fire-Induced Vulnerability Evaluation (FIVE-Rev1), Volume 5: Consolidated Fire Growth and Smoke Transport Model (CFAST), Volume 6: MAGIC, and Volume 7: Fire Dynamics Simulator (ADAMS Accession Nos. ML071650546, ML071730305, ML071730493, ML071730499, ML071730527, ML071730504, ML071730543).

51. U.S. Nuclear Regulatory Commission, NUREG/CR-7010, Volume 1, "Cable Heat Release, Ignition, and Spread in Tray Installations during Fire (CHRISTIFIRE), Phase 1: Horizontal Trays," July 2012 (ADAMS Accession No. ML12213A056).

52. U.S. Nuclear Regulatory Commission, NUREG-1855, Volume 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," March 2009 (ADAMS Accession No. ML090970525).

53. U.S. Nuclear Regulatory Commission, NUREG-1921, "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines, Final Report" July 2012 (ADAMS Accession No. ML12216A104).

54. U.S. Nuclear Regulatory Commission, NUREG-1934, "Nuclear Power Plant Fire Modeling Analysis Guidelines (NPP FIRE MAG)," November 2012 (ADAMS Accession No. ML12314A165).

55. U.S. Nuclear Regulatory Commission, Generic Letter 2006-03, "Potentially Nonconforming Hemyc and MT Fire Barrier Configurations," April 10, 2006 (ADAMS Accession No. ML053620142).

56. National Fire Protection Association Standard 101 (NFPA 101), "Life Safety Code," Quincy, MA.

57. National Fire Protection Association Standard 20 (NFPA 20), "Standard for the Installation of Stationary Pumps for Fire Protection," 1999 Edition, Quincy, MA.

58. National Fire Protection Association Standard 14 (NFPA 14), "Standard for the Installation of Standpipe and Hose Systems," 1974 Edition, Quincy, MA.

59. National Fire Protection Association Standard 10 (NFPA 10), "Standard for Portable Fire Extinguishers," 1975 Edition, Quincy, MA.

60. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Frequently Asked Question 06-0022 on Electrical Cable Flame Propagation Tests," May 5, 2009 (ADAMS Accession No. ML091240278).

61. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Frequently Asked Question 07-0030 on Establishing Recovery Actions," February 4, 2011 (ADAMS Accession No. ML110070485).

62. 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).

63. 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).

64. 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).

65. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046

Incipient Fire Detection Systems," November 23, 2009 (ADAMS Accession No. ML093220426).

66. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0048 Revised Fire Ignition Frequencies," September 1, 2009 (ADAMS Accession No. ML092190457).

67. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question 08-0050 Manual Non-Suppression Probability," September 14, 2009 (ADAMS Accession No. ML092190555).

68. 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," February 17, 2011 (ADAMS Accession No. ML110140183).

69. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association 805 Frequently Asked Question 09-0056 on Radioactive Release Transition," January 14, 2011 (ADAMS Accession No. ML102920405).

70. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association Standard 805 Frequently Asked Question 10-0059: National Fire Protection 805 Monitoring Program," March 19, 2012 (ADAMS Accession No. ML120750108).

71. Safety Evaluation Supporting Amendments 23 and 47 to Licenses DPR-71 and DPR-62, April 6, 1979 (ADAMS Accession No. 7904260112).

72. Marion, Alexander, Nuclear Energy Institute letter to John Hannon, U.S. Nuclear Regulatory Commission, transmitting Revision 0 of NEI 02-03, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," June 17, 2003 (ADAMS Accession No. ML031780500).

73. Electric Power Research Institute (EPRI) Technical Report TR-1006756, "Fire Protection Equipment Surveillance Optimization and Maintenance Guide," Final Report, Palo Alto, CA, Final Report July 2003.

74. National Fire Protection Association Standard 30 (NFPA 30), "Flammable and Combustible Liquids Code," Quincy, MA.

75. National Fire Protection Association Standard 51B (NFPA 51B), "Standard for Fire Prevention During Welding, Cutting, and Other Hot Work," Quincy, MA.

76. National Fire Protection Association Standard 241 (NFPA 241), "Standard for Safeguarding Construction, Alteration, and Demolition Operations," Quincy, MA.

77. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Closure of National Fire Protection Association 805 Frequently Asked Question Number 06-0021," November 13, 2007 (ADAMS Accession No. ML072420306).

78. Scarola, James, Progress Energy Carolinas, Inc., letter to U.S. Nuclear Regulatory Commission, Brunswick Steam Electric Plant, Unit Nos. 1 and 2, Docket Nos. 50-325 and 50-324/License Nos. DPR-71 and DPR-62, "Response to Generic Letter 2006-03, Potentially Nonconforming Hemyc and MT Fire Barrier Configurations," June 6, 2006 (ADAMS Accession No. ML061640386).

79. U.S. Nuclear Regulatory Commission, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," Chapter 9.5.1, "Fire Protection Program," Revision 3, July 1981 (ADAMS Accession No. ML052350030).

80. National Fire Protection Association Standard 72 (NFPA 72), "National Fire Alarm and Signaling Code," Quincy, MA.

81. National Fire Protection Association Standard 76 (NFPA 76), "Standard for the Fire Protection of Telecommunications Facilities," Quincy, MA.

82. NEI 05-04, Process for Performing Internal Events PRA Peer Reviews Using the ASME/ANS PRA Standard, Revision 2, Nuclear Energy Institute (NEI), Washington, DC, November 2008.

83. U.S. Nuclear Regulatory Commission, "Record of Review, Brunswick Steam Electric Plant, Units 1 and 2, LAR Attachment U- Table U-1 Internal Events PRA Peer Review- Facts and Observations (F&Os)," August 29, 2014, and "Record of Review, Brunswick Steam Electric Plants, Units 1 and 2, LAR Attachment V- Tables V-1 and V-2 Fire PRA Peer Review- Facts and Observations (F&Os)," October 17, 2014 (ADAMS Accession Nos. ML14297A266 and ML14297A262, respectively).

84. NEI 07-12, Fire Probabilistic Risk Assessment (FPRA) Peer Review Process Guidelines, Revision 1, Nuclear Energy Institute (NEI), Washington, DC, June 2010.

85. U.S. Nuclear Regulatory Commission, FPRA FAQ 14-0009, "Treatment of Electrical Panels Greater than 440V," Revision E, October 27, 2014 (ADAMS Accession No. ML14303A218.

86. Klein, Alexander R., U.S. Nuclear Regulatory Commission, memorandum to file, "Close-out of National Fire Protection Association 805 Frequently Asked Question 12-0064 on Hot Work/Transient Fire Frequency Influence Factors," January 17, 2013 (ADAMS Accession No. ML12346A488).

87. License Event Report (LER) 14-004-00, "Fire Related Unanalyzed Condition that Could Impact Equipment Credited in Safe Shutdown Analysis" (ADAMS Accession No. ML14149A244).

88. Shokri, M., and Beyler, C., "Radiation from Large Pool Fires," Journal of Fire Protection Engineering, Vol. 1, pp. 141-149, 1989.

89. Mudan, K., "Thermal Radiation Hazards from Hydrocarbon Pool Fires," Progress in Energy and Combustion Science, Vol. 10, pp. 59-80, 1984.

90. Wakamatsu, T., Hasemi, Y., Kagiya, K., and Kamikawa, D., "Heating Mechanism of Unprotected Steel Beam Installed Beneath Ceiling and Exposed to a Localized Fire: Verification Using the Real-Scale Experiment and Effects of the Smoke Layer," Proceedings of the Seventh International Symposium on Fire Safety Science, International Association for Fire Safety Science, London, UK, 2003, pp. 1099-1110.

91. Gottuk, D. and White, D., "Liquid Fuel Fires," Chapter 2-15, The SFPE Handbook of Fire Protection Engineering, 3rd Edition, National Fire Protection Association, Quincy, MA, 2002.

92. Babrauskas, V., "Heat Release Rates," Chapter 3-1, The SFPE Handbook of Fire Protection Engineering, 4th Edition, National Fire Protection Association, Quincy, MA 2008.

93. Lattimer, B., "Heat Fluxes from Fires to Surfaces," Chapter 2-14, The SFPE Handbook of Fire Protection Engineering, 4th Edition, National Fire Protection Association, Quincy, MA, 2008.

94. Peacock, R.D., McGrattan, K., Klein, B., Jones, W.W., and Reneke, P.A., "CFAST-Consolidated Model of Fire Growth and Smoke Transport (Version 6) Software Development and Model Evaluation Guide," NIST Special Publication 1086, National Institute of Standards and Technology (NIST), Gaithersburg, MD, December 2008.

95. McDermott, R., McGrattan, K., Hostikka, S., Floyd, J. "Fire Dynamics Simulator (Version 5) Technical Reference Guide Volume 2: Verification FDS Version 5.4," NIST Special Publication 1018-5, National Institute of Standards and Technology, Gaithersburg (NIST), MD, September 2009.

96. NRC FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics," Revision 1, June 26, 2013 (ADAMS Accession No. ML13182A708).

97. Walton W. and Thomas, P., "Estimating Temperatures in Compartment Fires," Chapter 3–6, SFPE Handbook of Fire Protection Engineering, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.

98. McGrattan, K., McDermott, R, Prased, K., Hostikka, S., and Floyd, J., "Fire Dynamics Simulator (Version 5) Technical Reference Guide Volume 3: Validation FDS Version 5.4," NIST Special Publication 1018-5, National Institute of Standards and Technology, Gaithersburg, MD, September 2009.

99. Heskestad, G., "Fire Plumes, Flame Height, and Air Entrainment," Chapter 2–1 of the SFPE Handbook of Fire Protection Engineering, 4th Edition, National Fire Protection Association, Quincy, MA, 2008.

100. Beyler, C., "Fire Hazard Calculations for Large, Open Hydrocarbon Fires," Chapter 3–10 of the SFPE Handbook of Fire Protection Engineering, 4th Edition, National Fire Protection Association, Quincy, MA, 2008.

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Attachments:
A. Table 3.8-1 – V&V Basis for Fire Modeling Correlations Used at Brunswick
B. Table 3.8-2 – V&V Basis for Fire Model Calculations of Other Models Used at Brunswick
C. Abbreviations and Acronyms

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Heskestad flame height correlation	Development of ZOI tables for transient and liquid fuel spill fires in Generic Fire Modeling Treatments document, ZOI calculations for electrical cabinets	NUREG-1805 (Reference 49) NUREG-1824 (Reference 50) SFPE Handbook (Reference 99)	• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.
Heskestad plume temperature correlation	Development of ZOI tables for transient and liquid fuel spill fires in Generic Fire Modeling Treatments document, ZOI calculations for electrical cabinets	NUREG-1805 (Reference 49) NUREG-1824 (Reference 50) SFPE Handbook (Reference 99)	• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Modak point source radiation model	Development of ZOI tables for transient and liquid fuel spill fires in Generic Fire Modeling Treatments document, ZOI calculations for electrical cabinets	NUREG-1805 (Reference 49) NUREG-1824 (Reference 50) SFPE Handbook (Reference 100)	• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 11). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.
Shokri and Beyler flame radiation model	Development of ZOI tables in Generic Fire Modeling Treatments	Peer-reviewed journal article (Reference 88)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 11). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Mudan flame radiation model	Development of ZOI tables in Generic Fire Modeling Treatments	Peer-reviewed journal article (Reference 89)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.
Plume heat flux correlation by Wakamatsu et al.	Development of ZOI tables in Generic Fire Modeling Treatments document	Peer-reviewed conference paper (Reference 90)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Hydrocarbon spill fire size correlation	Development of ZOI tables in Generic Fire Modeling Treatments document	SFPE Handbook (Reference 91)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.
Method for estimating heat release rate of pool fires	Pool fire burning rate calculations based on fuel characteristics and pool diameter	SFPE Handbook (Reference 92)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Flame extension correlation	Development of ZOI tables in Generic Fire Modeling Treatments document	SFPE Handbook (Reference 93)	• The correlation is validated in an authoritative publication. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in the seminal publication describing the correlation. The licensee provided justification for cases where the correlation was used outside the validated range (Response to FM RAI 3.b, Reference 13). on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.
Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad)	The Hot Gas Layer (Method of McCaffrey, Quintiere, and Harkleroad) correlation was used to calculate the hot gas layer temperature for a room with natural ventilation.	NUREG-1805 (Reference 49) NUREG-1824 (Reference 50) SFPE Handbook (Reference 100)	• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment A: Table 3.8-1, V&V Basis for Fire Modeling Correlations Used at Brunswick

Correlation	Application at Brunswick	V&V Basis	NRC Staff Evaluation of Acceptability
Hot Gas Layer (Method of Beyler)	The Hot Gas Layer (Method of Beyler) correlation was used to calculate the hot gas layer temperature for a room with no ventilation.	NUREG-1805 (Reference 49) NUREG-1824 (Reference 50) SFPE Handbook (Reference 97)	• The correlation is validated in NUREG-1824 and an authoritative publication of the SFPE Handbook. • The licensee stated that in most cases, the correlation has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the correlation was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this correlation in the Brunswick application is acceptable.

Attachment B: Table 3.8-2, V&V Basis for Other Fire Models and Related Calculations Used at Brunswick

Calculation	Application at Brunswick	V&V Basic	NRC Staff Evaluation of Acceptability
CFAST (Version 6)	MCR abandonment times calculations	NUREG-1824 (Reference 50) NIST Special Publication 1086 (Reference 94)	• The model is validated in NUREG-1824 and an authoritative publication of the National Institute of Standards and Technology. • The licensee stated that in most cases, the model has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the model was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this model in the Brunswick application is acceptable.
Fire Dynamics Simulator (Version 5)	Analysis of CT transformer fire scenarios in the diesel generator basement and fire induced flows in MCCs (the latter was not used in the final Fire PRA)	NUREG-1824 (Reference 50) NIST Special Publication 1018-5, Volume 2 (Reference 95) NIST Special Publication 1018-5, Volume 3 (Reference 98)	• The model is validated in NUREG-1824 and authoritative publications of the National Institute of Standards and Technology. • The licensee stated that in most cases, the model has been applied within the validated range reported in NUREG-1824. The licensee provided justification for cases where the model was used outside the validated range reported in NUREG-1824 (Response to FM RAI 3.b, Reference 13). Based on these observations, the NRC staff concludes that the use of this model in the Brunswick application is acceptable.

Attachment C: Abbreviations and Acronyms

ADAMS	Agencywide Documents Access and Management System
AHJ	authority having jurisdiction
ANS	American Nuclear Society
AOG	auxiliary off-gas
ASME	American Society of Mechanical Engineers
Brunswick	Brunswick Steam Electric Plant
BTP	Branch Technical Position
BWR	boiling-water reactor
BWRVIP	Boiling Water Reactor Vessels and Internals Project
CAROLFIRE	Cable Response to Live Fire
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
CSR	Cable spreading room
DESIREE-Fire	Direct Current Electrical Shorting in Response to Exposure Fire
DID RA	defense-in-depth recovery action
DID	defense-in-depth
DG	diesel generator
EEEE	existing engineering equivalency evaluation
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier system
F&O	facts and observations
FAQ	frequently asked question
FDS	fire dynamics simulator
FDT	fire dynamics tool
FIVE	Fire Induced Vulnerability Evaluation Methodology
FM	fire modeling
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
GDC	general design criteria
GFMT	generic fire modeling treatments
GL	generic letter
HEP	human error probability
HGL	hot gas layer
HRA	human reliability analysis
HRE	high(er) risk evolution
HRR	heat release rate
HVAC	heating, ventilation, and air conditioning
IEEE	Institute of Electrical and Electronics Engineers
KSF	key safety function
kV	kilovolt
kW	kilowatt

LAR	license amendment request
LER	license event report
LERF	large early release frequency
MCB	main control board
MCR	main control room
min	minute(s)
MSO	multiple spurious operation
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NIST	National Institute of Standards and Technology
NLO	Non-licensed operator
No.	number
NPO	non-power operation
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSCA	nuclear safety capability assessment
NSPC	nuclear safety performance criteria
OMA	operator manual action
PAU	physical analysis unit
PB	performance-based
PCE	plant change evaluation
PCS	primary control station
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PWR	pressurized-water reactor
QA	quality assurance
RA	recovery action
RAI	request for additional information
RB	reactor building
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RG	Regulatory Guide
RHR	residual heat removal
RI	risk-informed
RI/PB	risk-informed, performance-based
SE	safety evaluation
SER	safety evaluation report
SFPE	Society of Fire Protection Engineers
SOKE	state of knowledge correlation
SR	supporting requirement
SSA	safe shutdown analysis
SSC	structures, systems, and components
SSD	safe shutdown
TS	Technical Specification
UFSAR	updated final safety analysis report
V	Volt
V&V	verification and validation
VEWFDS	very early warning fire detectors

VFDR	variance from deterministic requirements
yr	year
ZOI	zone of influence

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

Sincerely,

/RA/

Andrew Hon, Project Manager
Plant Licensing Branch II-2
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-324 and 50-325

Enclosures:
1. Amendment No. 266 to DPR-71
2. Amendment No. 294 to DPR-62
3. Safety Evaluation

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