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

December 29, 2010

Mr. T. Preston Gillespie
Site Vice President
Oconee Nuclear Station
Duke Energy Carolinas, LLC
7800 Rochester Highway
Seneca, SC 29672

SUBJECT: OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3, ISSUANCE OF AMENDMENTS REGARDING TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED FIRE PROTECTION PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c) (TAC NOS. ME3844, ME3845, AND ME3846)

Dear Mr. Gillespie:

The Nuclear Regulatory Commission has issued the enclosed Amendment Nos. 371, 373, and 372 to Renewed Facility Operating Licenses DPR-38, DPR-47, and DPR-55, for the Oconee Nuclear Station, Units 1, 2, and 3, respectively. The amendments consist of changes to the licenses and Technical Specifications (TSs) in response to your application dated application dated May 30, 2008, as supplemented by letters dated October 31, 2008, January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009. Duke Energy Carolinas, LLC (Duke, the licensee), submitted a license amendment request (LAR) to allow the licensee to maintain a fire protection program in accordance with 10 CFR 50.48(c) for the Oconee Nuclear Station, Units 1, 2, and 3 (ONS), and change the license and TSs accordingly.

By letter dated April 14, 2010, the licensee resubmitted the application and superseded the contents of the application submitted by letter dated May 30, 2008, as supplemented October 31, 2008. This resubmitted LAR, however, does not supersede the supplements dated January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009. By letters dated September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, the licensee supplemented the April 14, 2010, application.

A new fire protection license condition will replace the existing fire protection license condition in each unit's license. As a result of placing the new license condition in each unit's license, the NRC will be issuing license pages 2 through 11 in each unit's license because of pagination issues. The only changes to the licenses are the changes to the fire protection license condition.

~~Enclosure 4 transmitted herewith contains security related information. When separated from Enclosure 4, this document is decontrolled.~~

Pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR), by letter dated December 6, 2010, the NRC sent the licensee the draft Safety Evaluation approving the proposed amendments for an opportunity for the licensee to comment on any proprietary or security-related aspects of the draft Safety Evaluation. By letter dated December 22, 2010, the licensee provided comments. The NRC reviewed and accepted all comments made by the licensee. Pursuant to 10 CFR 2.390 the NRC has redacted information as identified by blank space enclosed within double brackets as shown here [[ ]].

In addition, the December 6, 2010, letter also requested the licensee to provide comments on factual errors or clarity concerns contained in the draft Safety Evaluation. By letter dated December 22, 2010, the licensee provided comments. The NRC has considered each comment and changed the Safety Evaluation as appropriate.

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.

If you have any questions, please call me at 301-415-1345.

Sincerely,

John Stang

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

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 371 to DPR-38
2. Amendment No. 373 to DPR-47
3. Amendment No. 372 to DPR-55
4. Safety Evaluation ~~contains official use only security-related information~~

cc: Distribution via Listserv

~~Enclosure 4 transmitted herewith contains security-related information. When separated from Enclosure 4, this document is decontrolled.~~

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-269

OCONEE NUCLEAR STATION, UNIT 1

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 371
Renewed License No. DPR-38

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 1 (the facility), Renewed Facility Operating License No. DPR-38 filed by the Duke Energy Carolinas, LLC (the licensee), dated April 14, 2010, and supplemented January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-38 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 371, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the license is hereby amended by changing the Renewed Facility Operating License No. DPR-38 fire protection License Condition 3.D to read as follows:

   D. Fire Protection

   Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

   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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease.)

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

4. This license amendment is effective as of its date of issuance and shall be fully implemented prior to January 1, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

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Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-38
and the Technical Specifications

Date of Issuance: December 29, 2010

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-270

OCONEE NUCLEAR STATION, UNIT 2

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 373
Renewed License No. DPR-47

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 2 (the facility), Renewed Facility Operating License No. DPR-47 filed by the Duke Energy Carolinas, LLC (the licensee), dated April 14, 2010, and supplemented January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-47 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 373, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the license is hereby amended by changing the Renewal Facility Operating License No. DPR-47 fire protection License Condition 3.D to read as follows:

   D. Fire Protection

   Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

   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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease.)

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

4. This license amendment is effective as of its date of issuance and shall be fully implemented prior to January 1, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

G (a

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-47
and the Technical Specifications

Date of Issuance: December 29, 2010

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

DUKE ENERGY CAROLINAS, LLC

DOCKET NO. 50-287

OCONEE NUCLEAR STATION, UNIT 3

AMENDMENT TO RENEWED FACILITY OPERATING LICENSE

Amendment No. 372
Renewed License No. DPR-55

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

   A. The application for amendment to the Oconee Nuclear Station, Unit 3 (the facility), Renewed Facility Operating License No. DPR-55 filed by the Duke Energy Carolinas, LLC (the licensee), dated April 14, 2010, and supplemented January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations as set forth in 10 CFR Chapter I;

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

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

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

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

2. Accordingly, the license is hereby amended by page changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 3.B of Renewed Facility Operating License No. DPR-55 is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 372, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. Accordingly, the license is hereby amended by changing the Renewed Facility Operating License No. DPR-55 ONS fire protection License Condition 3.D to read as follows:

D. Fire Protection

Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease.)

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

4. This license amendment is effective as of its date of issuance and shall be fully implemented prior to January 1, 2013.

FOR THE NUCLEAR REGULATORY COMMISSION

Gloria Kulesa, Chief
Plant Licensing Branch II-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment:
Changes to Renewed Facility
Operating License No. DPR-55
and the Technical Specifications

Date of Issuance: December 29, 2010

ATTACHMENT TO LICENSE AMENDMENT NO. 371

RENEWED FACILITY OPERATING LICENSE NO. DPR-38

DOCKET NO. 50-269

AND

TO LICENSE AMENDMENT NO. 373

RENEWED FACILITY OPERATING LICENSE NO. DPR-47

DOCKET NO. 50-270

AND

TO LICENSE AMENDMENT NO. 372

RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DOCKET NO. 50-287

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

Remove Pages	Insert Pages
Licenses	Licenses
License No. DPR-38, pages 2-9	License No. DPR-38, pages 2-11
License No. DPR-47, pages 2-9	License No. DPR-47, pages 2-11
License No. DPR-55, pages 2-9	License No. DPR-55, pages 2-11
TSs	TSs
5.0-6	5.0-6

On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-38, issued on February 6, 1973, is superseded by Renewed Facility Operating License No. DPR-38, which is hereby issued to Duke Energy Carolinas, LLC, to read as follows:

1. This license applies to the Oconee Nuclear Station, Unit 1, a pressurized water reactor and associated equipment (the facility) owned and operated by Duke Energy Carolinas, LLC. The facility is located in eastern Oconee County, about eight miles northeast of Seneca, South Carolina, and is described in the "Updated Final Safety Analysis Report" (UFSAR) as supplemented and amended and the Environmental Report as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Duke Energy Carolinas, LLC (the licensee):

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the Oconee Nuclear Station Site in accordance with the procedures and limitations set forth in this license;

   B. 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 UFSAR as supplemented and amended;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess, and use at any time 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;

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

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Oconee Nuclear Station, Units 1, 2 and 3.

3. This 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 371, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

    (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

    (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

the following criteria: (1) its existing or proposed facilities are economically and technically feasible of interconnection with those of the applicant and (2) with the exception of municipalities, cooperatives, governmental agencies or authorities, and associations, it is, or upon commencement of operations will be, a public utility and subject to regulation with respect to rates and service under the laws of North Carolina or South Carolina or under the Federal Power Act; provided, however, that as to associations, each member of such association is either a public utility as discussed in this clause (2) or a municipality, a cooperative or a governmental agency or authority.

(c) Where the phrase "neighboring entity" is intended to include entities engaging or proposing to engage only in the distribution of electricity, this is indicated by adding the phrase "including distribution systems".

(d) "Cost" means any appropriate operating and maintenance expenses, together with all other costs, including a reasonable return on applicant's investment, which are reasonably allocable to a transaction. However, no value shall be included for loss of revenues due to the loss of any wholesale or retail customer as a result of any transaction hereafter described.

2. (a) Applicant will interconnect and coordinate reserves by means of the sale and exchange of emergency and scheduled maintenance bulk power with any neighboring entity(ies), when there are net benefits to each party, on terms that will provide for all of applicant's properly assignable costs as may be determined by the Federal Energy Regulatory Commission and consistent with such cost assignment will allow the other party the fullest possible benefits of such coordination.

(b) Emergency service and/or scheduled maintenance service to be provided by each party will be furnished to the fullest extent available from the supplying party and desired by the party in need. Applicant and each party will provide to the other emergency service and/or scheduled maintenance service if and when available from its own generation and, in accordance with recognized industry practice, from generation of others to the extent it can do so without impairing service to its customers, including other electric systems to whom it has firm commitments.

(c) Each party to a reserve coordination arrangement will establish its own reserve criteria, but in no event shall the minimum installed reserve on each system be less than 15%, calculated as a percentage of estimated peak load responsibility. Either party, if it has, or has firmly planned, installed reserves in excess of the amount called for by its own reserve criterion, will offer any such excess as may in fact be available at the time for which it is sought and for such period as the selling party shall determine for purchase in accordance with reasonable industry practice by the other party to meet such other party's own reserve requirement. The parties will provide such amounts of spinning reserve as may be adequate to avoid the imposition of unreasonable demands on the other party(ies) in meeting the normal contingencies of operating its (their) system(s). However, in no circumstances shall such spinning reserve requirement exceed the installed reserve requirement.

(d) Interconnections will not be limited to low voltages when higher voltages are available from applicant's installed facilities in the area where interconnection is desired and when the proposed arrangement is found to be technically and economically feasible.

(e) Interconnection and reserve coordination agreements will not embody provisions which impose limitations upon the use or resale of power and energy sold or exchanges pursuant to the agreement. Further, such arrangements will not prohibit the participants from entering into other interconnection and coordination arrangements, but may include appropriate provisions to assure that (i) applicant receives adequate notice of such additional interconnection or coordination, (ii) the parties will jointly consider and agree upon such measures, if any, as are reasonably necessary to protect the reliability of the interconnected systems and to prevent undue burdens from being imposed on any system, and (iii) applicant will be fully compensated for its costs. Reasonable industry practice as developed in the area from time to time will satisfy this provision.

3. Applicant currently has on file, and may hereafter file, with the Federal Energy Regulatory Commission contracts with neighboring entity(ies) providing for the sale and exchange of short-term power and energy, limited term power and energy, economy energy, nondisplacement energy, and emergency capacity and energy. Applicant will enter into contracts providing for the same or for like transactions with any neighboring entity on terms which enable applicant to recover the full costs allocable to such transaction.

4. Applicant currently sells capacity and energy in bulk on a full requirements basis to several entities engaging in the distribution of electric power at retail. In addition, applicant supplies electricity directly to ultimate users in a number of municipalities. Should any such entity(ies) or municipality(ies) desire to become a neighboring entity as defined in ¶1(b) hereof (either alone or through combination with other), applicant will assist in facilitating the necessary transition through the sale of partial requirements firm power and energy. The provision of such firm partial requirements service shall be under such rates, terms and conditions as shall be found by the Federal Energy Regulatory Commission to provide for the recovery of applicant's costs. Applicant will sell capacity and energy in bulk on a full requirements basis to any municipality currently served by applicant when such municipality lawfully engages in the distribution of electric power at retail.

5. (a) Applicant will facilitate the exchange of electric power in bulk in wholesale transactions over its transmission facilities (1) between or among two or more neighboring entities, including distribution systems with which it is interconnected or may be interconnected in the future, and (2) between any such entity(ies) and any other electric system engaging in bulk power supply between whose facilities applicant's transmission lines and other transmission lines would form a continuous electric path, provided that permission to utilize such other transmission lines has been obtained. Such transaction shall be undertaken provided that the particular transaction reasonably can be accommodated by applicant's transmission system from a functional and technical standpoint and does not constitute the wheeling of power to a retail customer. Such transmission shall be on terms that fully compensate applicant for its cost. Any entity(ies) requesting such transmission arrangements shall give reasonable notice of its (their) schedule and requirements.

   (b) Applicant will include in its planning and construction program, sufficient transmission capacity as required for the transactions referred to in subparagraph (a) of this paragraph, provided that (1) the neighboring entity(ies) gives applicant sufficient advance notice as may be necessary reasonably to accommodate its (their) requirements from a functional and technical standpoint and (2) that such entity(ies) fully compensates applicant for its cost. In carrying out this subparagraph (b), however, applicant shall not be required to construct or add transmission facilities which (a) will be of no demonstrable present or future benefit to applicant, or (b) which could be constructed by the requesting entity(ies) without duplicating any portion of applicant's existing transmission lines, or (c) which would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements. Where regulatory or environmental approvals are required for the construction or addition of transmission facilities, needed for the transactions referred to in subparagraph (a) of this paragraph, it shall be the responsibility of the entity(ies) seeking the transaction to participate in obtaining such approvals, including sharing in the cost thereof.

6. To increase the possibility of achieving greater reliability and economy of electric generation and transmission facilities, applicant will discuss load projections and system development plans with any neighboring entity(ies).

7. When applicant's plans for future nuclear generating units (for which application will hereafter be made to the Nuclear Regulatory Commission) have reached the stage of serious planning, but before firm decisions have been made as to the size and desired completion date of the proposed nuclear units, applicant will notify all neighboring entities, including distribution systems with peak loads smaller than applicant's, that applicant plans to construct such nuclear units. Neither the timing nor the information provided need be such as to jeopardize obtaining the required site at the lowest possible cost.

8. The foregoing commitments shall be implemented in a manner consistent with the provisions of the Federal Power Act and all other lawful local, State and Federal regulation and authority. Nothing in these commitments is intended to determine in advance the resolution of issues which are properly raised at the Federal Energy Regulatory Commission concerning such commitments, including allocation of costs or the rates to be charged. Applicant will negotiate (including the execution of a contingent statement of intent) with respect to the foregoing commitments with any neighboring entity including distribution systems where applicable engaging in or proposing to engage in bulk power supply transactions, but applicant shall not be required to enter into any final arrangement prior to resolution of any substantial questions as to the lawful authority of an entity to engage in the transactions. In addition, applicant shall not be obligated to enter into a given bulk power supply transaction if: (1) to do so would violate, or incapacitate it from performing any existing lawful contract it has with a third party; (2) there is contemporaneously available to it, a competing or alternative arrangement which affords it greater benefits which would be mutually exclusive of such arrangement; (3) to do so would adversely affect its system operations or the reliability of power supply to its customers; or (4) if to do so would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements.

D. Fire Protection

Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease).

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

E. Physical Protection

Duke Energy Carolinas, LLC, shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains safeguards information protected under 10 CFR 73.21, is entitled: "Duke Energy Physical Security Plan" submitted by letter dated September 8, 2004, and supplemented on September 30, 2004, October 15, 2004, October 21, 2004, and October 27, 2004.

F. In the update to the UFSAR required pursuant to 10 CFR 50.71(e)(4) scheduled for July, 2001, the licensee shall update the UFSAR to include the UFSAR supplement submitted pursuant to 10 CFR 54.21(d) as revised on March 27, 2000. Until the UFSAR update is complete, the licensee may make changes to the programs described in its UFSAR supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. The licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 27, 2000, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than February 6, 2013.

H. Mitigation Strategy License Condition

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

(a) Fire fighting response strategy with the following elements:

1. Pre-defined coordinated fire response strategy and guidance
2. Assessment of mutual aid fire fighting assets
3. Designated staging areas for equipment and materials
4. Command and control
5. Training of response personnel

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

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

4. This renewed license is effective as of the date of issuance and shall expire at midnight on February 6, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by Roy P. Zimmerman

Roy Zimmerman, Acting Director
Office of Nuclear Reactor Regulation

Attachment:

1) Appendix A - Technical Specifications Renewed License No. DPR-38

Date of Issuance: May 23, 2000

On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-47, issued on October 6, 1973, is superseded by Renewed Facility Operating License No. DPR-47, which is hereby issued to Duke Energy Carolinas, LLC, to read as follows:

1. This license applies to the Oconee Nuclear Station, Unit 2, a pressurized water reactor and associated equipment (the facility) owned and operated by Duke Energy Carolinas, LLC. The facility is located in eastern Oconee County, about eight miles northeast of Seneca, South Carolina, and is described in the "Updated Final Safety Analysis Report" (UFSAR) as supplemented and amended and the Environmental Report as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Duke Energy Carolinas, LLC (the licensee):

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the Oconee Nuclear Station Site in accordance with the procedures and limitations set forth in this license;

   B. 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 UFSAR as supplemented and amended;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess, and use at any time 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;

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

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Oconee Nuclear Station, Units 1, 2 and 3.

3. This 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 373, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1(d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

   (a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

   (b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

the following criteria: (1) its existing or proposed facilities are economically and technically feasible of interconnection with those of the applicant and (2) with the exception of municipalities, cooperatives, governmental agencies or authorities, and associations, it is, or upon commencement of operations will be, a public utility and subject to regulation with respect to rates and service under the laws of North Carolina or South Carolina or under the Federal Power Act; provided, however, that as to associations, each member of such association is either a public utility as discussed in this clause (2) or a municipality, a cooperative or a governmental agency or authority.

(c) Where the phrase "neighboring entity" is intended to include entities engaging or proposing to engage only in the distribution of electricity, this is indicated by adding the phrase "including distribution systems".

(d) "Cost" means any appropriate operating and maintenance expenses, together with all other costs, including a reasonable return on applicant's investment, which are reasonably allocable to a transaction. However, no value shall be included for loss of revenues due to the loss of any wholesale or retail customer as a result of any transaction hereafter described.

2. (a) Applicant will interconnect and coordinate reserves by means of the sale and exchange of emergency and scheduled maintenance bulk power with any neighboring entity(ies), when there are net benefits to each party, on terms that will provide for all of applicant's properly assignable costs as may be determined by the Federal Energy Regulatory Commission and consistent with such cost assignment will allow the other party the fullest possible benefits of such coordination.

(b) Emergency service and/or scheduled maintenance service to be provided by each party will be furnished to the fullest extent available from the supplying party and desired by the party in need. Applicant and each party will provide to the other emergency service and/or scheduled maintenance service if and when available from its own generation and, in accordance with recognized industry practice, from generation of others to the extent it can do so without impairing service to its customers, including other electric systems to whom it has firm commitments.

(c) Each party to a reserve coordination arrangement will establish its own reserve criteria, but in no event shall the minimum installed reserve on each system be less than 15%, calculated as a percentage of estimated peak load responsibility. Either party, if it has, or has firmly planned, installed reserves in excess of the amount called for by its own reserve criterion, will offer any such excess as may in fact be available at the time for which it is sought and for such period as the selling party shall determine for purchase in accordance with reasonable industry practice by the other party to meet such other party's own reserve requirement. The parties will provide such amounts of spinning reserve as may be adequate to avoid the imposition of unreasonable demands on the other party(ies) in meeting the normal contingencies of operating its (their) system(s). However, in no circumstances shall such spinning reserve requirement exceed the installed reserve requirement.

(d) Interconnections will not be limited to low voltages when higher voltages are available from applicant's installed facilities in the area where interconnection is desired and when the proposed arrangement is found to be technically and economically feasible.

(e) Interconnection and reserve coordination agreements will not embody provisions which impose limitations upon the use or resale of power and energy sold or exchanges pursuant to the agreement. Further, such arrangements will not prohibit the participants from entering into other interconnection and coordination arrangements, but may include appropriate provisions to assure that (i) applicant receives adequate notice of such additional interconnection or coordination, (ii) the parties will jointly consider and agree upon such measures, if any, as are reasonably necessary to protect the reliability of the interconnected systems and to prevent undue burdens from being imposed on any system, and (iii) applicant will be fully compensated for its costs. Reasonable industry practice as developed in the area from time to time will satisfy this provision.

3. Applicant currently has on file, and may hereafter file, with the Federal Energy Regulatory Commission contracts with neighboring entity(ies) providing for the sale and exchange of short-term power and energy, limited term power and energy, economy energy, nondisplacement energy, and emergency capacity and energy. Applicant will enter into contracts providing for the same or for like transactions with any neighboring entity on terms which enable applicant to recover the full costs allocable to such transaction.

4. Applicant currently sells capacity and energy in bulk on a full requirements basis to several entities engaging in the distribution of electric power at retail. In addition, applicant supplies electricity directly to ultimate users in a number of municipalities. Should any such entity(ies) or municipality(ies) desire to become a neighboring entity as defined in ¶1(b) hereof (either alone or through combination with other), applicant will assist in facilitating the necessary transition through the sale of partial requirements firm power and energy. The provision of such firm partial requirements service shall be under such rates, terms and conditions as shall be found by the Federal Energy Regulatory Commission to provide for the recovery of applicant's costs. Applicant will sell capacity and energy in bulk on a full requirements basis to any municipality currently served by applicant when such municipality lawfully engages in the distribution of electric power at retail.

5. (a) Applicant will facilitate the exchange of electric power in bulk in wholesale transactions over its transmission facilities (1) between or among two or more neighboring entities, including distribution systems with which it is interconnected or may be interconnected in the future, and (2) between any such entity(ies) and any other electric system engaging in bulk power supply between whose facilities applicant's transmission lines and other transmission lines would form a continuous electric path, provided that permission to utilize such other transmission lines has been obtained. Such transaction shall be undertaken provided that the particular transaction reasonably can be accommodated by applicant's transmission system from a functional and technical standpoint and does not constitute the wheeling of power to a retail customer. Such transmission shall be on terms that fully compensate applicant for its cost. Any entity(ies) requesting such transmission arrangements shall give reasonable notice of its (their) schedule and requirements.

   (b) Applicant will include in its planning and construction program, sufficient transmission capacity as required for the transactions referred to in subparagraph (a) of this paragraph, provided that (1) the neighboring entity(ies) gives applicant sufficient advance notice as may be necessary reasonably to accommodate its (their) requirements from a functional and technical standpoint and (2) that such entity(ies) fully compensates applicant for its cost. In carrying out this subparagraph (b), however, applicant shall not be required to construct or add transmission facilities which (a) will be of no demonstrable present or future benefit to applicant, or (b) which could be constructed by the requesting entity(ies) without duplicating any portion of applicant's existing transmission lines, or

(c) which would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements. Where regulatory or environmental approvals are required for the construction or addition of transmission facilities, needed for the transactions referred to in subparagraph (a) of this paragraph, it shall be the responsibility of the entity(ies) seeking the transaction to participate in obtaining such approvals, including sharing in the cost thereof.

6. To increase the possibility of achieving greater reliability and economy of electric generation and transmission facilities, applicant will discuss load projections and system development plans with any neighboring entity(ies).

7. When applicant's plans for future nuclear generating units (for which application will hereafter be made to the Nuclear Regulatory Commission) have reached the stage of serious planning, but before firm decisions have been made as to the size and desired completion date of the proposed nuclear units, applicant will notify all neighboring entities, including distribution systems with peak loads smaller than applicant's, that applicant plans to construct such nuclear units. Neither the timing nor the information provided need be such as to jeopardize obtaining the required site at the lowest possible cost.

8. The foregoing commitments shall be implemented in a manner consistent with the provisions of the Federal Power Act and all other lawful local, State and Federal regulation and authority. Nothing in these commitments is intended to determine in advance the resolution of issues which are properly raised at the Federal Energy Regulatory Commission concerning such commitments, including allocation of costs or the rates to be charged. Applicant will negotiate (including the execution of a contingent statement of intent) with respect to the foregoing commitments with any neighboring entity including distribution systems where applicable engaging in or proposing to engage in bulk power supply transactions, but applicant shall not be required to enter into any final arrangement prior to resolution of any substantial questions as to the lawful authority of an entity to engage in the transactions. In addition, applicant shall not be obligated to enter into a given bulk power supply transaction if: (1) to do so would violate, or incapacitate it from performing any existing lawful contract it has with a third party; (2) there is contemporaneously available to it, a competing or alternative arrangement which affords it greater benefits which would be mutually exclusive of such arrangement; (3) to do so would adversely affect its system operations or the reliability of power supply to its customers; or (4) if to do so would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements.

D. Fire Protection

Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease).

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

E. Physical Protection

Duke Energy Carolinas, LLC, shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains safeguards information protected under 10 CFR 73.21, is entitled: "Duke Energy Physical Security Plan" submitted by letter dated September 8, 2004, and supplemented on September 30, 2004, October 15, 2004, October 21, 2004, and October 27, 2004.

F. In the update to the UFSAR required pursuant to 10 CFR 50.71(e)(4) scheduled for July, 2001, the licensee shall update the UFSAR to include the UFSAR supplement submitted pursuant to 10 CFR 54.21(d) as revised on March 27, 2000. Until the UFSAR update is complete, the licensee may make changes to the programs described in its UFSAR supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. The licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 27, 2000, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than February 6, 2013.

H. Mitigation Strategy License Condition

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

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

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

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

4. This renewed license is effective as of the date of issuance and shall expire at midnight on October 6, 2033.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By Roy P. Zimmerman

Roy P. Zimmerman, Acting Director
Office of Nuclear Reactor Regulation

Attachment:

1) Appendix A - Technical Specifications Renewed License No. DPR-47

Date of issuance: May 23, 2000

On the basis of the foregoing findings regarding this facility, Facility Operating License No. DPR-55, issued on July 19, 1974, is superseded by Renewed Facility Operating License No. DPR-55, which is hereby issued to Duke Energy Carolinas, LLC, to read as follows:

1. This license applies to the Oconee Nuclear Station, Unit 3, a pressurized water reactor and associated equipment (the facility) owned and operated by Duke Energy Carolinas, LLC. The facility is located in eastern Oconee County, about eight miles northeast of Seneca, South Carolina, and is described in the "Updated Final Safety Analysis Report" (UFSAR) as supplemented and amended and the Environmental Report as supplemented and amended.

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Duke Energy Carolinas, LLC (the licensee):

   A. Pursuant to Section 104b of the Act and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location on the Oconee Nuclear Station Site in accordance with the procedures and limitations set forth in this license;

   B. 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 UFSAR as supplemented and amended;

   C. Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess, and use at any time 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;

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

   E. Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the Oconee Nuclear Station, Units 1, 2 and 3.

3. This 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:

A. Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2568 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 372, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. This license is subject to the following antitrust conditions:

Applicant makes the commitments contained herein, recognizing that bulk power supply arrangements between neighboring entities normally tend to serve the public interest. In addition, where there are net benefits to all participants, such arrangements also serve the best interests of each of the participants. Among the benefits of such transactions are increased electric system reliability, a reduction in the cost of electric power, and minimization of the environmental effects of the production and sale of electricity.

Any particular bulk power supply transaction may afford greater benefits to one participant than to another. The benefits realized by a small system may be proportionately greater than those realized by a larger system. The relative benefits to be derived by the parties from a proposed transaction, however, should not be controlling upon a decision with respect to the desirability of participating in the transaction. Accordingly, applicant will enter into proposed bulk power transactions of the types hereinafter described which, on balance, provide net benefits to applicant. There are net benefits in a transaction if applicant recovers the cost of the transaction (as defined in ¶1 (d) hereof) and there is no demonstrable net detriment to applicant arising from that transaction.

1. As used herein:

(a) "Bulk Power" means electric power and any attendant energy, supplied or made available at transmission or sub-transmission voltage by one electric system to another.

(b) "Neighboring Entity" means a private or public corporation, a governmental agency or authority, a municipality, a cooperative, or a lawful association of any of the foregoing owning or operating, or proposing to own or operate, facilities for the generation and transmission of electricity which meets each of

the following criteria: (1) its existing or proposed facilities are economically and technically feasible of interconnection with those of the applicant and (2) with the exception of municipalities, cooperatives, governmental agencies or authorities, and associations, it is, or upon commencement of operations will be, a public utility and subject to regulation with respect to rates and service under the laws of North Carolina or South Carolina or under the Federal Power Act; provided, however, that as to associations, each member of such association is either a public utility as discussed in this clause (2) or a municipality, a cooperative or a governmental agency or authority.

(c) Where the phrase "neighboring entity" is intended to include entities engaging or proposing to engage only in the distribution of electricity, this is indicated by adding the phrase "including distribution systems".

(d) "Cost" means any appropriate operating and maintenance expenses, together with all other costs, including a reasonable return on applicant's investment, which are reasonably allocable to a transaction. However, no value shall be included for loss of revenues due to the loss of any wholesale or retail customer as a result of any transaction hereafter described.

2. (a) Applicant will interconnect and coordinate reserves by means of the sale and exchange of emergency and scheduled maintenance bulk power with any neighboring entity(ies), when there are net benefits to each party, on terms that will provide for all of applicant's properly assignable costs as may be determined by the Federal Energy Regulatory Commission and consistent with such cost assignment will allow the other party the fullest possible benefits of such coordination.

(b) Emergency service and/or scheduled maintenance service to be provided by each party will be furnished to the fullest extent available from the supplying party and desired by the party in need. Applicant and each party will provide to the other emergency service and/or scheduled maintenance service if and when available from its own generation and, in accordance with recognized industry practice, from generation of others to the extent it can do so without impairing service to its customers, including other electric systems to whom it has firm commitments.

(c) Each party to a reserve coordination arrangement will establish its own reserve criteria, but in no event shall the minimum installed reserve on each system be less than 15%, calculated as a percentage of estimated peak load responsibility. Either party, if it has, or has firmly planned, installed reserves in excess of the amount called for by its own reserve criterion, will offer any such excess as may in fact be available at the time for which it is sought and for such period as the selling party shall determine for purchase in accordance with reasonable industry practice by the other party to meet such other party's own reserve requirement. The parties will provide such amounts of spinning reserve as may be adequate to avoid the imposition of unreasonable demands on the other party(ies) in meeting the normal contingencies of operating its (their) system(s). However, in no circumstances shall such spinning reserve requirement exceed the installed reserve requirement.

(d) Interconnections will not be limited to low voltages when higher voltages are available from applicant's installed facilities in the area where interconnection is desired and when the proposed arrangement is found to be technically and economically feasible.

(e) Interconnection and reserve coordination agreements will not embody provisions which impose limitations upon the use or resale of power and energy sold or exchanges pursuant to the agreement. Further, such arrangements will not prohibit the participants from entering into other interconnection and coordination arrangements, but may include appropriate provisions to assure that (i) applicant receives adequate notice of such additional interconnection or coordination, (ii) the parties will jointly consider and agree upon such measures, if any, as are reasonably necessary to protect the reliability of the interconnected systems and to prevent undue burdens from being imposed on any system, and (iii) applicant will be fully compensated for its costs. Reasonable industry practice as developed in the area from time to time will satisfy this provision.

3. Applicant currently has on file, and may hereafter file, with the Federal Energy Regulatory Commission contracts with neighboring entity(ies) providing for the sale and exchange of short-term power and energy, limited term power and energy, economy energy, nondisplacement energy, and emergency capacity and energy. Applicant will enter into contracts providing for the same or for like transactions with any neighboring entity on terms which enable applicant to recover the full costs allocable to such transaction.

4. Applicant currently sells capacity and energy in bulk on a full requirements basis to several entities engaging in the distribution of electric power at retail. In addition, applicant supplies electricity directly to ultimate users in a number of municipalities. Should any such entity(ies) or municipality(ies) desire to become a neighboring entity as defined in ¶1(b) hereof (either alone or through combination with other), applicant will assist in facilitating the necessary transition through the sale of partial requirements firm power and energy. The provision of such firm partial requirements service shall be under such rates, terms and conditions as shall be found by the Federal Energy Regulatory Commission to provide for the recovery of applicant's costs. Applicant will sell capacity and energy in bulk on a full requirements basis to any municipality currently served by applicant when such municipality lawfully engages in the distribution of electric power at retail.

5. (a) Applicant will facilitate the exchange of electric power in bulk in wholesale transactions over its transmission facilities (1) between or among two or more neighboring entities, including distribution systems with which it is interconnected or may be interconnected in the future, and (2) between any such entity(ies) and any other electric system engaging in bulk power supply between whose facilities applicant's transmission lines and other transmission lines would form a continuous electric path, provided that permission to utilize such other transmission lines has been obtained. Such transaction shall be undertaken provided that the particular transaction reasonably can be accommodated by applicant's transmission system from a functional and technical standpoint and does not constitute the wheeling of power to a retail customer. Such transmission shall be on terms that fully compensate applicant for its cost. Any entity(ies) requesting such transmission arrangements shall give reasonable notice of its (their) schedule and requirements.

   (b) Applicant will include in its planning and construction program, sufficient transmission capacity as required for the transactions referred to in subparagraph (a) of this paragraph, provided that (1) the neighboring entity(ies) gives applicant sufficient advance notice as may be necessary reasonably to accommodate its (their) requirements from a functional and technical standpoint and (2) that such entity(ies) fully compensates applicant for its cost. In carrying out this subparagraph (b), however, applicant shall not be required to construct or add transmission facilities which (a) will be of no demonstrable present or future benefit to applicant, or (b) which could be constructed by the requesting entity(ies) without duplicating any portion of applicant's existing transmission lines, or (c) which would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements. Where regulatory or environmental approvals are required for the construction or addition of transmission facilities, needed for the transactions referred to in subparagraph (a) of this paragraph, it shall be the responsibility of the entity(ies) seeking the transaction to participate in obtaining such approvals, including sharing in the cost thereof.

6. To increase the possibility of achieving greater reliability and economy of electric generation and transmission facilities, applicant will discuss load projections and system development plans with any neighboring entity(ies).

7. When applicant's plans for future nuclear generating units (for which application will hereafter be made to the Nuclear Regulatory Commission) have reached the stage of serious planning, but before firm decisions have been made as to the size and desired completion date of the proposed nuclear units, applicant will notify all neighboring entities, including distribution systems with peak loads smaller than applicant's, that applicant plans to construct such nuclear units. Neither the timing nor the information provided need be such as to jeopardize obtaining the required site at the lowest possible cost.

8. The foregoing commitments shall be implemented in a manner consistent with the provisions of the Federal Power Act and all other lawful local, State and Federal regulation and authority. Nothing in these commitments is intended to determine in advance the resolution of issues which are properly raised at the Federal Energy Regulatory Commission concerning such commitments, including allocation of costs or the rates to be charged. Applicant will negotiate (including the execution of a contingent statement of intent) with respect to the foregoing commitments with any neighboring entity including distribution systems where applicable engaging in or proposing to engage in bulk power supply transactions, but applicant shall not be required to enter into any final arrangement prior to resolution of any substantial questions as to the lawful authority of an entity to engage in the transactions. In addition, applicant shall not be obligated to enter into a given bulk power supply transaction if: (1) to do so would violate, or incapacitate it from performing any existing lawful contract it has with a third party; (2) there is contemporaneously available to it, a competing or alternative arrangement which affords it greater benefits which would be mutually exclusive of such arrangement; (3) to do so would adversely affect its system operations or the reliability of power supply to its customers; or (4) if to do so would jeopardize applicant's ability to finance or construct on reasonable terms facilities needed to meet its own anticipated system requirements.

D. Fire Protection

Duke Energy Carolinas, LLC, 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 revised licensee's amendment request dated April 14, 2010, supplemented by letters dated: January 30, 2009, February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, November 30, 2009, September 13, 2010, September 27, 2010, October 14, 2010, November 19, 2010, and December 22, 2010, approved in the NRC safety evaluation (SE) dated December 29, 2010. Except where NRC approval for changes or deviations is required by 10 CFR 50.48(c), and provided no other regulation, technical specification, license condition or requirement would require prior NRC approval, the licensee may make changes to the fire protection program without prior approval of the Commission if those changes satisfy the provisions set forth in 10 CFR 50.48(a) and 10 CFR 50.48(c), the change does not require a change to a technical specification or a license condition, and the criteria listed below are satisfied.

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

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.

Due to the need for the licensee to have an industry full-scope peer review of its Fire PRA and to resolve the findings of that peer review, the licensee is not allowed to self-approve quantitative risk-informed fire protection program changes, except those implementation items needing a plant change evaluation as part of the Transition License Condition below. To enable self-approval of quantitative risk-informed fire protection program changes, the licensee will need to make a 10 CFR 50.90 submittal to the NRC requesting to change this license condition. The submittal should describe how the licensee has addressed each of the peer review findings and justify the adequacy of its Fire PRA for use in this application.

Other Changes that May Be Made Without Prior NRC Approval:

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

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

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

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

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

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

Prior NRC review and approval are not required for changes to the licensee's fire protection program that have been demonstrated to have no more than a minimal risk impact. The licensee may use its screening process as approved in the NRC SE dated December 29, 2010, to determine that certain fire protection program changes meet the minimal risk criterion. The licensee shall ensure that fire protection defense-in-depth and safety margins are maintained when changes are made to the fire protection program.

Transition License Conditions

1) The licensee shall complete the items described in Section 2.9, Table 2.9-1, "Implementation Items," in the NRC SE dated December 29, 2010, prior to January 1, 2013. Implementation items that result in a risk increase, as part of a plant change evaluation, can be self-approved by the licensee, as long as the overall transition risk remains a decrease (i.e., collective risk increases of transition and implementation are offset by the PSW modification risk decrease).

2) To complete the transition to full compliance with 10 CFR 50.48(c), the licensee shall implement the modifications listed in Section 2.8, Table 2.8.1-1, "Committed Plant Modifications," in the NRC SE dated December 29, 2010.

3) The licensee shall maintain appropriate compensatory measures in place until completion of all modifications and implementation items delineated above.

E. Physical Protection

Duke Energy Carolinas, LLC, shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification and safeguards contingency plans including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contains safeguards information protected under 10 CFR 73.21, is entitled: "Duke Energy Physical Security Plan" submitted by letter dated September 8, 2004, and supplemented on September 30, 2004, October 15, 2004, October 21, 2004, and October 27, 2004.

F. In the update to the UFSAR required pursuant to 10 CFR 50.71(e)(4) scheduled for July, 2001, the licensee shall update the UFSAR to include the UFSAR supplement submitted pursuant to 10 CFR 54.21(d) as revised on March 27, 2000. Until the UFSAR update is complete, the licensee may make changes to the programs described in its UFSAR supplement without prior Commission approval, provided that the licensee evaluates each such change pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

G. The licensee's UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised on March 27, 2000, describes certain future inspection activities to be completed before the period of extended operation. The licensee shall complete these activities no later than February 6, 2013.

H. Mitigation Strategy License Condition

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

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

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

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

4. This renewed license is effective as of the date of issuance and shall expire at midnight on July 19, 2034.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By Roy P. Zimmerman

Roy P. Zimmerman, Acting Director
Office of Nuclear Reactor Regulation

Attachment:

1) Appendix A - Technical Specifications Renewed License No. DPR-55

Date of issuance: May 23, 2000

5.0 ADMINISTRATIVE CONTROLS

5.4 Procedures

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

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

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

c. Quality assurance for effluent and environmental monitoring; and

d. All programs specified in Specification 5.5.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

TRANSITION TO A RISK-INFORMED, PERFORMANCE-BASED

FIRE PROTECTION

PROGRAM IN ACCORDANCE WITH 10 CFR 50.48(c)

RELATED TO

AMENDMENT NO. 371 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-38

AMENDMENT NO. 373 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-47

AND

AMENDMENT NO. 372 TO RENEWED FACILITY OPERATING LICENSE NO. DPR-55

DUKE ENERGY CAROLINAS, LLC

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

Security-related information pursuant to Title 10 of *Code of Regulations* (10 CFR), Section 2.390 has been redacted from this document .
Redacted information is identified by blank space enclosed within double brackets as shown here [[ ]].

ENCLOSURE 4

TABLE OF CONTENTS

ACRONYMS

AB	auxiliary building
AC	alternating current
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater system
AHJ	authority having jurisdiction
AHU	air-handling unit
ANS	American Nuclear Society
AOPs	abnormal operating procedures
AOV	air-operated valve
ARP	auxiliary relay panel
AS	accident sequence
ASD	alternative shutdown
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ASW	auxiliary service water
ATWS	anticipated transient without scram
Aux	auxiliary
AWT	activity waste tank
BH	block house
BHUT	bleed holdup tank
Bld	bleed
B&WOG	Babock & Wilcox Owners Group
BTP	Branch Technical Position
BWR	boiling-water reactor
BWST	borated water storage tank
CAP	corrective action program
CBAST	concentrated boric acid storage tank
CCs	capability categories
CCDP	conditional core damage probability
CCF	common cause failure
CCW	condenser circulating water
CCWS	component cooling water system
CDF	core damage frequency
CFAST	Consolidate Model of Fire Growth and Smoke Transport
CFR	*Code of Federal Regulations*
Chem	Chemistry
Clrs	coolers
Cmp	component
CRD	control rod drive
CSA	Canadian Standards Association
CSD	cold shutdown
CSIP	charging/safety injection pump
CRS	Control Room Supervisor
CT(s)	current transformers
DBD	design basis document
DBS	design basis specification
DC	direct current
DECON	decontamination
Demin	demineralizer

DG	diesel generator
DHR	decay heat removal
DID	defense-in-depth
EC	engineering change
EEE	engineering equivalency evaluation
EEEEs	existing engineering equivalency evaluation
EFW	emergency feedwater
EHC	electro-hydraulic control
EOP	emergency operating procedures
EPRI	Electric Power Research Institute
ERFBS	electrical raceway fire barrier system
ERFIS	emergency response facility information system
ES	engineered safeguards
ESFAS	engineered safety features actuation system
ESV	essential siphon vacuum
Evp	evaporator
EWST	elevated water storage tank
F&O(s)	facts and observations
F&O(s)	findings and observations
FACP	fire alarm control panel
FAQ(s)	frequently asked questions
Fltr	filter
FPE	fire protection engineer
FPIE	full power internal event
FPDID	fire protection defense-in-depth
FPP	fire protection program
FPRA	fire probabilistic risk assessment
FR	*Federal Register*
FRE	fire risk evaluation
FSA	fire safety analyses
FSAR	Final Safety Analysis Report
GDC	General Design Criterion
GL	U.S. NRC Generic Letter
GPM	gallons per minute
HEAF	high energy arcing faults
HEP	human error probabilities
HFEs	human failure events
HGL	hot gas layer
HPI	high-pressure injection
HPSW	high-pressure service water
HR	human reliability
HRA	human reliability analysis
HRE	high(er) risk evolution(s)
HRR	heat release rates
HSS	high safety significant
HUT	holdup tank
HVAC	heating ventilation air conditioning
HWP	hotwell pump
I&E	instrument and electronic
IEEE	Institute of Electrical and Electronic Engineers

IMC	NRC Inspection Manual Chapter
IN	U.S. NRC Information Notice
ISFSI	Independent Spent Fuel Storage Installation
ISLOCA	interfacing system loss-of-coolant accident
KEO	Keowee
KHS	Keowee Hydro Station
KSF(s)	key safety function(s)
LAR	license amendment request
LD	letdown
LDST	letdown storage tank
LERF	large early release frequency
L/H	low and high
LPI	low-pressure injection
LPIP	low-pressure injection pump
LPSW	low-pressure service water
MCB	main control board
MCC	motor control center
MCR	main control room
MDEFDW	motor-driven emergency feedwater
MFW	main feedwater
MOV(s)	motor-operated valves
MR	modification required
MREM	milla roentgen
MS	moisture separator
MSH(s)	main steam headers
MSO(s)	multiple spurious operation(s)
MSRH	moisture separator reheater
MSV	milla sievert
MT	main turbine
MTOT	main turbine oil tank
MWHT	main waste holdup tank
NEI	Nuclear Energy Institute
NEIL	Nuclear Electric Insurance Limited
NFPA	National Fire Protection Association
NPO	non-power operation
NRC	U.S. Nuclear Regulatory Commission
NSCA	nuclear safety capability assessment
NSD	Nuclear System Directive
NPSH	net positive suction head
NUREG	documents prepared by the NRC staff
OMA(s)	operator manual action(s)
ONS	Oconee Nuclear Station, Units 1, 2, and 3
OSC	Oconee site calculation
OSFD	ONS simplified flow diagram
OS&Y	outside screw and yoke
PALS	post-accident liquid sample
PB	performance-based
PCS	primary control station
PDS	plant damage states
PIC	process instrumentation cabinet

P&ID	piping and instrument drawings
PIP	problem identification program
PMG	performance monitoring group
Pmp	pump
POS(s)	plant operating state(s)
PORV(s)	power-operated relief valve(s)
PRA	probabilistic risk assessment
PSA	probabilistic safety assessment
PSI	pounds per square inch
PSW	protected service water
PVC	poly-vinyl chloride
PWR	pressurized-water reactor
PWROG	PWR Owners Group
QA	quality assurance
RA(s)	recovery action(s)
RAB	reactor auxiliary building
RAI	request for additional information
RAW	risk achievement worth
RB	reactor building
RBS	reactor building spray
RBES	reactor building emergency sump
RBSP	reactor building spray pump
RC	reactor coolant
RCA	radiation-controlled area
RCMU	reactor coolant makeup
RCP	reactor coolant pump
RCS	reactor coolant system
Res	resin
RG	regulatory guide
RHR	residual heat removal
RI	risk-informed
RIAs	radiation indicating alarms
RI/PB	risk-informed, performance-based
RI/PB FPP	risk-informed, performance-based fire protection program
RIS	Regulatory Information Summary
RMA	radioactive material area
RP	radiation protection
RWST	refueling water storage tank
SC	success criteria
SD	Site Directive
SDQA	software and data quality assurance
SE	safety evaluation
SFP	spent fuel pool
SFPE	Society of Fire Protection Engineers
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SLCs	selected licensee commitments
SM	safety margin
SOG	Standard Operating Guide

Spt	spent
SRs	supporting requirements
SRP	standard review plan
SRST	spent resin storage tank
SSC	structures, systems, and components
SSD	safe shutdown
SSE	SSD earthquake
SSEL	SSD equipment list
SSF	standby shutdown facility
SSHR	secondary side heat removal
SSPS	solid state protection system
SWGR	switchgear
SY	systems
SYD	switchyard
SW	service water
TB	turbine building
TDEFDW	turbine-driven emergency feedwater
TEDE	total effective dose equivalent
TH	thermal-hydraulic
TP	thermoplastic
TR	Technical Report
TS(s)	technical specification(s)
UFSAR	Updated Final Safety Analysis Report
UL	Underwriters Laboratory
V	volt
V&V	verification and validation
VAC	volts alternating current
VCT	volume control tank
VEWFDS	very early warning fire detection system
VFDR	variation from deterministic requirements
/yr	per year
Xfer	transfer
ZOI	zone of influence

1.0 INTRODUCTION

1.1 Background

On June 16, 2004, the U.S. Nuclear Regulatory Commission (NRC or the Commission) revised its regulation Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50, Section 50.48 to include a new paragraph 50.48(c). The new paragraph incorporates by reference National Fire Protection Association Standard 805 (NFPA 805), "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants 2001 Edition," (Reference 1) hereafter referred to as NFPA 805. This change to the NRC's fire protection regulations provides licensees with the opportunity to adopt a performance-based (PB) fire protection program (FPP) as an alternative to the existing, deterministic fire protection regulations. Specifically, NFPA 805 allows the use of PB methods, such as fire modeling, and risk-informed (RI) methods, such as fire probabilistic risk assessment (PRA), to demonstrate compliance with the nuclear safety performance criteria.

In the related license amendment request (LAR) and this safety evaluation (SE), extensive reference is made to NFPA 805. In particular, when this SE refers to an FPP element as being in compliance with, or meeting the requirements of, NFPA 805, the NRC staff intends this to indicate that the element is in compliance with 10 CFR 50.48(c) as well as the applicable portions of NFPA 805.

1.2 Requested Licensing Action

By application dated May 30, 2008 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML081650475) (Reference 2), as supplemented by letters dated June 30, 2008 (ADAMS Accession No. ML081890193), April 21, 2008 (ADAMS Accession No. ML091170546), February 9, 2009 (ADAMS Accession No. ML090480143), February 23, 2009 (ADAMS Accession No. ML090700134), May 31, 2009 (ADAMS Accession No. ML091590045), August 3, 2009 (ADAMS Accession No. ML092190212), September 29, 2009 (ADAMS Accession No. ML092740624), and November 30, 2009 (ADAMS Accession No. ML093410007) (References 3, 4, 5, 6, 7, 8, 9, and 10, respectively), Duke Energy Carolinas, LLC (Duke, the licensee), requested a license amendment to allow the licensee to maintain a FPP in accordance with 10 CFR 50.48(c) for the Oconee Nuclear Station, Units 1, 2, and 3 (ONS).

By letter dated April 14, 2010 (ADAMS Accession No. ML101121042 (Reference 11), the licensee resubmitted the LAR and superseded the contents of the LAR submitted by letters dated May 30, 2008, and October 31, 2008. This resubmitted LAR, however, does not supersede the supplements dated January 30, 2009 (ADAMS Accession No. ML091040205), February 9, 2009, February 23, 2009, May 31, 2009, August 3, 2009, September 29, 2009, and November 30, 2009. By letters dated September 13, 2010 (ADAMS Accession No. ML102640110) (Reference 12), September 27, 2010 (ADAMS Accession No. ML102720409) (Reference 13), October 14, 2010 (ADAMS Accession No. ML102910093) (Reference 54), November 19, 2010 (ADAMS Accession No. ML103300227) (Reference 52), and December 22, 2010 (ADAMS Accession No. ML103620105) (Reference 59), the licensee supplemented the LAR on April 14, 2010.

Pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR), by letter dated December 6, 2010, the NRC sent the licensee the draft Safety Evaluation and provided the licensee with an opportunity to comment on any proprietary or security-related aspects of the draft SE. By letter dated December 22, 2010, the licensee provided comments. The NRC reviewed and accepted all comments made by the licensee.

In addition, the December 6, 2010, letter also requested the licensee to provide comments on factual errors or clarity concerns contained in the draft SE. By letter dated December 22, 2010, the licensee provided comments. The NRC has considered each comment and changed the Safety Evaluation as appropriate.

The licensee is requesting amendments to the ONS renewed operating facility licenses and technical specifications (TSs) to establish and maintain a PB FPP in accordance with the requirements of 10 CFR 50.48(c). Specifically, the licensee requests to transition from the existing deterministic fire protection licensing basis established in accordance with 10 CFR 50.48(b) and 10 CFR Appendix R to a PB FPP in accordance with 10 CFR 50.48(c), that uses risk information, in part, to demonstrate compliance with the fire protection and nuclear safety goals, objectives, and performance criteria of NFPA 805. As such, the proposed FPP at ONS is referred to as risk-informed, performance-based (RI/PB) FPP throughout this SE.

The licensee has 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.

By letter dated April 14, 2010, (Reference 11), the licensee stated in their Section 4.2.3 that safe shutdown (SSD) requirements for fire protection, turbine building (TB) flooding, and physical security requirements were resolved by NRC approval of the station standby shutdown facility (SSF) design in an SE dated April 28, 1983 (ADAMS Accession No. ML103370444) (Reference 24). The fire protection portions of the approval have been incorporated into the Nuclear Safety Capability Assessment (NSCA) and so Reference 24 is no longer applicable to the ONS FPP.

2.0 REGULATORY EVALUATION

Section 50.48, "Fire protection," of 10 CFR provides the NRC requirements for nuclear power plant fire protection. Paragraph 50.48(c) of 10 CFR outlines the NRC requirements applicable to licensees that choose to adopt a PB FPP as an alternative to meeting the requirements of 10 CFR 50.48(b) for plants licensed to operate before January 1, 1979, or the approved fire protection license conditions for plants licensed to operate after January 1, 1979. ONS Units 1, 2 and 3 received their operating licenses prior to January 1, 1979.

The NRC regulations include specific procedural requirements for implementing an RI/PB FPP based on the provisions of NFPA 805. In particular, 10 CFR 50.48(c)(3)(i) requires licensees which choose to adopt an RI/PB FPP in compliance with NFPA 805 to submit an LAR to the NRC that identifies any orders and license conditions that must be revised or superseded, and contains any necessary revisions to the plant's TSs and the bases thereof. The license conditions issued with these amendments will supersede the current fire protection license condition with a condition that allows implementation of an FPP in accordance with NFPA 805.

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 FPP or nuclear power plant as permitted by NFPA 805."

The intent of this paragraph is given in the statement of considerations for the final rule, which was published in the *Federal Register* on June 16, 2004 (69 FR 33536). The statement of considerations states:

> This paragraph requires licensees to complete all of the Chapter 2 methodology (including evaluations and analyses) and to modify their fire protection plan before making changes to the fire protection program or to the plant configuration. This process ensures that the transition to an NFPA 805 configuration is conducted in a complete, controlled, integrated, and organized manner. This requirement also precludes licensees from implementing NFPA 805 on a partial or selective basis (e.g., in some fire areas and not others, or truncating the methodology within a given fire area).
>
> The evaluations and analyses process in Chapter 2 of NFPA 805 provides for the establishment of the fundamental fire protection program, identification of fire area boundaries and fire hazards, determination by analysis that the plant design satisfies the performance criteria, identification of the structures, systems and components (SSCs) required to achieve the performance criteria, conduct of plant change evaluations, establishment of a monitoring program, development of documentation, and configuration control. Chapter 2 of NFPA 805 also provides for the use of a deterministic or performance-based approach to determine that the performance criteria are satisfied and provides for the use of tools such as engineering analyses, fire models, nuclear safety capability assessments, and fire risk evaluations to support development of these approaches. The methodology for the use of these tools is established in Chapter 4 of NFPA 805 (69 FR 33548).

In the LAR, the licensee has provided a description of the revised FPP 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 amendment request and concludes that:

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

(2) The licensee has completed its implementation of the methodology in Chapter 2, "Methodology," of NFPA 805, including completion of all the required evaluations and analyses outlined by the statement of considerations, and the NRC staff has approved the licensee's modified FPP, which reflects the decision to comply with NFPA 805, consistent with 10 CFR 50.48(c)(3)(ii).

Since items (1) and (2) satisfy the requirements of 10 CFR 50.48(c)(3), the NRC staff concludes that the licensee's implementation of the modified FPP that aligns with NFPA 805, including physical plant modifications as described in the LAR and supplements, in accordance with the implementation schedule set forth in this SE and the accompanying license condition, is sufficient to demonstrate compliance with 10 CFR 50.48(c).

The regulations also allow for flexibility that was not originally included in the NFPA 805 standard. Licensees that choose to adopt 10 CFR 50.48(c), but wish to use the PB methods permitted elsewhere in the standard to meet the fire protection requirements of NFPA 805, Chapter 3, "Fundamental Fire Protection Program and Design Elements," may do so by submitting an LAR in accordance with 10 CFR 50.48(c)(2)(vii). Alternatively, licensees may choose to use RI or PB alternatives to comply with NFPA 805 by submitting an LAR in accordance with 10 CFR 50.48(c)(4).

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, licensees may also submit additional elements of their FPP for which they wish to receive specific NRC review and approval, as set forth in Regulatory Guide (RG) 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, Regulatory Position C.2.2.1, published in the *Federal Register* on December 18, 2009 (74 FR 67253; Reference 14). 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. However, any submittal addressing these additional FPP elements should include sufficient detail to allow the NRC staff to assess whether the licensee's treatment of these elements meets the 10 CFR 50.48(c) requirements.

The purpose of the FPP established by NFPA 805 is to provide assurance, through a defense-in-depth (DID) philosophy, that the fire protection objectives are satisfied.

NFPA 805, Section 1.2, "Defense-in-Depth," states the following:

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

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

2.1. Applicable Regulations

The licensee's FPP will generally be considered acceptable if it meets the applicable regulatory criteria established by the following regulations:

- 10 CFR Part 50, Appendix A, GDC 3, "Fire protection," establishes the general criteria for fire and explosion protection of SSCs important to safety.

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

- 10 CFR 50.48(a), requires that each operating nuclear power plant have a fire protection plan that meets the requirements of GDC 3.

- 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 a 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 license condition(s) related to fire protection. Because NFPA 805 was incorporated by reference into 10 CFR, all requirements of the endorsed standard must be met, unless an exemption is granted by the NRC as allowed in 10 CFR 50.12, "Specific Exemptions."

- 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's 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 (ADAMS Accession No. ML092730314), (Reference 14), which provides guidance to licensees for implementing an RI/PB FPP in compliance with 10 CFR 50.48(c).

- RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 1, issued November 2002 (ADAMS Accession No. ML023240437), (Reference 15), which provides guidance to licensees on acceptability limits for RI changes to the licensing basis.

- RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued March 2009 (ADAMS Accession No. ML090410014), (Reference 16), which provides guidance to licensees on methods for determining the technical adequacy of probabilistic risk assessment (PRA) results when used for RI changes to the licensing basis.

- RG 1.189, "Fire Protection for Operating Nuclear Power Plants," Revision 2, issued October 2009 (ADAMS Accession No. ML092580550), (Reference 17), which provides guidance to licensees on the proper content and quality of engineering equivalency evaluations used to support the FPP.

- NUREG-0800, Section 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection Program," Revision 0, issued December 2009 (ADAMS Accession No. ML092590527), (Reference 18), which provides the NRC staff with guidance for evaluating LARs that seek to implement a PB FPP in accordance with 10 CFR 50.48(c).

- NUREG-0800, Section 19.1, "Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2, issued June 2007 (ADAMS Accession No. ML071700657), (Reference 19), which provides the NRC staff with guidance for evaluating the technical adequacy of a licensee's PRA results when used to request RI changes to the licensing basis.

- NUREG-0800, Section 19.2, "Review of Risk Information Used to Support Permanent Plant-Specific Changes to the Licensing Basis: General Guidance," Revision 0, issued June 2007 (ADAMS Accession No. ML071700658), (Reference 20), which provides the NRC staff with guidance for evaluating the risk information used by a licensee to support permanent, RI changes to the licensing basis.

It should be noted that during the course of the review of the ONS NFPA 805 LAR, several of the above guidance documents were revised to incorporate updated information and lessons learned during the course of the transition process. As such, the original ONS NFPA 805 LAR was submitted against earlier revisions of some of these documents (e.g., RG 1.205). The revised LAR submitted on April 14, 2010 (Reference 11), incorporated into the application many of the positions in the new document revisions. Accordingly, the NRC staff considers that the NFPA 805 revised LAR meets the intent of the current document revisions, and was reviewed as such.

2.3. Interim Staff Positions (NFPA 805 Frequently Asked Questions Process)

During the ongoing NFPA 805 pilot transition process, as well as throughout the subsequent non-pilot reviews, the NRC staff, industry, and other interested stakeholders expect to gain experience and develop lessons learned during the submission and subsequent review of each LAR to transition a licensee to an RI/PB FPP. The lessons learned are often converted into interim staff positions, which apply to the ongoing review until they can be formally incorporated into the NFPA 805 guidance documents such as Nuclear Energy Institute (NEI) document NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)" (ADAMS Accession No. ML081130188), (Reference 21), as endorsed, and RG 1.205.

The lessons learned and interim staff positions address the NRC's performance goals of maintaining safety, improving effectiveness and efficiency, reducing regulatory burden, and increasing public confidence. In most cases, the meetings and other interactions involved in promulgating interim staff positions are open to the public and feedback is welcomed. With respect to the NFPA 805 LARs, the NRC established the frequently asked questions (FAQs) process as described in Regulatory Information Summary (RIS) 2007-19, "Process for Communicating Clarifications of Staff Positions Provided in Regulatory Guide 1.205 Concerning Issues Identified during the Pilot Application of National Fire Protection Association Standard 805," (ADAMS Accession No. ML071590227), (Reference 22), to clarify issues encountered during the pilot transition process.

The FAQ process provides a means for the NRC staff to establish and communicate interim positions on technical and regulatory issues that emerge as experience is gained during review of the NFPA 805 LARs. Approved interim staff positions documented through the FAQ process are used where applicable in reviewing those portions of the LAR to which they apply.

The following table provides the current set of FAQs the NRC staff used in the preparation of this SE, as well as the SE section to which the FAQ was applied.

Table 2.3-1: NFPA 805 Frequently Asked Questions

FAQ #	Rev.	FAQ Title	Closure Memo ADAMS Accession Nos.	SE Section
06-0008	9	Fire Protection Engineering Evaluations	ML073380976	4.0
06-0022	3	Acceptable Electrical Cable Construction Tests	ML091240278	3.1
07-0032	2	10 CFR 50.48(a) and GDC Clarification	ML081400292	2.0
07-0039	2	Provide Update for NEI 04-02, Table B-2	ML091320068	3.2
07-0040	4	Non-Power Operations Clarification	ML082200528	3.5
08-0048	0	NUREG/CR-6850 Revised Fire Ignition Frequencies (ADAMS Accession No. ML052580075)	ML092190457	3.5

2.4. Orders, License Conditions and Technical Specifications

Paragraph 50.48(c)(3)(i) of 10 CFR Part 50 states that the LAR "must identify any orders and license conditions that must be revised or superseded, and contain any necessary revisions to the plant's technical specifications and the bases thereof."

2.4.1. Orders

The NRC staff reviewed Section 5.2.3, "Orders and Exemptions," and Attachment O, "Orders and Exemptions," of ONS's NFPA 805 License Amendment Request Transition Report, as revised on April 14, 2010 (Reference 11), hereafter referred to simply as the LAR, with regard to NRC-issued Orders pertinent to ONS that are being revised or superseded by the NFPA 805 transition process. The licensee determined that no Orders need to be superseded or revised to implement an FPP at ONS that complies with 10 CFR 50.48(c).

This review, conducted by the licensee, included an assessment of docketed correspondence files and electronic searches, including internal ONS records and ADAMS. The review was performed to ensure that compliance with the physical protection requirements, security orders, and adherence to commitments applicable to ONS are maintained. The NRC staff accepts the licensee's determination that no Orders need to be superseded or revised to implement NFPA 805 at ONS.

In addition, a specific review was performed of the license amendment that incorporated the mitigation strategies required by Section B.5.b of Commission Order EA-02-026 (ADAMS Accession No. ML072260290), (Reference 23) to ensure that any changes being made in order to comply with 10 CFR 50.48(c) do not invalidate existing commitments applicable to ONS. The licensee's review of this Order and the related license amendment demonstrated that changes to the FPP during transition to NFPA 805 will not affect the mitigation measures required by Section B.5.b.

2.4.2. License Conditions

The NRC staff reviewed LAR Section 5.2.1, "License Condition Changes," and Attachment M, "License Condition Changes," regarding changes the licensee is seeking to make to the ONS 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 the licensee requested, which supersedes the current ONS fire protection license condition 3.D, for consistency with the format and content guidance outlined by Regulatory Position C.3.1 of RG 1.205, Revision 1. This section of RG 1.205 outlines an approach acceptable to the NRC staff for promulgating a fire protection license condition in accordance with the requirements of NFPA 805. Overall, the licensee's replacement license condition conforms to the guidance in RG 1.205, Revision 1.

Furthermore, the revised license condition, as specified by the sample license condition, identifies the plant-specific modifications outlined in the LAR, and associated implementation schedules, which must be accomplished at ONS to complete the transition to NFPA 805. In addition, the revised license condition includes a requirement that appropriate compensatory measures will remain in place until implementation of the specified plant modifications are completed. The modifications, implementation schedules, and compensatory measures ensure that completion of the transition to NFPA 805 at ONS will be orderly and conducted in accordance with the applicable regulations and license conditions.

Once these and other implementation issues are completed, NFPA 805 will be fully in effect at ONS, and provided that the licensee implements the RI/PB FPP as described in the LAR, as supplemented, the licensee will be in full compliance with 10 CFR 50.48(c). These modifications and implementation schedules are identical to those identified in the LAR, as discussed in SE Sections 2.8.1 and 2.8.2, and explicitly reviewed in Section 3.0, of this SE.

The licensee's proposed license condition is consistent with the content and format of the sample license condition in RG 1.205, Revision 1. Section 4.0 of this SE discusses the proposed ONS FPP license condition.

2.4.3. Technical Specifications

The NRC staff reviewed LAR Section 5.2.2, "Technical Specifications" and Attachment N, "Technical Specification Changes," with regard to proposed changes to the ONS TSs that are being revised or superseded during the NFPA 805 transition process. According to the LAR, the licensee conducted a review of the ONS TSs, including proposed TS changes that have been submitted to the NRC for approval, to determine which TS sections will be impacted by the transition to an RI/PB FPP based on 10 CFR 50.48(c) The licensee identified three changes.

The first change is to delete TS Section 5.4.1. TS 5.4.1 currently states that written procedures shall be established, implemented, and maintained covering activities that include FPP implementation. As discussed in the LAR, TS Section 5.4.1 is being deleted because, after completion of the transition to NFPA 805, the requirement for establishing, implementing, and maintaining fire protection procedures is contained in 10 CFR 50.48(c), as specifically outlined in Section 3.2.3, "Procedures," of NFPA 805. The licensee has stated that the RI/PB FPP at ONS complies with the requirements of NFPA 805 Section 3.2.3 (see Section 3.1.1 of this SE).

The second change is to revise the bases of ONS TS 3.10.1, "Standby Shutdown Facility (SSF)" to delete reference to Appendix R of 10 CFR Part 50. The bases for TS 3.10.1 currently refer to "10 CFR 50 Appendix R fire" four different times. As discussed in the LAR, the bases for TS 3.10.1 are being changed since 10 CFR Part 50, Appendix R, is no longer an appropriate basis for the ONS FPP.

The third change is to revise the bases of ONS TS 3.10.2, "Standby Shutdown Facility (SSF) Battery Cell Parameters," to delete reference to Appendix R of 10 CFR Part 50. The bases for

TS 3.10.2 currently refer to "a 10 CFR 50 Appendix R fire" two different times. As discussed in the LAR, the bases for TS 3.10.2 are being changed since 10 CFR Part 50, Appendix R, is no longer an appropriate basis for the ONS FPP.

2.5. Updated Final Safety Analysis Report (UFSAR)

The NRC staff reviewed LAR Section 5.4 "Revision to the ONS UFSAR" and Attachment Q, "UFSAR Changes" with regard to the proposed changes to the UFSAR as a result of transitioning to NFPA 805. Attachment Q states that the ONS UFSAR will be revised in accordance with 10 CFR 50.71(e) after this SE is issued.

The ONS transition to NFPA 805 represents a complete change in the licensing basis for their FPP. The NRC staff performed a review in order to determine that the licensee's proposed UFSAR changes are consistent with the RI/PB FPP described in the LAR (see below). The licensee's proposed changes to the UFSAR impact Section 9.5.1, "Fire Protection." Attachment Q provides an outline of the major sections and anticipated content of UFSAR Section 9.5.1 when it is revised. The major sections include:

- Section 9.5.1.1, "Design Basis Summary," will contain a general discussion of compliance with NFPA 805, Chapter 3, "Fundamental Fire Protection Program and Design Elements," a general discussion on NFPA 805, Chapter 4, "Performance Goal, Objectives and Criteria," (which includes discussions of nuclear safety for power and non-power conditions and a discussion of defense-in-depth), and a summary of radioactive release.

- Section 9.5.1.2, "Systems Description," will include a definition of power block structures, NSCA and equipment selection criteria, and required fire protection systems and features selection criteria.

- Section 9.5.1.3, "Safety Evaluation," will describe the methodology used to identify fire hazards, identify NSCA compliance strategies at power and non-power conditions on a fire area basis, demonstrate compliance with radioactive release criteria, summarize Fire PRA results, and summarize conclusions regarding compliance with NFPA 805.

- Section 9.5.1.4, "Inspection and Testing," will include information on inspection, testing, and surveillance methodologies and the monitoring program methodology.

- Section 9.5.1.5, "Personnel Qualification and Training," will include information on qualification and training of FPP personnel and the fire brigade.

2.6. Exemptions

The NRC staff reviewed LAR Section 5.2.3, "Orders and Exemptions," Attachment O, "Orders and Exemptions," and Attachment K, "Existing Licensing Action Transition" with regard to previously-approved exemptions to Appendix R to 10 CFR Part 50, which the transition to a FPP licensing basis in conformance with NFPA 805 will supersede. The licensee requested and received NRC approval for exemptions from 10 CFR Part 50, Appendix R. The licensee identified the following eight exemptions to 10 CFR Part 50, Appendix R, that are being superseded by the ONS FPP that complies with 10 CFR 50.48(c):

1. Auxiliary Building (AB) Lack of 3-hour fire rated Barrier (ADAMS Accession No. ML012000058), (Reference 27). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.2.a for the lack of 3-hour rated barrier separation between SSD circuits between the West Penetration Room Fire Areas and the Balance of Plant Fire Area.

2. AB Lack of 3-hour fire rated penetration seals (Reference 27). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.2.a for the lack of 3-hour fire rated barrier pipe penetrations separation between SSD circuits between the West Penetration Room Fire Areas and the Balance of Plant Fire Area.

3. AB Non-rated Expansion Joints (Reference 27). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.2.a for the lack of 3-hour fire rated cork in the expansion joint at the ceiling between the West Penetration Room Fire Areas and the Balance of Plant Fire Area.

4. Lack of Control Room Suppression (ADAMS Accession No. ML011990218), (Reference 38). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.3 for the lack of fixed suppression in the Control Rooms.

5. Outside and SSF Emergency Lighting (ADAMS Accession No. ML011990375), (Reference 39). This is an exemption from 10 CFR Part 50, Appendix R, Section III.J for the lack of 8-hour emergency lighting.

6. Reactor Building (RB) 20 feet Separation without Intervening Combustibles (Reference 27). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.2.d for the lack of 20 feet horizontal distance separation between SSD circuits in the RB with no intervening combustibles.

7. RB Unrated Containment Mechanical Penetrations (Reference 27). This is an exemption from 10 CFR Part 50, Appendix R, Section III.G.2.a for the lack of 3-hour fire rated barrier pipe penetrations separation between the West Penetration Fire Areas and the RB Fire Areas.

8. SSF Lack of Instrumentation per III.L.2 (ADAMS Accession No. ML091310038), (Reference 40). This is an exemption from 10 CFR Part 50, Appendix R, Section III.L.2 for the lack of a source range flux monitor and steam generator (SG) pressure indication at the SSF.

The NRC staff individually addresses the applicability and continuing validity of these exemptions as incorporated into the NFPA 805 FPP as part of the staff's review of the appropriate section or fire area involved (see SE Sections 3.2 and 3.3).

2.7. Self Approval Process for Post-Transition Fire Protection Program Changes

Upon completion of the implementation of the PB FPP and issuance of the license conditions, changes to the approved FPP must be evaluated to ensure that they are acceptable.

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

> In the event of a change to a previously approved fire protection program element, a risk-informed plant change evaluation shall be performed and the results used as described in 2.4.4 to ensure that the public risk associated with

fire-induced nuclear fuel damage accidents is low and that adequate defense-in-depth and safety margins are maintained.

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

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

As stated in RG 1.205, Regulatory Position C.3.1, the NRC may allow licensees to implement certain changes without prior NRC review and approval. A plant change evaluation must be performed for changes to the previously approved FPP, as stated above. An exception is for changes to certain NFPA 805, Chapter 3, requirements; this is discussed in Section 2.7.2. The specific implementation guidance documents associated with NFPA 805 (NEI 04-02, Section 5.3, and RG 1.205, Regulatory Position C.3.2) address the screening process and other requirements necessary to allow self-approval of plant changes with the potential to impact the RI/PB FPP. Changes that do not meet the acceptance criteria of the license condition may either be cancelled or the licensee may request a change to the FPP under 10 CFR 50.90.

2.7.1. Self Approval Using the Plant Change Evaluation Process

The NRC staff reviewed LAR Section 4.7.2, "Compliance with Configuration Control Requirements in Section 2.7.2 of NFPA 805," and LAR Attachment M, "License Condition Changes," for compliance with the NFPA 805 Plant Change Evaluation requirements.

The licensee will utilize a multi-step process for identifying and evaluating proposed changes to the plant that impact the FPP. The first step of the process is an initial review of the proposed plant change to determine if it has the potential to impact (change) the NFPA 805 FPP. This is accomplished through a series of questions/checklists contained in current ONS procedures. Initial reviews that identify potential FPP changes are further reviewed by a team of qualified individuals having relevant experience (i.e., Fire Protection, SSD/NSCA, Fire PRA) to determine the specific FPP changes, if any. If FPP changes are determined to exist as a result of the proposed plant change, a plant change evaluation must be performed. If the plant change is determined to comply with NFPA 805, Chapter 3 and/or Section 4.2.3, then a deterministic approach can be used.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the licensee's change evaluation process consists of the following four subtasks:

- Defining the Change
- Performing the Preliminary Risk Screening
- Performing the Risk Evaluation
- Evaluating the Acceptance Criteria

The licensee's plant change evaluation process starts with defining the change or altered condition to be evaluated and a review of the baseline configuration as defined by the existing licensing basis (i.e., the approved NFPA 805 FPP element).

Once the change has been defined, along with its relationship to the deterministically compliant condition or previously approved FPP element, a preliminary risk screening is performed. The licensee's preliminary risk screening process is modeled after the process provided in NEI 02-03, Revision. 0, "Guidance for Performing a Regulatory Review of Proposed Changes to the Approved Fire Protection Program," (ADAMS Accession No. ML031780500), (Reference 50), which it expects to address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.). The staff notes that NEI 02-03, although discussed in NEI 04-02, Revision 2, was not endorsed in RG 1.205, Revision 1 (i.e., it was not reviewed and endorsed by the NRC since it was only a "referenced document"). The NRC staff reviewed the licensee's preliminary risk screening process and determined that it meets the "assessment of the acceptability of risk" requirement of Section 2.4.4 of NFPA 805.

If the change to be evaluated does not screen out during the preliminary risk screening, the licensee's process allows a more detailed risk evaluation to be performed. These detailed evaluations may include fire modeling and risk assessment techniques. By letter dated August 3, 2009 (ADAMS Accession No. ML092190212 (Reference 8), the licensee stated that post-transition (to NFPA 805) plant changes requiring a detailed risk evaluation will be evaluated using the Fire PRA. The licensee also stated that its process for ensuring configuration control of the Fire PRA model complies with the American Society of Mechanical Engineers (ASME) Standard for PRA quality and ensures that the licensee maintains an as-built, as-operated PRA model of the plant. Section 3.4.3 of this SE discusses the technical adequacy of the licensee's Fire PRA, including the licensee's process to ensure that the Fire PRA remains current. In Section 3.4.3 of this SE, the NRC staff concludes that the licensee's PRA used to perform the risk assessments in accordance with NFPA 805 Section 2.4.4 (plant change evaluations) and Section 4.2.4.2 (fire risk evaluation) is of sufficient quality to support this application to transition to 10 CFR 50.48(c) because the remaining resolutions of findings on the internal events PRA and Fire PRA are not expected to change the substantial estimated risk decrease associated with this transition into a risk increase.

The proposed license condition as discussed in Section 4.0 of this SE does not allow the licensee to self-approve risk-informed changes to the FPP. The proposed license condition requires the licensee to submit a license amendment application (per 10 CFR 50.90) requesting such self-approval capability.

Based on the licensee's described process, the detailed risk evaluation will involve risk calculations for both CDF and LERF that will be used to model the proposed change and calculate the change in risk (i.e., ΔCDF and ΔLERF) with respect to the baseline configuration. Consistent with RG 1.205, Revision 1, Regulatory Position C.2.2.4.3, the post-transition baseline risk (used to evaluate cumulative risk impacts) is the risk of the plant at the point of full implementation of NFPA 805 (i.e., after completing all plant modifications and implementation items that the licensee has committed to make).

The final step in the plant change evaluation process involves determining whether the proposed change is acceptable with respect to risk, DID, and safety margin (SM), such that prior NRC review and approval is not required to implement the change. This step utilizes the guidance provided in NEI 04-02 and RG 1.205, Revision 1. As stated above, before achieving full compliance with 10 CFR 50.48(c) by implementing the plant modifications and implementation items listed in SE Sections 2.8 and 2.9 and subject to the NFPA 805 license condition and other license conditions (i.e., during full implementation of the transition to NFPA 805), RI changes to the licensee's FPP may not be made without prior NRC review and approval unless the change has been demonstrated to have no more than a minimal risk impact using the initial review and/or preliminary risk screening process discussed above (i.e., use of the detailed risk evaluation is not approved at this time). In addition, the licensee is required to ensure that fire protection DID and SMs are maintained during the transition process. The

NFPA 805 license condition includes the appropriate limitations, acceptance criteria, and other attributes to form an acceptable method for meeting Regulatory Position C.3.1 of RG 1.205, Revision 1, with respect to the requirements for FPP changes during transition, and therefore demonstrate compliance with 10 CFR 50.48(c).

2.7.2. Self Approval of Changes to NFPA 805, Chapter 3, Requirements

The NFPA 805 license condition also includes a provision for self-approval of changes to the NFPA 805, Chapter 3 fundamental FPP elements and design requirements for which an engineering evaluation demonstrates that the alternative to the NFPA 805, Chapter 3, element is functionally equivalent or adequate for the hazard. These two types of engineering evaluations, discussed in detail below, are not plant change evaluations because they conclude that the change to the NFPA 805, Chapter 3, requirement still maintains the function of the NFPA 805 requirement.

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. These fire protection engineering evaluations can use qualitative analyses. The basis of approval for a functionally equivalent evaluation is that it maintains the function of the NFPA 805 requirement. As such, the determination that the condition is functionally equivalent means that the evaluated condition complies with the code requirement.

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 utilize 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 (AHJ) for Section 1.7 type equivalencies 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 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 (with respect to the ability to meet the nuclear safety and radioactive release performance criteria), using a relevant technical requirement or standard.

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

- Fire Alarm and Detection Systems (Section 3.8)

- Automatic and Manual Water-Based Fire Suppression Systems (Section 3.9)
- Gaseous Fire Suppression Systems (Section 3.10)
- 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.

2.8. Implementation

Regulatory Position C.3.1 of RG 1.205, Revision 1, provides guidance that the NFPA 805 license condition presented in the LAR should include the following: (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 commitment to maintain appropriate compensatory measures in place until implementation of the modifications is completed.

2.8.1. Modifications

The NRC staff reviewed LAR Attachment S, "Plant Modifications and Confirmatory Items," which describes the ONS plant modifications necessary to implement the NFPA 805 licensing basis as proposed. These modifications are identified in the LAR as necessary to bring ONS into compliance with either the deterministic or PB requirements of NFPA 805. LAR Table S-1 in Attachment S provides a description of each of the proposed plant modifications and presents the problem statement explaining why the modification is needed. This table also explains for each modification, as appropriate, that compensatory measures are currently in place for existing deficiencies associated with 10 CFR Part 50, Appendix R compliance, and that compensatory measures will be established when the NFPA 805 FPP becomes effective and will remain in effect until the modification is completed.

The NRC staff's review confirmed that the modifications identified in LAR Table S-1 are the same as those identified in LAR Table B-3, "Fire Area Transition," on a fire area basis, as the modifications being credited in the proposed NFPA 805 plant configuration and licensing basis. The NRC staff also confirmed that the LAR Table S-1 modifications and associated implementation schedule are the same as those provided in the licensee's proposed NFPA 805 license condition (LAR Attachment N), and for which the licensee has committed to keep the appropriate compensatory measures in place until the modifications have been completed.

The plant modifications committed to in LAR Table S-1 must be completed in order for ONS to be in full compliance with 10 CFR 50.48(c) (NFPA 805). As discussed above, these modifications will be implemented in accordance with the schedule provided in the NFPA 805 license condition.

In addition, the licensee has committed to keep the appropriate compensatory measures in place for each modification until the modification has been fully implemented. Table 2.8.1-1 presents a simplified version of LAR Table S-1 and incorporates supplementary information provided by the licensee (Reference 59). The NRC will perform follow-up inspections to ensure that all items in Table 2.8.1 below have been completed prior to implementation of the license amendments.

Table 2.8.1-1: Committed Plant Modifications

Item No.	Problem Statement	Modification Description	Modification Completion	Proposed Compensatory Measures*
[[				]]

Item No.	Problem Statement	Modification Description	Modification Completion	Proposed Compensatory Measures*
[[				]]

[[ ]]

2.8.2. Schedule

In the LAR Section 5.5 and as supplemented, the licensee provided the overall schedule for completing the NFPA 805 transition at ONS. The licensee stated that it would complete the implementation of the new program, including procedure changes, process updates, and training for affected plant personnel, within 24 months after NRC approval, as conveyed by the date of issuance of this SE.

LAR Attachment S provides an implementation completion schedule for each of the identified plant modifications. This implementation schedule is provided in Table 2.8.1-1, and in the proposed license condition. In addition, the proposed license condition includes a statement that appropriate compensatory measures will remain in place until implementation of these modifications is fully implemented (see Section 4.0 of this SE).

2.9. Summary of Implementation Items

LAR Table S-2 in Attachment S provides a list of "Confirmatory Items" for ONS. These confirmatory items, referred to by the NRC as implementation items, are items that the licensee has not fully completed or implemented as of the issuance date of the SE, 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, NFPA 805 programs that have not been fully implemented, personnel training that is still underway, etc.). These items do not impact the bases for the safety conclusion made by the NRC staff in the associated SE.

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

As a result of its review of the ONS LAR, the NRC staff identified additional items that are contained in Table 2.9-1. For tracking purposes, the NRC staff has assigned a unique identifying number to each item. The table also specifies the associated section of the SE in which the item is identified, as well as the appropriate licensee document, which denotes that the action associated with the item is still ongoing, and provides some additional level of detail regarding what the change will entail.

Table 2.9-1: Implementation Items

Item	SE Section	Implementation Item Description	LAR Section
1.	Attachment A: Section 3.2.2.4, Management Policy on AHJ	The Design Basis Specification for Fire Protection will be updated to include the statement that the NRC is the AHJ for fire protection changes requiring approval.	LAR Table S-2, and Attachment A, Subsection 3.2.2.4
2.	Attachment A: Section 3.3.1.2(2), Control of Combustible Materials	Fleet Directive NSD-313, "Control of Flammable and Combustible Materials," will be updated to include the statement that plastic-sheeting materials shall conform to the requirements of NFPA 701 or equivalent.	LAR Table S-2, and Attachment A, Subsection 3.3.1.2.2
3.	Attachment A: Section 3.3.1.3.3, Control of Ignition Sources for Leak Testing	Appropriate station procedure(s) for leak or air flow testing will be updated to preclude the use of open flames or combustion generated smoke.	LAR Table S-2, and Attachment A, Subsection 3.3.1.3.3
4.	Attachment A: Section 3.3.3, Interior Finishes	Fleet Directive NSD-318 "Coatings Program," will be updated to include the specifications for Class A walls/ceilings and Class I floor finishes.	LAR Table S-2, and Attachment A Subsection 3.3.3
5.	Attachment A: Section 3.3.5.2, Electrical Raceway Construction Limits	Appropriate station electrical specifications will be updated to specify only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables.	LAR Table S-2, and Attachment A Subsection 3.3.5.2
6.	Attachment A: Section 3.3.9, Transformers	Transformer deluge system flow test procedures will be updated to include drainage inspections as part of the annual flow tests.	LAR Table S-2, and Attachment A Subsection 3.3.9
7.	Attachment A: Training and Drills, Subsection 3.4.3.(c)(3)	Station Fire Brigade Training documentation will be updated to include guidance to ensure fire drills are conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards.	LAR Table S-2, and Attachment A (Subsection 3.4.3.(c)(3))
8.	Section 3.7 and Attachment A: Section 3.2.3, Subsection 3.2.3.(3)	Implement the monitoring program described in SE Section 3.7.	LAR Table S-2, and Attachment A Subsection 3.2.3.(3)
9.	Attachment A: Section 3.4.2.1	Pre-fire Plans will be updated to include any changes to equipment important to nuclear safety and other updates pertinent to the NFPA 805 Transition.	LAR Table S-2, and Attachment A Subsection 3.4.2.1
10.	Attachment A: Section 3.4.2.3	Standard Operating Guidelines (SOGs) will be updated to include a SOG with the location of the Pre-Fire Plans.	LAR Table S-2, and Attachment A Subsection 3.4.2.3

11.	Attachment A: Section 3.8.2,	ONS code compliance calculation will be updated to ensure required fire detection devices are installed in accordance with NFPA 72, 2007 Edition.	LAR Table S-2, and Attachment A Subsection 3.8.2
12.	Attachment A: Section 3.9.1	Validate hydraulics calculations for all required automatic or manual water-based suppression systems.	LAR Table S-2, and Attachment A Subsection 3.9.1
13.	Attachment D, Fire Area [[ ]]	The SSD procedure and analysis will be updated to incorporate the monitoring and/or adjustment of the following parameters required during operation of the SSF diesel generator (DG): generator current, voltage, power and frequency. The controls and indications required to monitor and adjust these parameters are currently not included in the SSD analysis.	LAR Table S-2, Attachment C, Fire Area [[ ]]
14.	Section 3.2.4: Transition of Operator Manual Actions to Recovery Actions	Recovery Actions – Station procedures will be updated to reflect new NSCA strategies (including supporting communication coverage) and perform training as necessary. The following actions will be performed: 1) An evaluation to ensure that the hand-held radios operate in the locations of the recovery actions when needed, either with or without repeaters. 2) Development of SSD procedures for [[ ]] 3) Provide training to the operators on the new SSD procedures for [[ ]] 4) Conduct drills to ensure viability on the new [[ ]] safe shutdown procedures.	LAR Table S-2, Attachment C, Fire Areas [[ ]], [[ ]], and [[ ]] and Attachment G
15.	Section 3.5.2: Fire Protection during NPO Modes	Revise Fleet Directive NSD-403 and Site Directive (SD) 1.3.5 with the definition of high(er) risk evolution (HRE) to address non-power operation (NPO) criteria, e.g., Plant Operating State (POS) 1B. Also, reconcile NSD-403 and SD 1.3.5 Thermal Margin Criteria with the criteria in FAQ 07-0040 as needed.	LAR Attachment D VFDR ID # Oconee site calculation (OSC)-9268-01
16.	Section 3.5.2: Fire Protection during NPO Modes	Develop a process to evaluate the potential effects of a fire upon habitability and the impact of increased DID fire protection actions that can be added to the establishment of high	LAR Attachment D VFDR ID # OSC-9268-02

		confidence [[ ]] per Fleet Directive NSD-403.	
17.	Section 3.5.2: Fire Protection during NPO Modes	Implement procedural guidance to monitor [[ ]]	LAR Attachment D VFDR ID # OSC-9268-03
18.	Section 3.5.2: Fire Protection during NPO Modes	Develop procedural controls to monitor [[ ]] flow path during higher risk evolutions (HREs) for the outage risk management procedures.	LAR Attachment D VFDR ID # OSC-9268-04
19.	Section 3.5.2: Fire Protection during NPO Modes	Develop procedural controls for use of [[ ]] during HREs for the outage risk management procedures applicable to NPO key safety function (KSF).	LAR Attachment D VFDR ID # OSC-9268-05
20.	Section 3.5.2: Fire Protection during NPO Modes	Develop procedural controls on the [[ ]] for the outage risk management procedures.	LAR Attachment D VFDR ID # OSC-9268-06
21.	Section 3.5.2: Fire Protection during NPO Modes	Ensure capability to access (i.e., an operator can be dispatched to manually throttle) motor-operated valves (MOVs) [[ ]]	LAR Attachment D VFDR ID # OSC-9268-07
22.	Section 3.5.2: Fire Protection during NPO Modes	Ensure capability to access (i.e., an operator can be dispatched to manually open and close, respectively) manual valves [[ ]]	LAR Attachment D VFDR ID # OSC-9268-08
23.	Section 3.5.2: Fire Protection during NPO Modes	Complete the analysis of NPO fire impacts for fire zones following installation of the NFPA 805 committed modifications. After implementation, update Oconee Site Calculation (OSC)-9313 and its NPO recommendations for affected fire zones.	LAR Attachment D VFDR ID # OSC-9313-02
24.	Section 3.5.2: Fire Protection during NPO Modes	Develop procedure guidance for pre-emptive re-alignment of and the removal of power from the MOVs [[ ]]	LAR Attachment D VFDR ID # OSC-9313-03
25.	Section 3.5.2: Fire Protection during NPO Modes	Revise NSD-403, SD 1.3.5 and ONS technical procedures to implement the recommendations in OSC-9313, Attachment 1, subject to resolution of open Items (i.e., Items 15 through 24).	LAR Attachment D VFDR ID # OSC-9313-07

26.	Section 3.1.3.10: Fire Hose Standpipes Use of outside Fire Hydrant Appurtenances	Revise FPP Design Basis Specification for the [[ ]] fire hydrants.	LAR Attachment L Approval Request #10
27.	Section 3.8.2: Configuration Control	Configuration control procedures which govern the various ONS documents and databases will be revised to reflect the new RI/PB FPP licensing bases.	LAR Section 4.7.2
28.	Section 3.8.3: Quality	Training Position Specific Guides will be developed to identify and document required training and mentoring to ensure individuals are appropriately qualified per NFPA 805, Section 2.7.3.4.	LAR Section 4.7.3
29.	Section 3.8.3: Quality	Post-transition quality requirements from NFPA 805 that are not currently part of the ONS processes will be revised to include any additional requirements.	LAR Section 4.7.3
30.	Attachment D: [[ ]]	Operator Guidance – ONS procedures will be updated to include the following: 1) Guidance for maintaining the plant safe and stable following loss of all [[ ]] 2) Guidance for operation of [[ ]]	LAR Attachment C [[ ]]
31.	Attachment D: [[ ]]	Resolve the physical location issue of the [[ ]] requirements by revising the fire risk evaluation to denote the physical separation aspects of the [[ ]]	LAR Attachment C [[ ]]
32.	Attachment D: [[ ]]	Incorporate [[ ]] into FPP site documents after the modification is implemented.	LAR Attachment C
33.	Section 3.2.1: Section 3.3.1.7, Section 3.3.3.3, Section 3.5.2.4, Section 3.5.2.5,	Incorporating all related non-coordinated information in the NSCA and NPO Pinch Point Analysis, and updating the Fire PRA model, to include the results of the breaker coordination study	LAR Sections 4.2, 4.3, & 4.5, RAI Response (Reference 54)
34.	Section 3.1.3.4 Use of Unqualified Video / Communication / Data Cables	Future acceptable cable construction qualifications will be included in the Power Generation Electrical Discipline Design Criteria Manual. A specific line item will be added that video / communication / data cables shall be plenum rated and/or tested in accordance with Institute of Electrical and Electronic Engineers (IEEE) 383-1974, IEEE 1202-1991, CSA 22.2 No.	LAR Section 4.1, RAI Response (Reference 12)

		0.3, NFPA 262, UL 44, UL 83, UL 1581, UL 1666, or UL 1685 as accepted in FAQ 06-0022. Electrical wiring, including video, phone, and communications, installed above a suspended ceiling shall be rated for plenum use, routed in metallic conduit, routed in cable tray with solid metal top and bottom covers, or armored cable.	
35.	Attachment A: Section 3.3.1.3.4 Plant Administrative Procedures	Appropriate directives will be updated to clearly indicate that only portable electric heaters are permitted to be used in plant areas with equipment important to nuclear safety or where there is the potential for radiological release due to fire. Portable fuel-fired heaters are not permissible in these areas.	LAR Section 4.1, request for additional information (RA)I Response (Reference 52)
36.	Section 3.1.3.7 [[ ]]	The fire brigade will develop a SOG for fighting a fire [[ ]] Training is already performed on tactics for fighting fires of this nature but training will be reinforced with a new SOG. The Fire Brigade Administrator will review the Pre-Fire Plans to determine if enhancement is necessary.	LAR Section 4.1, RAI Response (Reference 52)
37.	Section 3.7 Monitoring Program	Develop instructions for the software program to collect availability and reliability data on SSCs in the Monitoring Program.	LAR Section 4.6, RAI Response (Reference 52)
38.	Section 3.2.1 NSCA Methods	Revise the B-2 Table to include additional clarification of alignment with the NEI guidance.	LAR Section 4.2, RAI Response Reference (12)
39.	Section 3.2.1 NSCA Methods	Development and documentation of a long term SSD program including analysis, equipment reviews, recovery actions, modifications, and procedural guidance.	LAR Section 4.2, RAI Response (References 52 & 54)
40.	Section 3.2.1 NSCA Methods	Complete activities needed to provide assurance that fire-induced open secondary circuits of current transformers will not impact the ability to achieve and maintain the fuel in a safe and stable condition.	LAR Section 4.2, RAI Response (Reference 12)
41.	Section 3.4.3 PRA Quality	With regard to the Internal Events PRA, complete the following: • Determine whether the HRA model needs to be updated or upgraded. • Update/upgrade the HRA model. • If HRA model was upgraded conduct a focus-scope peer review of the revised internal events PRA model	LAR Section 4.5, RAI Response (Reference 59)

		with respect to HRA. • Disposition all findings from the peer review and revise the internal events PRA, as appropriate.	

42	Section 3.4.3
	Fire PRA Quality
		With regard to the Fire PRA, complete the following:
		• Update/upgrade the Fire PRA as appropriate to resolve NRC staff review findings in SE Attachment C, Table 3.4-2.
		• Complete an industry full-scope peer review of the revised Fire PRA that is performed to the ASME/ANS RA-Sa-2009 PRA standard, as endorsed by RG 1.200, Rev. 2. The full-scope peer review will include specific focus on the following elements:
		  • Influence on the target set from fire propagation beyond the ignition source due to intervening combustibles and cables on:
		    • Expanding the zone of influence (ZOI), both vertically and horizontally, and
		    • hot gas layer (HGL) formation, including the effects on fire detection and brigade response.
		  • Modeling of high-energy arcing faults on [[ ]] bus ducts.
		  • Deviation from NUREG/CR-6850 guidance and as modified by closed FAQs will be treated as described in NEI 07-12 (Fire Probabilistic Risk Assessment Peer Review Guidelines) and the fire aspects of ASME/ANS PRA Standard, as endorsed by RG 1.200.
		• Disposition findings from the full-scope Fire PRA peer review and revise the Fire PRA as appropriate.
			LAR Section 4.5,
			RAI Response
			(Reference 59)

43	Section 3.4.6 Cumulative Risk and Combined Changes	Confirm that the risk decrease from the as-built [[ ]] continues to bound the cumulative VFDR transition risk once the [[ ]]are installed	LAR Section 4.2, RAI Response (Reference 52)
44	Section 3.2.1: Section 3.3.1.7, Section 3.3.3.3,	The breaker coordination study will be updated to include all new NFPA 805 SSD equipment list (SSEL)-related	LAR Section 4.2, RAI Response (Reference 52)

	Section 3.5.2.4, Section 3.5.2.5,	power supplies (i.e., PSW) for power and non-power operations, and additional plant modification will be defined if necessary to ensure that the assumptions of the Fire PRA and NSCA remain valid.	
45.	Section 3.8.1: Documentation	The ONS "Fire Protection Program Design Basis Document" and supporting documentation will be revised to incorporate NFPA 805 documents.	LAR Section 4.7.1
46	Section 3.2.1 NSCA Methods	Licensee agreed to eliminate the *"10 minute free of fire damage"* assumption. The ONS FPP and supporting documentation (including the B-2 Table, B-3 Table, all applicable fire risk evaluations, Fire PRA, NSCA, and operator manual action(s) (OMA) feasibility calculations) will be revised to eliminate the assumptions. Compliance will be demonstrated consistent with NFPA 805, Section 4.2.4.2.	LAR Section 4.2, RAI Response (Reference 52).
47	Attachment B: Section 3.1.1.7	Revised calculation OSC-9291, NFPA 805 Transition B-2 Table, Section 3.1.1.7 to reword the alignment basis to clearly state that [[ ]] is not credited for deterministic analysis and therefore not analyzed for its availability in the deterministic analysis. The licensee also states that alignment statement will be revised to ensure the proper relationship with the alignment basis.	RAI Response (Reference 12)

3.0 TECHNICAL EVALUATION

The following sections evaluate the technical aspects of the requested license amendment to transition the FPP at ONS to one based on NFPA 805 in accordance with 10 CFR 50.48(c). While performing the technical evaluation of the licensee's submittal, the NRC staff utilized the guidance provided in NUREG-0800, Chapter 9.5.1.2, "Risk-Informed, Performance-Based Fire Protection" (Reference 18), 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 SE Section 2.0. Specifically:

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

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

- Section 3.3 provides the results of the NRC staff's review of the fire modeling methods to demonstrate the ability to meet the nuclear safety performance criteria using a fire modeling PB approach.

- Section 3.4 provides the results of the NRC staff's review of the fire risk assessments used by the licensee to demonstrate the ability to meet the nuclear safety performance criteria using a FRE PB approach.

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

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

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

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

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

3.1. NFPA 805 Fundamental FPP Elements and Minimum Design Requirements

NFPA 805, Chapter 3, contains the fundamental elements of the FPP and specifies the minimum design requirements for fire protection systems and features that are necessary to meet the standard. 10 CFR 50.48(c) takes exception to three specific requirements of NFPA 805, Chapter 3, and provides alternative requirements as follows:

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

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

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

Furthermore, Section 3.1 of NFPA 805 specifically allows the use of alternatives to the NFPA 805, Chapter 3, fundamental FPP requirements that have been previously approved by the NRC, which is the authority having jurisdiction (AHJ), as denoted in the NFPA 805 standard. The licensee identified an implementation action to modify the ONS FPP to include the statement that the NRC is the AHJ (SE Section 2.9, Table 2.9-1, Item 1).

3.1.1. Compliance with NFPA 805, Chapter 3, Requirements

The licensee used the systematic approach described in NEI 04-02, Revision 2 (Reference 21), as endorsed by the NRC in RG 1.205, Revision 1 (Reference 14), to assess the proposed ONS FPP against the NFPA 805, Chapter 3, requirements.

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

The methods used by the licensee 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 FPP and Design Elements (NFPA 805, Chapter 3)," also called the B-1 Table, as "Comply."

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

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

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

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

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

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

Attachment A,Table 3.1-1, "NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix," in Attachment A to this SE, provides the specific FPP elements and minimum design requirements from NFPA 805, Chapter 3, as appropriately modified by 10 CFR 50.48(c). In addition, the table describes each fundamental FPP element from NFPA 805, Chapter 3, and identifies which of the methods the licensee used as the means for achieving compliance with the requirement.

Attachment A, Table 3.1-1 also provides the results of the NRC staff's evaluation of the licensee's compliance statement for each FPP element. 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.

For approximately 68 percent 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. In these instances, based on the validity of the licensee's statements, the NRC staff finds the licensee's compliance method/strategy acceptable.

For approximately 2 percent of the 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 FPP element as clarified.

For approximately 14 percent of the NFPA 805, Chapter 3, requirements, the licensee demonstrated compliance with the fundamental FPP element through the use of EEEEs. Based on the licensee's statement of validity provided in Tables B-1 and B-3, the NRC staff finds the licensee's statements of compliance in these instances acceptable.

Approximately 1 percent of the requirements were supplanted by an alternative that was previously approved by the NRC. In two instances, NRC approval was documented in the original August 11, 1978, FPP SE report (Reference 26), and the other two instances were approved in an NRC Exemption (ADAMS Accession No. ML012000058) dated August 21, 1989 (Reference 27). NFPA 805 allows the justification for exemptions to be carried forward in the transition to NFPA 805 as PB evaluations.

In each instance, the licensee evaluated the basis for the original NRC approval and determined that in all cases the bases were still valid. The NRC staff reviewed the information provided by the licensee 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 of the previously approved alternatives to the NFPA 805, Chapter 3, requirements, as well as the NRC staff's review of this information, the NRC staff finds the licensee's statements of compliance in these instances acceptable.

In the compliance statements for approximately 12 percent (13 of the requirements) of the NFPA 805, Chapter 3, requirements, the licensee used more than one of the above strategies to demonstrate compliance with all aspects of the fundamental FPP elements. In each of these cases, the NRC staff found the compliance statements acceptable, for the reasons outlined above.

In 11 instances, the licensee requested approval for the use of PB methods to demonstrate compliance with a fundamental FPP element. In accordance with 10 CFR 50.48(c)(2)(vii), the licensee requested specific approval be included in the license amendment approving the transition to NFPA 805. The requested PB methods pertain to the following requirements further discussed in Section 3.1.3 in this SE.

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

- Sections that do not contain any technical requirements (e.g., NFPA 805, Chapter 3, Section 3.1, and Section 3.4.5).
- Sections that are not applicable to ONS because of the following:
  - The licensee states that ONS does not have systems of this type installed (e.g., the NFPA 805, Chapter 3, Section 3.9.4 requirements for diesel-driven water fire pumps, Section 3.10 requirements for gaseous suppression systems, and Section 3.11.5 requirements for electrical raceway fire barrier systems).
  - The requirements are structured with an applicability statement (e.g., NFPA 805, Chapter 3, Section 3.4.1(a)(2) and Section 3.4.1(a)(3), wherein the determination of which NFPA code(s) apply to the fire brigade depends on the type of brigade specified in the FPP).

As documented in Attachment A, SE Table 3.1-1 and discussed above, the NRC staff evaluated the results of the licensee's assessment of the proposed ONS 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 LAR, and supplements, the NRC staff finds the RI/PB FPP acceptable with respect to the fundamental FPP elements and minimum design requirements of NFPA 805, Chapter 3, as modified by 10 CFR 50.48(c)(2), because the licensee accomplished the following:

- used an overall process consistent with NRC staff approved guidance to determine the state of compliance with each of the applicable NFPA 805, Chapter 3, requirements.
- provided appropriate documentation of ONS's state of compliance with the NFPA 805, Chapter 3, requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:
  - with the requirement directly.
  - with the intent of the requirement (or element) given adequate justification.
  - via previous NRC staff approval of an alternative to the requirement.
  - through the use of an EEEE.
  - through the use of a combination of the above methods.
  - through the use of a PB method that the NRC staff has specifically approved in accordance with 10 CFR 50.48(c)(2)(vii).

3.1.2. Identification of Power Block

The NRC staff reviewed the ONS structures identified in LAR Table I-1, "ONS Power Block Definition," 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 ONS RI/PB FPP that have additional requirements in accordance with 10 CFR 50.48(c) and NFPA 805. As stated in the LAR, power block equipment includes SSCs required for the safe and reliable operation of the station. It includes all safety-related and balance-of-plant systems and components required for operation, including radioactive waste processing and storage, the 230 kV switchyard, Keowee Dam and associated structures, and the PSW Facility and associated electrical duct banks.

This equipment does not include buildings or structures that support station staff, such as offices or storage structures, or the ventilation and support systems focused only on habitability of those structures. The NRC staff finds that the licensee has appropriately evaluated the structures and equipment at ONS, and adequately documented a list of those structures that fall under the definition of "power block" in NFPA 805.

3.1.3. Performance-Based Methods for NFPA 805, Chapter 3, Elements

Performance-Based Methods, Section 50.48(c)(2)(vii):

> The prohibition in Section 3.1 of NFPA 805 that does not permit the use of performance-based methods for the Chapter 3 fundamental fire protection program elements and minimum design criteria is not endorsed. The NRC takes this exception in order to provide licensees greater flexibility in meeting the fire protection program elements and minimum design requirements of Chapter 3 by the use of performance-based methods (including the use of risk-informed methods) described in the NFPA 805 standard. This approach is acceptable to NRC because the rule requires NRC review and approval prior to the licensee's use of those methods, and the rule sets forth criteria for evaluating the acceptability of the licensee's proposed use of performance-based methods in meeting the fire protection program elements and minimum design requirements.

Final Rule, Voluntary Fire Protection Requirements for Light Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative, (69 FR 33536, 33543, June 16, 2004) (describing Performance-Based Methods of Section50.48(c)(2)(vii)).

In accordance with 10 CFR 50.48(c)(2)(vii), a licensee may request NRC approval for use of the PB methods permitted elsewhere in the standard as a means of demonstrating compliance with the prescriptive fundamental FPP elements and minimum design requirements of NFPA 805, Chapter 3. According to 10 CFR 50.48(c)(2)(vii), an acceptable PB approach accomplishes 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 SSD capability).

In LAR Section 4.1.2.3, "NFPA 805, Chapter 3, Requirements Not Specifically Met nor Previously Approved by NRC," and Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested NRC staff review and approval of 11 PB methods to demonstrate an equivalent level of fire protection for the following NFPA 805, Chapter 3, requirements:

- Section 3.3.1.2, which concerns the use of untreated wood for use in concrete forming at ONS,
- Section 3.3.5.1, which concerns the use of wiring above suspended ceilings at ONS,
- Section 3.3.5.3, which concerns the use of low-voltage cable at ONS that does not comply with an acceptable flame propagation test,
- Section 3.3.7.1, which concerns the storage of bulk quantities of flammable gas cylinders at ONS,
- Section 3.3.12(1), which concerns collection of oil mist from the reactor coolant pump (RCP) oil system at ONS,
- Section 3.5.3, which concerns the omission of relief valves on the high-pressure service water (HPSW) and Keowee fire pumps at ONS,
- Sections 3.5.3, 3.5.16, 3.6.1, and 3.6.2, which concerns the use of fire hose stations and fire pumps at ONS that do not meet NFPA 14 and NFPA 20, respectively,
- Section 3.5.6, which concerns the use of fire pumps that have an automatic stop function at ONS,
- Sections 3.5.7, 3.5.10, and 3.5.15, which concerns the use of a fire protection system at ONS that does not have sectionalizing valves and does not meet NFPA 24,
- Section 3.5.16, which concerns the use of the fire protection fire water system at ONS that has dual purposes, and
- Sections 3.5.3 and 3.5.4, which concerns the use of a fire protection system at KHS that cannot provide 100 percent of the required flow rate and pressure at all times.

The NRC staff's review of the licensee's proposed methods is provided below.

3.1.3.1 Use of Non-treated Wood

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.3.1.2, "Control of Combustible Material," regarding the use of untreated wood for concrete forming. Specifically the licensee has requested approval of a PB method to justify the use of non-pressure impregnated or fire-retardant (untreated) wood within the ONS power block for concrete forming.

The licensee stated that the basis for the approval request of this PB method is:

- In some cases, the chemicals used in the treatment of fire-retardant wood affect concrete curing.

- A small quantity of untreated wood used for concrete forming is acceptable because the magnitude of the additive combustible material would be insignificant as compared to the total fire load in the area.

- The locations of concrete forming are generally not in close proximity to ignition sources.

- Concrete forming is for temporary use and not for permanent plant installation.

The licensee stated that the use of untreated wood for concrete forming would have no adverse impact on nuclear safety performance because (1) concrete forming is used infrequently within the ONS power block, (2) it is generally in such small quantities that it would have a negligible impact to the in-situ fire load and would be within the permissible transient fire load, and (3) if the quantity of untreated wood exceeds the permissible limits established in ONS administrative controls, a fire protection engineer (FPE) review would be conducted and result in the identification and implementation of special precautions or limitations, as necessary, on the use of the untreated wood.

By letter dated September 13, 2010, (Reference 12), the licensee stated, the proposed PB method maintains SMs and conservatisms because (1) the quantity of untreated wood used in concrete forming is minimal and the quantity is reviewed with special precautions or limitations identified as necessary in order to minimize fire risk, and (2) the precautions and limitations ensure that the quantity of these materials is maintained within the limitations and assumptions of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the introduction of untreated wood for concrete forming does not impact fire protection DID because (1) its use is administered under the ONS combustible control program, and (2) automatic or manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised since quantities of untreated wood used for concrete forming cannot be introduced such that they may challenge any elements of the FPP without appropriate compensatory measures being identified during the work review process.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the use of untreated wood for concrete forming will have no impact on the radiological release performance criteria because (1) the introduction of untreated wood does not change the conclusion of the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the existence of the small quantity of untreated wood and (2) untreated wood does not add additional radiological materials to the area or challenge systems boundaries that contain such.

The NRC staff finds that the proposed PB method:

- does not impact the NFPA 805 nuclear safety performance measures (goals, objectives, and performance criteria) because (1) the quantity of untreated wood is expected to be relatively small when compared to the total combustible loading in the area, (2) the difference in combustible loading between treated and untreated wood for small quantities exposed during a limited amount of time will not present a challenge to the fire protection features in place, and (3) administrative controls used while untreated wood is in the plant provides additional assurance of minimal impact should the untreated wood be exposed to a nearby fire,

- maintains the SMs of the licensee's analyses based on the licensee's statement that the precautions and limitations identified to minimize fire risk ensure that the quantity of untreated wood used for concrete forming is maintained within the limitations and assumptions of the Fire PRA,

- maintains fire protection DID since automatic or manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the NFPA 805 radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.3.1.2 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 SM, and maintains adequate fire protection DID.

3.1.3.2 Use of Compressed Flammable Gas Storage in the Power Block

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested NRC staff review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.3.7, "Bulk Flammable Gas Storage." The licensee has requested approval of a PB method to justify the storage of flammable gas cylinders in four locations in the AB where gas bottles are already installed. The locations identified are the chemistry labs and the post-accident monitoring instrumentation rooms.

Plant administrative controls contain specific instructions for segregating and storing compressed gas cylinders. Chemistry labs use hydrogen on a daily basis and have reserve tanks staged for continued use. The Chemistry labs and reference gases are located in the ONS AB, which also contains systems, equipment, and components important to nuclear safety. The hydrogen cylinders are stored and controlled in accordance with ONS administrative and operating procedures.

The licensee stated that the basis for the approval request of this PB method is:

- Staging of flammable gas cylinders is required in four locations, which house systems, equipment, or components important to nuclear safety. Typically, one bottle is connected to the system and the minimum number of required bottles are staged in the area for continued use.

- The flammable gas cylinders in this evaluation exist in the plant or a fire protection engineering review will be performed prior to any new installation.

- Gas cylinders staged but not in use are segregated and stored in accordance with ONS administrative procedures and design review processes.

- The flammable gas cylinders are stored in locations that do not impact equipment important to nuclear safety:

  - The Chemistry Labs are located on the 796' elevation of the Units 2 and 3 AB (Fire Zones 90 and 86).

  - The post-accident monitoring instrumentation is located on the 838' elevation of the AB in the Units 1/2 air-handling unit (AHU) and Unit 3 AHU rooms (Fire Zones 119 and 116).

By letter dated September 13, 2010, (Reference 12), the licensee stated that the storage of flammable gas cylinders in the four identified locations will have no impact on the nuclear safety performance criteria because (1) the four locations have been analyzed in the Fire PRA in the current configuration which includes the presence of the flammable gas cylinders, and (2) hydrogen fires in the four locations do not impact any targets.

Similarly, the licensee stated by letter dated September 13, 2010, (Reference 12) that the proposed PB method maintains SMs and conservatisms because (1) the four locations have been analyzed in the current Fire PRA in their current configuration which includes the presence of the flammable gas cylinders, (2) hydrogen fires in these locations do not impact any targets, and (3) the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the storage of flammable gas cylinders in the four locations does not impact fire protection DID because (1) ONS administrative controls require the introduction of a new compressed gas cylinder be evaluated by the fire hazard review process, and (2) the introduction of flammable gas cylinders does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method has no impact on the radiological release performance criteria because (1) the introduction of flammable gas cylinders in the four locations does not change the conclusion of the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the existence of the gas cylinders, and (2) flammable gas cylinders do not add additional radiological materials to the area or challenge systems boundaries that contain such.

The NRC staff finds that the proposed PB method:

- does not impact the NFPA 805 nuclear safety performance measures (goals, objectives, and performance criteria) because (1) the specific locations of the cylinders have been analyzed in the Fire PRA for the proposed configuration, (2) the bottle use and storage are controlled by procedure, (3) gas bottles are segregated by procedure and design process, and (4) hydrogen fires due to failure of the gas cylinders do not impact any targets,

- maintains the SMs of the licensee's analyses based on the licensee's statement that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,
- maintains fire protection DID since automatic or manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and
- will have no effect on the NFPA 805 radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.3.7 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 SM, and maintains adequate fire protection DID.

3.1.3.3 Use of Non-listed / Unapproved Wiring above the Suspended Ceiling

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.3.5.1 regarding wiring above suspended ceilings. The licensee has requested approval of a PB method to justify the use of existing wiring above suspended ceilings that may not comply with the requirements of this section. As described by the licensee, this concerns areas at ONS currently with suspended ceilings inside the power block consisting of offices, labs, elevator lobbies, corridors, and change rooms. The areas include:

- Control Rooms / Lobbies
- TB office areas
- AB stair and/or elevator lobbies
- AB office areas (Health Physics/Chemistry)
- AB Change Areas
- 838' elevation AB corridor

With the exception of the Control Rooms, the licensee stated that these areas are not risk significant.

The licensee stated that the basis for the approval request of this PB method is:

- Power and control cables comply with requirements of "plenum rated" equivalent or armored.
- The wiring above ceilings in offices, lobbies, and laboratories, do not pose a hazard:

- Low voltage is not susceptible to shorts causing a fire. The licensee defines that video/communication/data cables are low voltage.
- By eliminating cables with the potential shorts, this eliminates ignition sources and therefore the jacketing of cable is not relevant.
- There is no equipment important to nuclear safety in the vicinity of these cables.
- Beginning in 2006, any new cables installed and the replacement of existing cables as part of upgrades are "plenum rated."

- The installation of detection above the Control Room ceilings will promptly identify a fire thereby enhancing fire brigade response time. The installation of detection is a committed plant modification (Section 2.8.1 of this SE; Modification 5).
- New power, control or instrumentation cable installed is constructed similar to or superior to the original cable and meets the requirements of IEEE-383, "IEEE Standard for Type Test of Class IE Electric Cables, Field Splices, and Connections for Nuclear Power Generating Stations," IEEE Standard 383.

The licensee further stated that for the cabling above the suspended ceilings in the control rooms has a very low possibility of a fire due to limited combustible loading, discontinuity of combustibles, and the inherent features of the electrical circuit design. In addition, the ventilation in the control rooms is a closed-loop system, which recirculates the air where either the existing detection or the control room operators who are continuously present in the area would quickly identify the smoke. The licensee indicated that an engineering change (EC) has been developed to install detection above the suspended ceiling area in the control room. The installation of new detection above the control room ceilings can promptly identify a fire thereby enhancing fire brigade response time and minimizing the impact to fire risk.

The licensee stated that the wiring above the suspended ceiling that may not comply with the requirement of NFPA 805, Section 3.3.5.1, does not impact the nuclear safety performance criteria because (1) with the exception of the control room, wiring above suspended ceilings is not in the vicinity of nuclear safety equipment, (2) power and control cables are armor jacketed, in metallic conduit, or plenum rated, and (3) low voltage cable is not susceptible to shorts that would result in a fire.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the wiring above suspended ceiling that may not comply with the requirement of NFPA 805, Section 3.3.5.1 does not impact fire protection DID because the video/communication/data cables do not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method has no impact on the radiological release performance criteria because (1) the location of cables above suspended ceilings does not change the conclusion of the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the existence of the low-voltage cables, and (2) the cables do not add additional radiological materials to the area or challenge systems boundaries that contain such. The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) the space enclosing these cables are non-combustible, (2) the location of wiring above suspended ceilings has a minimum amount of nearby ignition sources considering the adjacent armored power and control cables, (3) the video/communication/data cables are low energy and therefore pose a low fire ignition hazard due to hot shorts, and (4) fire detection will be installed above the suspended ceilings in the control rooms,

- maintains the SMs of the licensee's analyses based on the licensee's statements that (1) power and control cables comply with requirements of "plenum rated" equivalent or armored, (2) limited unqualified low-voltage wiring, and (3) that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

The NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.3.5.1, 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 SM, and maintains adequate fire protection DID.

3.1.3.4 Use of Unqualified Video/Communication/Data Cables

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested NRC staff review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.3.5.3 regarding an acceptable flame propagation test for electric cable construction. The licensee has requested approval of a PB method to justify the use of existing wiring that may not comply with the requirements of this code section. As described by the licensee, video/communication/data cables installed at ONS are not necessarily tested in accordance with the flame propagation test requirements of IEEE-383 or any other qualification standard outlined in FAQ 06-0022 as endorsed by the NRC.

The licensee stated that the basis for the approval request of this PB method is:

- Power and control cable installed is constructed similar to or superior to the original cable and meets the requirements of IEEE-383.

- All new power, control or instrumentation cable installed will be constructed similar to or superior to the original cable and will meet the requirements of IEEE-383 or plenum rated

(NFPA 262, "Standard Method of Test for Flame Travel and Smoke of Wires and Cables for Use in Air Handling Spaces").

- Video/communication/data cables are low voltage and not susceptible to cause shorts and fires.

By letter dated September 13, 2010, (Reference 12), the licensee stated that acceptable cable construction qualifications will be included in the Power Generation Electrical Discipline Design Criteria Manual. The licensee identified an implementation action to modify this manual to add a specific line item that video/communication/data cables shall be plenum-rated and/or tested in accordance with IEEE 383-1974, IEEE 1202-1991, CSA (Canadian Standards Association) 22.2 No. 0.3, NFPA 262, UL (Underwriters Laboratory) 44, UL 83, UL 1581, UL 1666, or UL 1685 as accepted in FAQ 06-0022 (SE Section 2.9, Table 2.9-1, Item 34). Electrical wiring, including video, phone, and communications, installed above suspended ceilings shall be rated for plenum use, routed in metallic conduit, routed in cable tray with solid metal top and bottom covers, or armored cable.

By letter dated September 13, 2010, (Reference 12) the licensee stated that the proposed PB method maintains SMs and conservatisms because the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the video/communication/data cables that may not comply with the requirement of NFPA 805, Section 3.3.5.3 do not impact fire protection DID because (1) cable flame spread criteria are controlled by the licensee's design processes, and (2) video/communication/data cable construction does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method has no impact on the radiological release performance criteria because (1) the construction of cables does not change the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the existence of the cables, and (2) cables do not add additional radiological materials to the area or challenge systems boundaries that contain such.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) existing video/communication/data cables do not constitute a significant fire hazard, (2) future installation of cable will require appropriately approved flame spread criteria, (3) adjacent power and control cables are stated by the licensee to have acceptable flame spread qualities and therefore do not contribute significantly to the hazard, and (4) the licensee will require that all future installations of cable will comply with NFPA 805, Section 3.3.5.3,

- maintains the SMs of the licensee's analyses based on the licensee's statements that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.3.5.1, 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 SM, and maintains adequate fire protection DID.

3.1.3.5 Allow Reactor Coolant Pump (RCP) Oil Mist Without Collection

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.3.12(1) regarding the ONS RCP oil collection system. The licensee has requested approval of a PB method to justify not collecting oil mist resulting from pump/motor operation. This system was designed and was reviewed in accordance with 10 CFR Part 50, Appendix R, Section III.O to collect leakage from pressurized and nonpressurized leakage sites in the RCP oil system.

This however did not include collection of oil mist as a result of RCPs pump/motor operation. As stated in Attachment L of the LAR, oil misting is not leakage due to equipment failure, but an inherent occurrence in the operation of large rotating equipment. It is normal for large motors to lose some oil through seals and the oil to potentially become 'atomized' in the ventilation system. This atomized oil mist can then collect on surfaces in the vicinity of the RCP, as the pump design is not completely sealed to permit airflow for cooling. The oil mist resulting from normal operation will not adversely impact the ability of a plant to achieve and maintain SSD even if ignition occurred.

The licensee stated that the basis for the approval request of this PB method is:

- The oil collection system is designed to collect leakage from pressurized and nonpressurized leakage sites in the RCP oil system.

- Oil misted from normal operation is not leakage; it is normal motor oil consumption.

- Oil misted from normal operation does not significantly reduce the oil inventory. The oil released as misting does not account for an appreciable heat release rate or accumulation near potential ignition sources or non-insulated reactor coolant piping.

- The RCPs use a synthetic oil of higher flash point, approximately 450 °F.

- There are redundant RCPs and they are not required to achieve or maintain fire SSD.

The licensee stated that the lack of an oil mist collection system for the RCPs does not impact the nuclear safety performance criteria because (1) oil mist does not significantly contribute to a fire heat release rate, (2) the synthetic oil used has a higher flashpoint of approximately 450°F, and (3) the equipment is not required for SSD.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) the oil mist resultant from normal operation will not adversely impact the ability of a plant to achieve and maintain fire SSD even if ignition occurred, (2) the RCPs are not required to achieve and maintain fire SSD, and (3) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the oil mist from normal pump operation does not impact fire protection DID because (1) the RCP oil collection systems are controlled by the licensee's design processes, (2) oil misting does not result in compromising automatic or manual fire suppression functions, and (3) does not impact the post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method has no impact on the radiological release performance criteria because (1) the entire RB during power operations is an environmentally sealed radiological area, (2) the potential for oil mist from the RCPs does not change the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the existence of the misting, and (3) the oil mist does not add additional radiological materials to the area or challenge systems boundaries that contain such.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) RCP oil collection fire does not impact the ability to achieve SSD, (2) RCPs are not components identified as necessary for SSD, (3) oil mist is not in the immediate proximity to an ignition source or non-insulated reactor coolant piping, (4) oil collection is provided for in all areas where leakage from pressurized and non-pressurized leak sites exist in the oil system, (5) the oil collection system has been seismically qualified to prevent oil spillage reaching areas which may be above the flash point of the lubricating oil, and (6) upper and lower oil pots have been modified with a shield to catch oil and carry it to a tank to reduce fire potential,

- maintains the SMs of the licensee's analyses based on the licensee's statements that (1) the RCPs are not required to achieve and maintain SSD following a fire in the vicinity and (2) the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.3.12(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 SM, and maintains adequate fire protection DID.

3.1.3.6 Fire Pump Circulation Relief Valves

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.5.3 regarding fire pump design and installation compliance with NFPA 20, Section 5.11. The licensee has requested approval of a PB method to justify the omission of circulation relief valves on the HPSW and Keowee fire pumps.

NFPA 805 has incorporated the requirement for circulation relief valves as specified in NFPA 20, Section 5.11.1.2, "The valve shall provide flow of sufficient water to prevent the pump from overheating when operating with no discharge."

The HPSW pumps are not standard fire pumps but large industrial pumps and subsequently built to different original standards. The HPSW pumps are utilized as the ONS fire pumps. These pumps have a dual function: to supply water for fire suppression and to provide sealing/cooling water to various components. The two electric-driven HPSW pumps are provided with cooling lines designed to cool the pump motor and also provide some flow to prevent the pump from overheating.

The HPSW pumps do not generally operate without flow. When the HPSW pump operates in a fire event, the ONS fire brigade response procedure instructs that a deluge system or hydrant be opened if the flow on the system is assessed less than 1,450 gallons per minute (gpm) in order to maintain greater than the manufacturer's minimum flow.

The Keowee fire pump is not provided with an automatic relief valve. However, the Keowee pump automatically shuts down when flow stops such that it will not run at shutoff pressure.

The licensee stated that the basis for the approval request of this PB method is:

- The HPSW pumps have procedures in place to ensure there is acceptable flow to prevent overheating therefore circulation relief valves are not necessary.

- The Keowee fire pump has an automatic shutdown feature to prevent overheating therefore circulation relief valve is not necessary.

The licensee stated that omission of circulation relief valves on the HPSW and Keowee fire pumps does not impact the nuclear safety performance criteria because the pumps are operable and measures are in place to ensure the pumps do not overheat.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) alternative measures are provided to ensure the HPSW and Keowee fire pumps will not overheat and (2) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the omission of circulation relief valves on the HPSW and Keowee fire pumps does not impact fire protection DID because (1) procedures ensure the HPSW pumps do not overheat by manually opening an excess flow path, (2) the Keowee pump is maintained free of overheating by the auto-stop feature, and (3) the lack of circulation relief valves does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability since the pumps are functional and measures are in place to ensure the pumps do not overheat.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the omission of circulation relief valves on the HPSW and Keowee fire pumps has no impact on the radiological release performance criteria because (1) the features of the fire pumps do not change the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the lack of circulation relief valves, and (2) the fire pumps provide radiological clean water to the HPSW system and do not cross-tie to contaminated water piping.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria because the (1) Keowee pump is designed using shutoff interlocks to prevent pump damage, (2) the HPSW pumps are procedurally controlled during operations to ensure the pumps do not overheat, and (3) the HPSW pumps are addressed in the brigade procedures to ensure at least a minimum flow is maintained,

- maintains the SMs of the licensee's analyses based on the licensee's statements that (1) alternative measures are provided to ensure the HPSW and Keowee fire pumps will not overheat, and (2) the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since (1) procedures ensure the HPSW pumps do not overheat by manually opening an excess flow path, (2) the Keowee pump is maintained free of overheating by the auto-stop feature, and (3) automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Section 3.5.3 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 SM, and maintains adequate fire protection DID.

3.1.3.7 Insufficient Pressure for Reactor Building Hose Stations

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff 's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Sections 3.5.3, 3.5.16, 3.6.1, and 3.6.2 regarding fire protection water supply to the RBs. The licensee has requested approval of a PB

method to justify the existing ONS fire pumps, standpipes, and water mains for the RBs that do not meet certain aspects of NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection." Approval is requested for the use of the low-pressure service water (LPSW) system to supply the RB hose stations/standpipes at less than the required pressure for the RB hose stations/standpipes.
The pressure at various standpipe elevations does not meet the minimum pressure prescribed by NFPA 14, "Standard for the Installation of Standpipe and Hose Stations." The licensee stated that the RB hose stations cannot meet the system demands of pressure and flow as required for fire fighting with hose stations in the RBs.

The hydraulic calculations indicate that a flow of 100 gpm, provides the residual pressure of approximately 21 pounds per square inch (psi) at the highest elevation inside the RB and 56 psi in the lower elevations. Both of these values are less than the required pressure of 65 psi (Code of Record: NFPA 14 - 1978).

The licensee stated that the basis for the approval request of this PB method is:

- Quick detection and suppression of a fire by the fire brigade is generally an inherent assumption in the Fire PRA, but in the case of a fire in the RB, no credit for manual suppression is given in the containment Fire PRA model.

- The licensee committed to the NRC to use the LPSW to supply the hose stations/standpipes in the RBs. The LPSW pumps were never designed to be able to provide the required pressures for the hose stations in RB.

- The fire hazards in the RBs are minimized and higher pressure for hose station operations are provided at the lower elevations where there is a higher concentration of combustibles.

- The ONS fire brigade has low-pressure nozzles available and is trained on their use.

- There are six additional carbon dioxide fire extinguishers staged at each RB personnel hatch.

- The primary purpose of the hose stations in containment is to act as back-up manual suppression during non-power operation (NPO) modes.

By letter dated November 19, 2010 (Reference 52), the licensee stated that during power operations, the expected response to a fire in containment is not to enter the RB and let the fire burn out either via fuel consumption or lack of oxygen. The fire brigade is trained and standard operating procedures direct them to preferably only enter an area to fight a fire with a charged hose line. The charged hose line would be connected to the HPSW system via hose stations in the AB with an alternative connection supplied from a yard fire hydrant. The HPSW system is the normal plant fire protection water system. There is a hose station located in the immediate area adjacent to each personnel hatch in the AB. During NPO modes a fire could be attacked using the existing RB hose stations (if at the fire brigade's discretion that the fire is within the capabilities of the LPSW hose stations), or with a hose line connected to the HPSW system as described above, or with a hose line connected to the HPSW system through the yard hydrants if the exterior equipment hatch is open.

The licensee stated that the fire brigade will develop a Standard Operating Guide (SOG) for fighting a fire in the RB. Training is already performed on tactics for fighting fires of this nature but training will be reinforced with a new SOG. The Fire Brigade Administrator will review the Fire Plans to determine if enhancement is necessary. This item is being tracked by the ONS Corrective Action Program and is an implementation item (Section 2.9, Table 2.9-1, Item 36).

By letters dated September 13, 2010 and November 19, 2010, (References 12 and 52) the licensee stated that low pressure in the RB hose stations does not impact the nuclear safety performance criteria because (1) in the case of a fire in the ONS RB, no credit for manual suppression is given in the containment Fire PRA model, and (2) the RBs are not accessed during power operation unless in an emergency. The fire brigade does not use the hose stations located in containment in the event of a significant fire in the RB.

By letter dated November 19, 2010, (Reference 52), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) the low pressure in the RB hose stations are provided for limited use by trained fire brigade members, (2) alternative equipment such as fire extinguishers and charged hose station from outside the RB can be used to fight a fire, and (3) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the RB hose stations having insufficient pressure do not impact fire protection DID because (1) administrative controls to prevent fires are still in place, (2) the availability of alternate suppression means for the RB does not result in compromising automatic or manual fire suppression functions, and (3) post-fire SSD capability remains unaffected because no credit is provided for these hose stations.

By letters dated September 13, 2010 and November 19, 2010, (Reference 12 and 52) the licensee stated the low-pressure hose stations in the RBs has no impact on the radiological release performance criteria because (1) the entire RB in which the subject fire hose stations are located is in an environmentally sealed radiological area, and (2) the limited pressure of fire hose stations in the RBs has no impact on the radiological release performance criteria.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria because (1) the containment portion of the Fire PRA does not take credit for manual suppression, (2) the analysis assumes the hose stations are not available or immediately accessible during power operations, (3) there is no increased risk or change in delta CDF or LERF as manual suppression is not credited in the containment Fire PRA, and (4) the RB FREs indicate that based on the containment configuration, limited exposed combustibles, volume of the containment, and slow or limited fire propagation will reduce the impact of fire in the containment,

- maintains the SMs of the licensee's analyses based on the licensee's statements that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised,

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Sections 3.5.3, 3.5.16, 3.6.1, and 3.6.2 requirements because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.3.8 Fire Pump Automatic/Remote Stop Feature

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.5.6 regarding automatic stop features on fire pumps, and Section 3.5.3 regarding NFPA 20 requirements for fire pumps. The licensee has requested approval of a PB method to justify the automatic stop function on existing ONS HPSW pumps (fire water) and the Keowee fire pump, and to justify remote operation of the HPSW pumps from the control room. Both of these features are contrary to the NFPA 20, "Standard for the Installation of Stationary Pumps for Fire Protection."

As described by the licensee:

- The HPSW pumps can be stopped automatically by the level switches in the elevated water storage tank (EWST), manually in the control room, locally at the switchgear breaker, and locally at the pump.

- The HPSW system has a jockey pump to maintain normal system pressure during service water (SW) loads. If the pressure falls below the setpoint at which the jockey pump cannot maintain the HPSW system, the altitude valve, located at the base of the EWST, opens to supply system pressure and flow. The HPSW pump(s) stop based upon set fill level of the EWST.

- A start/run/off/base/standby switch is provided in the control room on auxiliary control board, 1AB3, for the HPSW pumps (both A and B). This permits the pumps to be manually operated in order to avoid pressure disruptions in the system.

- The Keowee fire pump stops automatically based on sensing low/no water flow.

The licensee stated that the basis for the approval request of this PB method is:

- The NRC previously accepted the use of the HPSW system for fire protection use.

- The Keowee fire pump has an automatic shut off on low/no flow. The system is routinely tested to demonstrate operability.

- When the pumps are operating, they are monitored by trained operators who can control the pumps as necessary.

The licensee stated that remotely/automatically stopping the fire pumps does not impact the nuclear safety performance criteria because (1) nuclear safety is not affected, and (2) the pumps are available and monitored by trained operators.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) the fire pumps operate automatically and are monitored and controlled by trained operators, and (2) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the means of remotely/automatically stopping the fire pumps does not impact fire protection DID because (1) pump controls are maintained by procedures to ensure pumps are available by taking manual control, (2) suppression is maintained by the availability of a redundant fire pump, and (3) means are available to ensure fire pumps are functional during a fire event, which does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that remotely/automatically stopping the fire pumps has no impact on the radiological release performance criteria because (1) the features of the fire pumps do not change the radiological release evaluation performed for each fire zone that potentially contaminated water is contained and smoke is monitored since fire brigade control of water runoff and smoke is not hindered because of the lack of auto-stop feature, and (2) the fire pumps provide radiologically clean water to the HPSW system and do not cross-tie to contaminated water piping.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) the pumps are available and monitored by trained operators under the controls of procedures, (2) fire brigade operations recognize proper fire water supply conditions, and (3) monitoring programs will be in place to ensure proper operation of the fire water supply pumps,

- maintains the SMs of the licensee's analyses based on the licensee's statements that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Sections 3.5.6 and 3.5.3 requirements because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.3.9 KHS Fire Main and Standpipe Use

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Sections 3.5.7, 3.5.10, and 3.5.15 regarding yard loop connections, yard loop design, and hydrant connections. The licensee has requested approval of a PB method to justify the omission of the requirement for a fire main loop and fire hydrants at the KHS .

KHS does not have a fire main loop or fire hydrants. However, KHS has two outside fire hose standpipes that use fire hydrant appurtenances as their controlling devices. The outside fire hose standpipes (fire hydrants) are equipped with two 2-1/2" hose connections (no other/larger connections).

The KHS underground fire piping consists of an 8-inch pipe that supplies the transformer water spray system and a 4-inch pipe, which tee's and supplies the two-yard hydrants/outside fire hose standpipes. There is no yard fire loop in accordance with the elements on NFPA 805 Sections 3.5.7, 3.5.10, and 3.5.15. In addition, in accordance with NFPA 24, Sections 5.2.1 and 13.1, the piping servicing the fire hydrants is not provided with piping greater than 6 inches in diameter.

These devices do not meet the requirements for fire hydrants as they are supplied via a 4-inch underground main. It can be best determined that the two locations with fire hydrants' appurtenances are used as external fire hose standpipes because the fire hydrant offers a drain function of the barrel therefore no freeze protection is required.

Revision of the Design Basis Specification for Fire Protection to state that the KHS fire hydrants are not designed, nor intended to function as fire hydrants but to act as external automatic wet standpipes for fire brigade/fire department response as required is an implementation item (SE Section 2.9, Table 2.9-1, Item 26).

The licensee stated that the basis for the approval request of this PB method is:

- A yard fire loop is not required given the fire protection water required at the KHS .
- The fire hydrants are installed as fire hose standpipes for the fire brigade and act as a wet standpipe.

The licensee stated that the layout of the fire service main at the KHS does not impact the nuclear safety performance criteria because (1) the Keowee fire protection system is not required for the overall nuclear fire safety at ONS, (2) a fire is not simultaneously postulated at ONS and KHS and (3) KHS is the emergency power for ONS, such that in the event of a loss-of-offsite power, KHS provides the power to shutdown ONS, while if KHS were unavailable, the licensee would proceed on a controlled shutdown using normal power.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) the layout of the fire service main at KHS does not impact fire protection and the fire hydrants are functionally automatic wet standpipes for fire brigade/fire department operations, and (2) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the fire service main at KHS does not impact fire protection DID because (1) administrative controls to prevent fires are not affected, (2) suppression is maintained by the inherent design and objectives of the Keowee fire service main, and (3) the presence of this fire service main does not compromise automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the layout of the fire service main at KHS has no impact on the radiological release performance criteria because there are no radiological concerns at the KHS plant location and therefore no capability to influence a potential radiological release.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) KHS is not required for the overall nuclear fire safety at ONS, (2) the fire service main layout at KHS adequately provides fire suppression water to the limited demands of the hydrant standpipes and transformer deluge systems, (3) fire in these locations do not affect the nuclear safety performance of the power block, and (4) KHS is the emergency power for ONS, such that in the event of a loss-of-offsite power, KHS provides the power to shutdown ONS, while if KHS were unavailable, the licensee would proceed on a controlled shutdown using normal power,

- maintains the SMs of the licensee's analyses based on the licensee's statements that the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since (1) administrative controls to prevent fires are not affected, and (2) automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) because the layout of the fire service main at Keowee has no impact on fire suppression activities.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS in lieu of the corresponding NFPA 805, Sections 3.5.7, 3.5.10, and 3.5.15 requirements because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.3.10 Use of Dual Purpose Fire Protection Water Supplies

In LAR Attachment L, "NFPA 805, Chapter 3, Requirements for Approval," the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Section 3.5.16 regarding dedicated fire protection water supply system. The licensee has requested approval of a PB method to justify the use of the ONS and Keowee service water (SW) systems for purposes other than fire protection water supply. The HPSW system is used for dual purposes including fire protection (suppression systems, hose stations, and fire hydrants) and SW uses including supplying bearing lubrication or cooling water to the condenser circulation water pumps and motors, the primary instrument air compressor, the leak rate test compressors, and backup cooling water to the turbine driven emergency feedwater (EFW) pump oil coolers and the high-pressure injection (HPI) pump motors. In addition, the hydrants and/or hose stations may be used for other functions such as wash down and truck/tank filling at ONS.

As described by the licensee, this usage of yard hydrants and standpipes would require control room notification of fire protection system water for plant evolutions other than fire protection under the following conditions: (1) the Work Control Center Senior Reactor Operator is notified of the evolution, (2) fire brigade procedures provide for steps to secure non-essential use of the fire water system immediately if a fire occurs, and (3) the HPSW pump capacity far exceeds the largest fire suppression water demand.

The licensee stated that the use of fire protection water for these non-fire protection system water demands would have no adverse impact on the ability of the fire protection water supply system to provide required flow and pressure based on two redundant 6,000 gpm HPSW pumps. The largest suppression system demand is the Unit 2 TB mezzanine system, which requires 2,723 gpm, plus 1,000 gpm fire hose allowance and 318 gpm additional SW for a total demand of 4,041 gpm. Assuming the maximum of 500 gpm non-fire-related system flow, there is still approximately 1,500 gpm of margin, with just one pump in operation, in excess of the required HPSW system demands. The licensee concluded that neither the flow and pressure available to any automatic water based suppression system, nor the manual fire suppression demands when needed, will be adversely impacted by the proposed change since the non-fire protection water demand would be secured before hose streams were used.

The Keowee SW system is used for dual purposes including fire protection (suppression systems and hose stations) and SW uses including dilution flow, supplying cooling to air compressors and heating ventilation air conditioning (HVAC) units, and for tank usage. In addition, the hose stations may be used for other functions such as wash down and truck/tank filling. The SW demands are generally taken before the fire pump. The largest SW demand is the dilution flow line that has a valve that automatically closes upon actuation of the fire pump to allow sufficient flow to the fire pump.

The licensee stated that the basis for the approval request of this PB method is:

- The HPSW system has excess capacity.
- The Keowee SW system has an automatic valve to cease high SW flow demands in the event of the fire pump start.
- Appropriate personnel are notified when using the HPSW system.
- Fire brigade response procedure includes a step to make a public address announcement to discontinue use of the HPSW system for non-essential purposes.

The licensee stated that the use of the HPSW and Keowee SW systems for non-fire protection uses does not impact the nuclear safety performance criteria because (1) the HPSW system has excess capacity to supply the demands of the HPSW system above the greatest sprinkler system demand, and (2) the Keowee SW system has a valve that automatically closes upon actuation of the fire pump.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the proposed PB method maintains SMs and conservatisms because (1) the HPSW system has excess capacity to supply the demands of the HPSW system above the greatest sprinkler system demand, (2) the Keowee SW system has a valve that automatically closes upon actuation of the fire pump, and (3) the PB method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA.

By letter dated September 13, 2010, (Reference 12), the licensee stated that use of the HPSW system for non-fire protection uses does not impact fire protection DID because (1) the Work Control Center Senior Reactor Operator is notified of the evolution and fire brigade procedures provide for steps to secure non-essential use of the fire water system immediately if a fire occurs, (2) suppression is maintained by excess capacity, operational guidance, and automatic equipment functions to maintain sufficient fire fighting water, and (3) the use of the HPSW pumps and Keowee, SW system do not compromise automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire SSD capability.

By letter dated September 13, 2010, (Reference 12), the licensee stated that the use of the HPSW system for non-fire protection uses, including the use of hydrants and hose for purposes other than fire, has no impact on the radiological release performance criteria because (1) there are no radiological hazards at Keowee and (2) the HPSW system is radiologically clean and does not cross-tie to contaminated water piping.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) the licensee's statements that the HPSW has excess capacity to supply the demands of the greatest sprinkler system demand, (2) the Keowee SW system has an automatic valve to cease high SW flow if the fire pump starts, and (3) plant-wide fire notification measures to stop non-essential water use are in place,

- maintains the SMs of the licensee's analyses based on the licensee's statements that (1) that the HPSW system has excess capacity to supply the demands of the HPSW system above the greatest sprinkler system demand, (2) the Keowee SW system has a valve that automatically closes upon actuation of the fire pump, and (3) the method does not change the assumptions and limitations of the analytical methods used in the development of the Fire PRA,

- maintains fire protection DID since (1) the HPSW pumps have the excess capacity to supply the demands of the HPSW system in addition to the greatest sprinkler system demand, and (2) both systems automatic and manual fire suppression functions, fire protection for systems and structures, and post-fire SSD capability are not compromised, and

- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on suppression activities inside the radiation-controlled area (RCA).

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at ONS and Keowee in lieu of the corresponding NFPA 805, Section 3.5.16 requirement because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.3.11 KHS Fire Protection Fire Pump

By letter dated November 19, 2010, (Reference 52) the licensee requested the NRC staff's review and approval of a PB method to demonstrate an equivalent level of fire protection for the requirement of NFPA 805, Sections 3.5.3, and 3.5.4 regarding the Keowee fire pump. The licensee has requested approval of a PB method to justify the omission for the installation of a second fire pump at Keowee.

Keowee is provided with one electric motor-driven fire pump. There is no secondary/back-up fire pump. The existing pump is installed in accordance with NFPA 20 and is capable of providing the required flow and pressure to the Keowee hose stations. Keowee is spatially separated from any other Oconee power block areas by approximately 3,000 feet and is an extension of the Oconee "power block" as Keowee is used as emergency power.

Keowee is a credited SSD system; however a fire at Keowee does not impact the SSD capability at Oconee using other credited power systems. A fire at Oconee is not postulated concurrent with a fire at Keowee; therefore, a fire at Oconee does not impact the ability of Keowee to provide emergency power.

The licensee stated that the basis for the approval request of this PB method is:

- A fire at Keowee does not impact the SSD capability at ONS.

- A fire at ONS is not concurrent with a fire at Keowee.

- Keowee is a separate structure located a significant distance away from any other Oconee power block structures.

- The main purpose of the fire pump at Keowee is to supply the Keowee fire hose stations.

- Compensatory measures are provided in the event the Keowee fire pump is out of service.

The NRC staff finds that the proposed PB method:

- does not impact the nuclear safety performance measures (goals, objectives and performance criteria) because (1) a fire is not simultaneously postulated at Oconee and Keowee (2) Keowee is the emergency power source for Oconee. If a fire at Keowee were to render it unavailable, Oconee would proceed to shutdown using normal power, and (3) a fire at Keowee does not impact the SSD capability at ONS.

- maintains the SMs of the licensee's analyses based on the licensee's statements that; (1) the single electric fire motor-driven pump at Keowee does not impact "SSD" fire protection for the "power block" or power production areas of the turbine/auxiliary/reactor buildings or the SSF, and (2) the fire pump is used to supply the flow and pressure requirements to the Keowee fire hose stations and has been evaluated in accordance with NFPA 20.

- maintains fire protection DID since (1) a fire at Keowee does not impact the SSD capability of ONS, and (2) ONS has compensatory measures in place in the event the Keowee fire pump is out of service.
- will have no effect on the radiological release performance measures (goals, objectives, and performance criteria) since there will be no impact on suppression activities inside the RCA.

In accordance with 10 CFR 50.48(c)(2)(vii), the NRC staff finds the proposed PB method acceptable for application at Keowee in lieu of the corresponding NFPA 805, Sections 3.5.3 and 3.5.4 requirements because it satisfies the performance goals, performance objectives, and performance criteria specified in NFPA 805 related to nuclear safety and radiological release, maintains sufficient SM, and maintains adequate fire protection DID.

3.1.4. Conclusion for Section 3.1

The NRC staff reviewed ONS's RI/PB FPP for compliance with each of the requirements of NFPA 805, Chapter 3, 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 finds the ONS RI/PB FPP acceptable with respect to the fundamental FPP elements and minimum design requirements of NFPA 805, Chapter 3, as modified by 10 CFR 50.48(c)(2), because the licensee accomplished the following:

- Used an overall process consistent with NRC staff-approved guidance to determine the state of compliance with each of the applicable NFPA 805, Chapter 3, requirements.
- Provided appropriate documentation of ONS's state of compliance with the NFPA 805, Chapter 3, requirements, which adequately demonstrated compliance in that the licensee was able to substantiate that it complied:
  - With the requirement directly.
  - With the intent of the requirement (or element) given adequate justification.
  - Via previous NRC staff approval of an alternative to the requirement.
  - Through the use of an EEEE.
  - Through the use of a combination of the above methods.
- Used PB methods that the NRC staff has specifically approved in accordance with 10 CFR 50.48(c)(2)(vii).

3.2. Nuclear Safety Capability Assessment (NSCA) Methods

NFPA 805 is a PB fire protection standard that allows engineering analyses to be used to show that FPP features and systems provide sufficient capability to meet the regulatory requirements. Specifically, Section 2.4, "Engineering Analyses," states the following:

> Engineering analysis is an acceptable means of evaluating a FPP 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.

NFPA 805, Chapter 1, defines the nuclear safety goal, objectives, and performance criteria that the FPP must meet as follows:

Nuclear Safety Goal

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

Nuclear Safety Objectives

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

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

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

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

Nuclear Safety Performance Criteria

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

(a) Reactivity Control. Reactivity control shall be capable of inserting negative reactivity to achieve and maintain subcritical conditions. Negative reactivity inserting shall occur rapidly enough such that fuel design limits are not exceeded.
(b) Inventory and Pressure Control. With fuel in the reactor vessel, head on and tensioned, inventory and pressure control shall be capable of controlling coolant level such that subcooling is maintained for a [pressurized water reactor] (PWR) and shall be capable of maintaining or rapidly restoring reactor water level above top of active fuel for a [boiling water reactor] (BWR) such that fuel clad damage as a result of a fire is prevented.
(c) Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.
(d) Vital Auxiliaries. Vital auxiliaries shall be capable of providing the necessary auxiliary support equipment and systems to assure that the systems required under (a), (b), (c), and (e) are capable of performing their required nuclear safety function.
(e) Process Monitoring. Process monitoring shall be capable of providing the necessary indication to assure the criteria addressed in (a) through (d) have been achieved and are being maintained.

3.2.1. Compliance with Nuclear Safety Capability Assessment Methods

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

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

This section of the SE evaluates the conformance of licensee's methodology to the first three of the above-listed topics. SE Section 3.5 addresses the assessment of the fourth topic.

RG 1.205, Revision 1 (Reference 14) endorses with exceptions and clarifications, NEI 04-02, Revision 2 (Reference 21). In addition, when applied in conjunction with RG 1.205, Chapter 3 of industry guidance document NEI 00-01, "Guidance for Post-Fire SSD Circuit Analysis" Revision 1, (Reference 56) and Revision 2 (Reference 28) provides an acceptable deterministic methodology. This NRC-endorsed method documents in a table format (i.e., NEI 04-02, Table B-2, "NFPA 805 Chapter 2 – Nuclear Safety Transition – Methodology Review") the licensee's comparison of its post-fire SSD analyses from its existing fire protection licensing basis to the guidance in NEI-00-01, Chapter 3. The NRC staff reviewed LAR Section 4.2.1, "Nuclear Safety Capability Assessment Methodology Review," and Attachment B, "NEI 04-02 Table B-2 – NSCA Methodology Review," against these guidelines.

The licensee states that NSCA were performed on a fire area basis. Once the systems needed to achieve and maintain safe and stable conditions in accordance with NFPA 805 were identified, components needed to ensure the capability of these systems to achieve the nuclear safety performance criteria of NFPA 805 were identified and a comprehensive list of equipment, referred to by the licensee as the SSEL was developed. In addition to components required to achieve the nuclear safety performance criteria of NFPA 805, the SSD equipment list (SSEL) includes components in system flow paths that require operation or repositioning to allow the system to function, and components that could spuriously operate and impair SSD. Based on the cables and components present in the fire area of concern, an assured success path is determined. The fire area analysis methodology considers the occurrence of multiple fire-induced failures and multiple spurious actuations.

For the majority of the NEI 00-01 attributes listed in LAR Attachment B, the licensee stated that the approach used to conduct the post-fire SSD analyses aligns with the NEI 00-01, Revision 1 (Reference 56) guidance. However, there were several attributes for which the licensee stated that they either align with the intent of the NEI 00-01 guidance or do not align with the NEI 00-01 guidance. Table 3.2-1, "Nuclear Safety Capability Assessment Method Review," in Attachment B to this SE, identifies each applicable NEI 00-01 guidance section, documents whether the licensee stated that it met the NEI 00-01 guidance, or provided justification for meeting the intent of that guidance or not meeting that guidance, and presents the NRC staff's evaluation of the acceptability of the licensee's justification.

The NEI 00-01 guidance, as endorsed in RG 1.205, is only one acceptable means to demonstrate compliance. Therefore, the NRC staff reviewed the instances where the licensee deviated from NFPA 805 standard. The NRC staff determined that, in all cases in which the licensee stated to have met the intent of the NEI 00-01 guidance, the alternative methodology used by the licensee was an acceptable means to meet the NFPA 805 requirement. For instance, Section 3.1.1.4 of NEI 00-01 states that cables and equipment required for alternative shutdown must be independent of the fire area of concern. The licensee aligns with the intent of this guidance through the transfer of control to the SSF, which isolates required systems and equipment from the effects of a fire for the fire areas of concern. Following transfer of control to the SSF, the equipment credited for an SSF shutdown meets the intent of the guidance provided in NEI 00-01. Also, although Section 3.1.2.2 of NEI 00-01 states that RCS pressure should be controlled by controlling the rate of charging/makeup to the RCS, the licensee states that pressure control may be accomplished utilizing reactor makeup from the SSF or injection from HPI in conjunction with pressurizer heaters, safety relief valves, pressure operated relief valve (s) (PORV's), RCS loop high point vent valves, or reactor head vent valves and controlling decay heat removal rates. Since an assured RCS pressure success path has been determined for each fire area, the intent of the guidance provided in NEI 00-01 is met.

Specific cases where the licensee states that it does not align with the NEI 00-01 guidance include the following:

1. <u>Sections 3.3.1.7, 3.3.3.3, and 3.5.2.4 of NEI 00-01 - Common Power Supplies</u>

If the upstream protection device (i.e., feeder breaker or fuse) of a required power supply is not properly coordinated with downstream (load) protection devices, a fire induced fault (e.g., short to ground) affecting any of the load circuits could cause the upstream (feeder) breaker to trip prior to causing a trip of the individual load protection device. This scenario, which is referred to as the "Common Power Supply Concern," would result in a loss of electrical power to all circuits connected to the affected power supply, including those circuits relied on to achieve the nuclear safety performance criteria.

Section 2.4.2.2.2 of NFPA 805 states that circuits whose fire induced failure could cause the loss of a power supply required to achieve the nuclear safety performance criteria shall be identified. The evaluation performed to demonstrate conformance to this criterion is referred to as a "Coordination Study." Specific issues to be considered in the study are discussed in Sections 3.3.1.7, 3.3.3.3, and 3.5.2.4 of NEI 00-01. In the licensee's October 31, 2008 (Reference 4) submittal the licensee stated that it had assumed that electrical protection devices were properly coordinated and did not consider the impact of inadequate breaker coordination when selecting cables. The NRC staff also noted that the licensee's existing coordination study does not include all power supplies required to achieve the nuclear safety objectives of NFPA 805 during power and non-power operations.

By letter dated August 3, 2009 (Reference 8), the licensee stated that the existing coordination study currently relied on in the LAR would require further enhancement to meet Section 2.4.2.2.2 of NFPA 805 and that a revised breaker coordination study was underway. By letter dated September 13, 2010 (Reference 12) the licensee stated that the revised breaker coordination study had been completed and identified modifications to four breakers that have an overall risk increase due to their lack of coordination with the upstream protective device.

The four breakers are being modified to maintain the Fire PRA risk profile reported in the LAR (see SE Section 3.4 a more detailed discussion). The plant modifications are described in SE Section 2.8.1 and the licensee stated that appropriate compensatory actions will be implemented until the item is fully resolved.

In the cited supplementary information, the licensee also states that the revised coordination study meets the requirements of NFPA 805, Section 2.4.2.2.2(a), for circuits that share a common power supply with circuits required to achieve the nuclear safety performance criteria. The results of the coordination study will be documented in the NSCA, NPO Pinch Point Analysis, and the Fire PRA. Incorporating all related non-coordinated information in the NSCA and NPO Pinch Point Analysis, and updating the Fire PRA model, to include the results of the breaker coordination study is an implementation item (SE Section 2.9, Table 2.9-1, Item 33). Updating the breaker coordination study to include all new NFPA 805 SSEL-related power supplies (i.e., PSW) for power and non-power operations, and defining additional plant modification if necessary to ensure that the assumptions of the Fire PRA and NSCA remain valid, is an implementation item (SE Section 2.9, Table 2.9-1, Item 44).

Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach has adequately addressed the issue of inadequate breaker coordination and that the licensee's approach will provide the required electrical protection under NFPA 805, Section 2.4.2.2.2.(a) upon completion of the identified plant modifications and implementation items.

2. Section 3.5.1.3 of NEI 00-01 - Timing of Fire-induced Failures

For plants transitioning to NFPA 805, Section 2.2.1 of RG 1.205 states that for cases where the NRC did not specifically approve certain aspects of the plant's current FPP (e.g., through an approved request under 10 CFR 50.12, "Specific Exemptions") licensees can submit elements of their plant's FPP, if they want explicit approval of these elements under 10 CFR 50.48(c).

In LAR Attachment T, "Prior-Approval Clarification Request" the licensee requests the staff to document that it had previously approved several assumptions related to the timing of fire damage. Specifically, LAR Attachment T states:

> "As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as a "prior approval" recognition that during the 10 minutes required to activate the SSF, fire growth will not have reached a point where fire damage will preclude operator actions from the Control Room nor will any spurious operations or loss of offsite power conditions occur within the first 10 minutes following the identification of a confirmed active fire" (emphasis added).

In summary, the licensee requested the NRC staff to document that the NRC previously approved the following limitations of fire damage for a 10 minute period of time needed to activate the SSF:

- no damage circuits and controls located in the control room
- no spurious operations
- no loss of offsite power

Section 3.5.1.3 of NEI 00-01 states that circuit failure types resulting in spurious operations should be assumed to exist until action has been taken to isolate the given circuit from the fire area. The licensee states it does not align with this guidance because its pre-transition license basis did not assume spurious actuations or hot shorts due to a fire for the first 10 minutes of the event (i.e., no spurious operations or hot shorts are assumed to occur for 10 minutes after confirmation of an active fire).

In supplementary information provided in a letter dated November 19, 2010 (Reference 52) the licensee agreed to eliminate the "*10 minute free of fire damage*" assumptions outlined above and to perform an evaluation using NFPA 805 risk-informed processes. Specifically, the licensee states that it will utilize a risk-informed approach to evaluate scenarios that previously relied upon the 10-minute prior approval. This will involve a thorough review of existing analyses to identify new variances. Changes to the FPP, as a result of these variances, will be resolved using the change evaluation process. Upon completion of this activity, all applicable FRE(s) will be updated and compliance will be demonstrated consistent with NFPA 805, Section 4.2.4.2. Completion of these activities is an implementation item (SE Section 2.9, Table 2.9-1, Item 46).

For a variance to be supported by the risk-informed process, its risk differential in terms of core damage frequency must be calculated. As part of its response, the licensee provided the estimated ΔCDF and ΔLERF for three VFDRs it has identified thus far. Specifically, the licensee states that it performed a sensitivity study to estimate the delta risk for four valves, which were excluded from the FRE process due to deterministic application of the 10-minute assumption. The licensee states that the cumulative delta risk from these potential VFDRs is within the available PSW risk offset margin for all fire areas where the SSF is credited (i.e., AB Fire Area).

The licensee states that elimination of the 10-minute assumption will prohibit ONS from deterministically complying with ONS UFSAR, Section 3.1.11 which requires that following the loss of the control room function, the reactor must be able to be shutdown and maintained in a safe and stable condition.

As discussed by the NRC staff in closure memorandum for FAQ 07-0032 (ADAMS Accession No. ML081400292), conformance to 10 CFR 50.48(c) meets or exceeds the requirements of 10 CFR 50.48(a) and GDC 3. Therefore, the NRC staff agrees that conformance to 10 CFR 48(c) also satisfies ONS specific UFSAR Criterion 11 for fire response.

As a result of its elimination of the "*10 minute free of fire damage assumption*," the licensing basis of the SSF following a fire will be dictated by the NFPA 805 risk-informed process. As such, the time allowance for performing certain SSF actions during a fire will be established by analyses required to support the risk-informed operation of the SSF.

Based on the information provided in the licensee's November 19, 2010 (Reference 52) supplement the NRC staff finds that the licensee's process for eliminating the "*10 minute free of fire damage assumption*" provides reasonable assurance that the safety objectives of NFPA 805 are satisfied.

3. <u>Section 3.5.1.5 of NEI 00-01 - Multiple Spurious Operations</u>

Section 3.5.1.5 of NEI 00-01 provides guidance for the analysis of multiple spurious actuations (MSOs). The LAR states that it does not align with this guidance because MSOs were not addressed in its pre-transition SSD analysis. The licensee further states that MSOs are resolved through transition to NFPA 805.

The PB approach taken by the licensee utilizes FREs in accordance with NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluation." To ensure that all potentially significant fire scenarios have been evaluated, this approach requires that potential MSO combination be identified and included. The licensee states that the fire area analysis methodology assumes multiple fire induced failures and multiple spurious actuations, based on the SSD cables and components present in the fire area of concern. All postulated SSD cable and component failures were identified and a resolution provided at the cable or component level for the credited train.

The NRC staff was concerned that the MSO analysis was limited to only SSD cables and components. In its October 14, 2010 response (Reference 54), the licensee states that the methodology assumed multiple fire induced failures and multiple spurious actuations, based on the cables and components present in the fire area of concern, and was not limited to SSD cables and components.

Based on the information provided in the October 14, 2010 (Reference 54), the NRC staff finds that the licensee's basis for alignment to Section 3.5.1.5 of NEI 00-01 acceptable. SE Section 3.2.3 provides the NRC staff evaluation of the licensee's MSO assessment process.

4. <u>Section 3.5.2.5 of NEI 00-01 – Common Enclosure Circuits</u>

Section 3.5.2.5 of NEI 00-01 provides guidance for addressing the Common Enclosure Associated Circuit concern described in Section 2.4.2.2.2 of NFPA 805. One element of this concern is that fire-induced electrical faults on cables that lack appropriately sized fuses or circuit breakers may cause secondary fires outside of the immediate fire area.

In the LAR, the licensee credits its original breaker coordination study to address Common Enclosure concerns. However, as discussed above, the original ONS coordination study does not satisfy applicable NFPA 805 or NEI 00-01 criteria. By letter dated September 13, 2010 (Reference 12) the licensee stated that a revised breaker coordination study had been completed. In Enclosure 3 of the LAR (Reference11) the licensee states that the second phase of the revised coordination study includes a review of the cable damage curves to determine if the electrical circuit design provides proper protection. The results of this review were entered into the ARTRAK database and analyzed in the Fire Area/Fire Zone impacts. All power supplies required by the NSCA, PRA, and NPO, as identified on the associated equipment list, were included in the breaker coordination study scope of "SSD related" power supplies. The licensee further states that the coordination study meets the requirements of NFPA 805, Section 2.4.2.2.2.(b), for circuits that share a common enclosure with circuits required to achieve nuclear safety performance criteria. In addition, the licensee states that a review of recent modifications confirms that adequate electrical circuit protection has been maintained as part of the design change process. In addition, the licensee states that the results of the coordination study will be documented in the NSCA, NPO Pinch Point Analysis, and the Fire PRA.

Incorporating all related non-coordinated information in the NSCA and NPO Pinch Point Analysis, and updating the Fire PRA model, to include the results of the breaker coordination study is an implementation item (SE Section 2.9, Table 2.9-1, Item 33).

Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach has adequately addressed common enclosure associated circuit concern at ONS and that the licensee will provide the required electrical protection under NFPA 805, Section 2.4.2.2.2.(b) upon completion of the identified implementation items.

5. <u>Section 3.1.1.9 of NEI 00-01 – 72-hour Coping Duration</u>

The nuclear safety goal, objectives, and performance criteria of NFPA 805 allow more flexibility than the previous deterministic FPPs based on Appendix R to 10 CFR Part 50, as well as, in part, NEI 00-01 Chapter 3, since NFPA 805 only requires the licensee to maintain the fuel in a safe and stable condition rather than achieve and maintain cold shutdown. In the LAR, the licensee states that the NFPA 805 licensing basis for ONS is to achieve and maintain safe and stable conditions in the plant, which is defined in NFPA 805, Section 1.6.56, as follows:

> For fuel in the reactor vessel, head on and tensioned, safe and stable conditions are defined as the ability to maintain Keff <0.99, with a reactor coolant temperature at or below the requirements for hot shutdown for a boiling water reactor and hot standby for a pressurized water reactor. For all other configurations, safe and stable conditions are defined as maintaining Keff <0.99 and fuel coolant temperature below boiling.

However, the licensee further states, in the LAR, that the nuclear safety goal of NFPA 805 was accomplished at ONS by developing and analyzing a comprehensive list of systems and equipment to identify those critical components required to achieve and maintain hot standby for 72 hours following a fire from at-power conditions. The licensee also states that long-term actions would be required to maintain hot standby beyond the proposed 72-hour "mission time."

NFPA 805 does not define a time period in which a safe and stable condition should be evaluated. Therefore, demonstrating the ability to maintain hot standby for only the first 72 hours following a fire does not, by itself, provide an adequate level of assurance that the nuclear safety goal of NFPA 805 is satisfied.

By letter dated November 19, 2010, (Reference 52), the licensee states that following stabilization at hot standby, assessment and repair activities, would commence to restore plant equipment needed to enable an RCS cool down in a safe and controlled manner. ONS does not anticipate a need to maintain a unit in hot standby for greater than 72 hours. Following stabilization at hot standby, assessment and repair activities would commence to restore plant equipment needed to enable PCS cool down in a safe and controlled manner. For the most limiting fire scenarios, it is anticipated that the end state of the cooldown would be an RCS temperature of approximately 250°F with a long-term strategy for reactivity, decay heat removal and inventory/pressure control. Long-term subcooled natural circulation decay heat removal is provided by supplying lake water to the steam generators and steaming to atmosphere. The extended coping period at these conditions is based on the significant volume of water available for decay heat removal and reduced need for primary makeup to only match nominal system losses. Since the scope of repair activities needed to maintain safe and stable conditions beyond 72 hours is dependent upon the magnitude and location of a potential fire, the licensee states that the mitigation strategy, damage assessment procedures, and repair equipment for both main control room (MCR) and SSF Shutdown scenarios will be established as part of a *"long term safe shutdown program"* which is to be included in the scope of the NFPA 805 program. The licensee states that this program will be completed as part of its implementation of NFPA 805 and will include mitigation strategies, damage assessments, required equipment, and procedural guidance. Any changes that need to be made to the *"long term safe shutdown program"* during implementation will be resolved using the change evaluation process.

The licensee states that based on the following factors, the qualitative risk associated with the recovery of long term SSD equipment is expected to be insignificant based on the following factors:

- The number of required recovery actions is limited.
- Procedures will be in place for each recovery action.
- The staff will be trained in the use of the recovery procedures.
- Required tools and replacement parts will be maintained on site.
- The 72 hour coping period provides a reasonable assurance that adequate time is provided to augment plant staffing and complete the recovery actions.

The predetermined strategy with supporting procedures and repair equipment to prepare for transition from the initial hot standby condition to long-term decay heat removal provides assurance that the fuel will be maintained in a safe and stable condition. The completion of required activities described above is an implementation item (SE Section 2.9, Table 2.9-1, Item 39).

Based on the information provided in the LAR, as supplemented by the licensee's November 19, 2010 (Reference 52) response, the NRC staff finds that the licensee approach has adequately demonstrated the capability to achieve and maintain the fuel in a safe and stable condition for an indefinite period following a fire.

A. Current Transformer Circuit Analysis

Attachment B of the LAR states that ONS does not align with the guidance contained in Section 3.5.2.1 of NEI 00-01 because it disagreed that an open circuit in the secondary winding of current transformers (CTs) could cause secondary fires. In addition to not meeting NFPA 805 or NEI 00-01 expectations, the staff noted that the licensee's assumption and alignment basis statements were not consistent with the ONS Design Basis Document (DBD) for fire protection, which states:

- CTs may induce secondary fires through the fire-induced opening of circuitry associated with the secondary side windings of the CT

- The impact of the fire-induced opening of CT secondary-side circuits will be considered. Resolution will be provided through proper CT qualification or the performance of a fire hazards analysis to determine if a secondary fire ignition will be a concern.

By letter dated September 13, 2010 (Reference 12) the licensee states:

- The internal NRC memorandum referenced in the LAR is not part of the ONS fire protection licensing basis. Accordingly, the following statement will be removed from the B-2 Table, Section 3.5.2.1:

  > The NRC disagreed with the conclusion formed by Brookhaven National Lab that this was a credible event. Based on EPRI [Electric Power Research Institute] data and documented in NRC internal correspondence, this was determined to be an "overly conservative" position and "lacked substantiation."

- The assumption associated with the secondary CT circuits is being removed to ensure that ONS has properly evaluated the effects of an open secondary CT as prescribed in NFPA 805, Section 2.4.2 and guided in NEI 00-01, Section 3.5.2.1.

- The analysis for secondary CT circuits will utilize the methodology in the ONS DBD. All CT cables are analyzed and any consequences will be included in the NSCA.

- The B-2 Table alignment basis statement will be revised to state that ONS aligns with the guidance provided in NEI 00-01.

- The NSCA shall be revised to ensure the analysis of secondary CT circuits is carried forward.

Completion activities needed to assure that fire-induced open secondary circuits of current transformers will not impact the ability to achieve and maintain the fuel in a safe and stable condition is identified as an Implementation Item (SE Section 2.9, Table 2.9-1, Item 40).

Based on the information contained in the LAR, as supplemented, the NRC staff finds the licensee's proposed method for addressing fire-induced open secondary circuits in current transformers acceptable.

B. Monitoring and Diagnostic Instrumentation

By letter dated April 14, 2010, (Reference 11), the licensee provided its response to the NRC staff's concerns regarding the adequacy of process monitoring and diagnostic instrumentation assured to remain available in the event of fire. For shutdown from the main control room, the licensee states that the following process monitoring and diagnostic instrumentation are available:

- RCS Temperature
- moisture separator (MS) Pressure
- Pressurizer Level
- SG Level'
- BWST Level
- Source Range Flux
- PSW Flow to A & B SG
- HPI Header Flow
- HPI Seal Injection Flow
- Letdown Storage Tank Level
- Letdown Storage Tank Temperature
- RB Pressure

For shutdown from the SSF, the following process monitoring and diagnostic instrumentation are identified:

- RCS Temperature
- Pressurizer Pressure
- Pressurizer level
- SSF auxiliary service water (ASW) Flow
- SSF RC Makeup Flow
- SSF RC Makeup Suction Pressure
- SSF RC Makeup Discharge Pressure

The specific process monitoring and diagnostic instrumentation assured to remain available for each fire area, has been identified by the licensee in calculation OSC-9695, "Oconee Nuclear Safety Capability Assessment for Units 1, 2, and 3," April 8, 2010. The licensee states that the

LAR B-2 Table will be revised to show that OSC-9695 is the reference for the NSCA that identifies the above process instrumentation. Revision of the B-2 Table is an implementation item (SE Section 2.9, Table 2.9-1, Item 38).

Both neutron instrumentation and SG pressure indication are available in the MCR. For SSF shutdown, however, the licensee states the use of boron sampling in lieu of neutron source range monitoring instrumentation and the lack of SG pressure instruments have been previously accepted by the staff and documented in an exemption. Attachment K of the LAR states that NRC acceptance of this configuration (i.e., the lack of neutron instrumentation and SG pressure indication at the SSF) is documented by the NRC staff in an SE dated August 31, 1983 (ADAMS Accession No. ML091310038) (Reference 40).

With regard to diagnostic instrumentation, the licensee states that the process monitoring instrumentation described in OSC-9695, provides the operator with adequate indication to quickly recognize and mitigate abnormal plant transients. In addition, the post-fire SSD procedure will contain a list of all credited instrumentation for each affected fire area.

With regard to the effects of fire to instrument sense lines, the licensee's April 14, 2010 letter (Reference 11) states that the potential for inaccurate instrument indications and/or spurious equipment actuations that could occur as a result of an instrument sensing line being exposed to a fire and increased temperatures has been considered in the analysis. Instrument sensing lines that could prevent the fulfillment of the SSD performance criteria have been identified and included in the fire area compliance assessment for review. Based on an evaluation of the materials used in the primary sensing line pipes and fittings, the licensee states that fire will not impact the sensing line fluid boundary. However, exposure of the copper tubing used on the secondary plant systems is evaluated as a loss of the instrumentation function (i.e., indication or interlock).

C. <u>Offsite Power</u>

The April 2010 LAR (Reference 11) Alignment Basis indicates that the availability of offsite power has been analyzed. By letter dated September 13, 2010 (Reference 12), the licensee states that the credited power supplies are the Keowee Hydro Station (KHS) and the SSF DG and neither the KHS nor the SSF DG requires offsite power. The licensee also states that the adverse consequences of offsite power being available are considered in the NSCA.

The licensee has created an action item to revise calculation OSC- 9291, NFPA 805 Transition B-2 Table, Section 3.1.1.7 to reword the alignment basis to clearly state that offsite power is not credited for the deterministic analysis and therefore not analyzed for its availability in the deterministic analysis. The licensee also states that the alignment statement will also be revised to ensure the proper relationship with the alignment basis. This is an implementation item (SE Section 2.9, Table 2.9-1, Item 47).

D. <u>Alignment Basis Level of Detail</u>

The licensee submitted its initial LAR for the transition to NFPA 805 on May 30, 2008 (Reference 2). As a result of changes needed to comply with RG 1.205, Revision 1, this LAR was superseded by an LAR submitted on April 14, 2010 (Reference 11). A review of the 2010 LAR identified several sections of Table B-2 that had been modified to the extent that the level of detail provided was below the level needed to readily confirm alignment with the NEI guidance. In a letter dated September 13, 2010 (Reference 12), the licensee identifies a total of

57 sections of the LAR B-2 Table that had been modified. Of these, the licensee determined that fifteen sections had insufficient detail to clearly demonstrate conformance to the applicable sections of NEI 00-01. The licensee states that the LAR B-2 Table will be revised to include additional clarification of alignment with the NEI guidance. Revision of the LAR B-2 Table will be tracked in the corrective action program and include these additional clarifications as an implementation item (SE Section 2.9, Table 2.9-1, Item 38).

E. Armor Jacketed Cable Grounding

In a letter dated November 30, 2009 (Reference 10), the licensee states that the interlocked armor on the cables at ONS are terminated and grounded as required by ONS Engineering Design Criteria DC-4.11 which states that the armor of interlocked armor cable be electrically continuous and grounded to equipment enclosure at each end of the cable. The licensee also states that a sample of plant design changes have been reviewed to ensure the original design criteria is being referenced in the change modifications with regards to grounding the cable armor. Based on its review of drawings, cable specifications, and modifications, the licensee states that it has a high degree of confidence that the as-installed configuration of the armor cable grounding scheme is consistent with the original plant design.

Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee has adequately addressed the issue of grounding of armored cable to preclude inter-cable shorts.

F. Section 3.2.1 Conclusion

Based on the information provided in the licensee's submittal, the NRC staff finds the method the licensee used to perform the NSCA with respect to selection of systems and equipment, selection of cables, and identifying the location of equipment and cables acceptable, because the method used either met the NRC-endorsed guidance directly or met the intent of the endorsed guidance with adequate justification as documented in Table 3.2-1 (see SE Attachment B).

3.2.2. Applicability of Feed and Bleed

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

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

The NRC staff reviewed LAR Table 5-3, "10 CFR 50.48(c) - Applicability/Compliance References," and Attachment C, "NEI 04-02 Table B-3, Fire Area Transition," to evaluate whether the licensee meets the feed and bleed requirements. The licensee stated in LAR Table 5-3 that feed and bleed is not utilized as the sole fire protected SSD path at ONS for any scenario. The NRC staff verified this by reviewing the designated SSD path listed in LAR Attachment C for each fire area. Although loss of pressurizer heaters was considered possible, this review confirmed that all fire area analyses include the SSD equipment necessary to provide decay heat removal without relying on feed and bleed and the PORV is not the only means of pressure control for potential solid water operation. In addition, all fire areas either met the deterministic requirements of NFPA 805, Section 4.2.3; or the PB evaluation performed in accordance with NFPA 805, Section 4.2.4, demonstrated that the integrated assessment of risk, DID, and SMs for the fire area was acceptable.

The NRC staff determined that based on the information provided in LAR Table 5-3, as well as the fire area analyses documented in LAR Attachment C, the licensee meets the requirements of 10 CFR 50.48(c)(2)(iii) because feed and bleed is not utilized as the sole fire-protected SSD path at ONS.

3.2.3. Assessment of Multiple Spurious Operations (MSOs)

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

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

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

Accordingly, the NRC staff reviewed LAR Section 4.2.1.4, "Evaluation of Multiple Spurious Operations," and Attachment F, "Fire-Induced Multiple Spurious Operations Resolution," to determine whether the licensee has adequately addressed MSO concerns at ONS. The licensee's chosen approach used an expert panel to identify potential MSO combinations, which needed to be considered in the NSCA, as well as to assess the plant-specific vulnerabilities associated with these MSO combinations.

The expert panel was a diversified group of subject matter experts in the following areas:

- Operations
- Fire Protection Engineering
- Post-Fire SSD Analysis
- Probabilistic Risk Assessment (PRA)
- Fire Protection/Post-Fire SSD Consultant

The expert panel utilized guidance provided in NEI 00-01, Revision 1, Appendix F, Section F.4.2, and "Expert Panel Review" (ADAMS Accession No. ML050310295), (Reference 56). The expert panel, which was conducted in February 2006, considered: the post-fire SSD analysis for ONS, the self-assessment process identified in NEI 04-06 (Reference 44), insights provided by the internal events PRA for ONS, industry and plant-specific operating experience, and a review of the ONS Simplified Flow Diagram (OSFD) of the reactor coolant system (RCS). The expert panel generated a list of MSO scenarios in an effort to reflect the intent of the guidance provided in NRC Regulatory Information Summary (RIS) 2004-03, Revision 1 (Reference 45).

Subsequent to the expert panel meeting, the generic list of fire-induced MSO scenarios provided by the PWR Owners Group (PWROG) as part of the update process for NEI 00-01, Revision 2, were compared to the MSO scenarios identified by the expert panel and, if not already considered, were added to the ONS-specific list of MSO scenarios. The results of both the expert panel review and the review of the PWROG MSO scenarios were incorporated into the NSCA as well as the Fire PRA.

The MSO combinations included in the NSCA were evaluated with respect to compliance to the deterministic requirements of NFPA 805 Section 4.2.3, "Deterministic Approach." For those situations in which the MSO combination did not meet the deterministic requirements of NFPA 805, the components and associated cables were added to the scope of the FREs performed for the associated fire area.

The licensee's alignment basis for Section 3.5.1.5(B) of NEI 00-01 stated that circuit failures were considered to occur on each conductor of each SSD cable. However, the alignment basis did not specifically state if the failures were considered to occur individually (e.g., conductor A shorts to B; conductor A shorts to C) or concurrently (e.g., conductor A shorts to B and C). The NRC staff requested the licensee to provide additional clarification with regard to its evaluation of intra-cable circuit failures within a single multi-conductor cable. In its September 13, 2010 (Reference 12), response to a staff request for additional information, the licensee states that any and all potential spurious actuations that may result from intra-cable shorting were considered. Such failures were considered to occur concurrently, regardless of number, in accordance with the guidance provided in NEI 00-01, Section 3.5.1.5(B).

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

3.2.4. Transition of Operator Manual Actions to Recovery Actions

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

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

NFPA 805, Section 4.2.3.1, states that:

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

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

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

The NRC staff reviewed LAR Section 4.2.1.3, "Transition of Operator Manual Actions to Recovery Actions," and 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 based its approach for transitioning operator manual actions (OMAs) into the 10 CFR 50.48(c) RI/PB FPP as recovery actions on NEI 04-02, Revision 2, Section 4.6, "Regulatory Submittal and Transition Documentation," as endorsed with exceptions by RG 1.205, Revision 1 (Reference 14).

The population of OMAs addressed during the NFPA 805 transition process at ONS included the existing OMAs in the deterministic FPP, as well as those being added during the NFPA 805 transition to address VFDRs of NFPA 805 Section 4.2.3. OMAs meeting the definition of a recovery action are required to comply with the NFPA 805 requirements outlined above. The licensee states that all pre-transition OMAs at ONS are actions that take place at the primary control stations and are therefore not recovery actions per NFPA 805, Section 1.6.52.

However, during the resolution of VFDRs, the licensee identified 12 recovery actions to satisfy the DID requirements of NFPA 805, Section 1.2. The licensee stated that all of these recovery actions were reviewed to verify that they could not have an adverse impact that would increase risk and none were found to have an adverse impact on the Fire PRA results.

The licensee stated that it subjected all recovery actions to a feasibility review. In accordance with the NRC-endorsed guidance in NEI 04-02, the feasibility criteria used were based on the nine attributes provided in Section B.5.2, "Methodology Success Path Resolution Considerations," of Appendix B, "Nuclear Safety Analysis," to NFPA 805. LAR Attachment G includes Table G-1, "Feasibility Criteria – Recovery Actions and DID Actions (Based on NFPA 805, Appendix B.5.2(e) and NEI 04-02, Revision 2)," that lists the nine attributes used to assess recovery action feasibility. A feasibility evaluation was performed for each identified recovery action on a fire-area basis. Based on the results of these evaluations, the licensee determined that the recovery actions met all except the following feasibility criteria:

- Communication. Hand-held radios are relied upon to ensure communication between the field operator and the main and SSF control rooms. An evaluation will be performed to ensure the radios operate in the locations of the recovery actions, either with or without repeaters.

- Procedures. Procedures do not exist for the non-SSF fire areas. SSD procedures will be developed for Fire Areas RB1, RB2, and RB3.

- Training. No training is provided for fire areas utilizing the MCR for unit shutdown. Training will be provided to the operators on the new SSD procedures developed for Fire Areas RB1, RB2, and RB3.

- Drills. Emergency drills will be conducted on the new SSD procedures developed for Fire Areas RB1, RB2, and RB3.

Completion of these activities is considered an implementation item (SE Section 2.9, Table 2.9-1, Item 14). The NRC staff finds that the licensee's application of feasibility criteria for recovery actions, is consistent with the endorsed guidance found in NEI 04-02, Revision 2, and is therefore acceptable.

The licensee stated that no specific recovery actions were modeled in the Fire PRA. Instead, the fire-induced failure of the cables that prompted the associated recovery actions was included as VFDRs. Therefore, the risk of the VFDRs includes or bounds the risk of the identified recovery actions even if the recovery actions were modeled and assumed to be perfectly reliable. The NRC staff finds this bounding approach to demonstrating the reliability of the identified recovery actions acceptable.

While performing the review of the licensee's treatment of the transition of OMAs to recovery actions, the NRC staff identified several issues that required the licensee to provide additional information in order to adequately demonstrate compliance with specific portions of the applicable NFPA 805 requirements. In response to RAIs (References 42 and 43), the licensee stated that it defined primary control station (PCS) actions in the LAR as follows:

- Actions inside the MCRs,
- Actions inside the SSF control room,
- Actions inside the SSF facility to transfer control from the MCR to the SSF,
- Actions inside the SSF facility to operate manual valves, and

"Deployment and operation of the SSF submersible pump" was also defined as a PCS action in the LAR. This action includes actions to retrieve, assemble, and deploy the portable submersible pump, including making necessary hose(s) and electrical power connections that are not predominantly conducted in the SSF or deployed during the initial transfer of control from the control room. By letter dated November 19, 2010 (Reference 52), the licensee stated that deployment and operation of the SSF submersible pump is reclassified as a Fire Area AB VFDR and is being transitioned as a recovery action. The licensee further stated that this recovery action would only be required to support long-term SSF operation in very specific scenarios. These scenarios are modeled in the flooding PRA, but are not modeled in the Fire PRA. The licensee provided characteristics of the actions (e.g., proceduralized and periodically evaluated) based on which action meets the feasibility criteria. The NRC staff finds that the licensee's application of feasibility criteria for this recovery action is consistent with the endorsed guidance found in NEI 04-02, Revision 2, and is therefore acceptable.

As discussed previously, the NRC staff requested the licensee provide additional information regarding the assumption that no spurious operations or loss-of-offsite power would occur within the first 10 minutes following confirmation of an active fire event during which time the operators are transferring control to the SSF. In its letters dated August 3, 2009 (Reference 8), April 14, 2010 (Reference 11), and November 19, 2010 (Reference 52), the licensee stated that the 10 minute delay was not credited in the Fire PRA, but that the Fire PRA assumed, for each associated fire scenario, applicable spurious operations and failure of all equipment determined from the fire analysis. In addition, component failures or spurious actuations caused by any fire-induced damage were not subsequently credited as recovered by an OMA in the Fire PRA. The NRC staff finds this approach acceptable because fire-induced functional failures were not recovered in the Fire PRA, which resulted in a conservative assessment of the fire risk.

Based on the above considerations, the NRC staff finds that the licensee has followed the endorsed guidance of NEI 04-02 and RG 1.205 regarding the transition of OMAs to recovery actions in accordance with NFPA 805, thereby meeting the regulatory requirements of 10 CFR 50.48(c). The NRC staff concludes that the feasibility criteria applied to recovery actions are acceptable based on conformance with the endorsed guidance contained in NEI 04-02.

3.2.5. 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 NSCA at ONS. The NRC staff found 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 nuclear safety performance criteria of NFPA 805, Section 1.5.

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

The NRC staff reviewed the licensee's process to identify and analyze MSOs. Based on the information provided in the LAR, as supplemented, the process used to identify and analyze MSOs at ONS is considered comprehensive and thorough. Through the use of an expert panel, potential MSO combinations were identified and included as necessary into the NSCAs as well as the applicable FREs. The NRC staff also 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 found that, based on the information provided in the LAR, as supplemented, the process used by the licensee to review, categorize, and address recovery actions during the transition from the existing deterministic fire protection licensing basis to an RI/PB FPP is consistent with the NRC-endorsed guidance contained in NEI 04-02 and RG 1.205 regarding the transition of OMAs to recovery actions and other actions required to be taken at a PCS. Therefore, this process meets the regulatory requirements of 10 CFR 50.48(c) and the guidelines of NFPA 805.

3.3. Fire Modeling

NFPA 805 allows the use of fire modeling as a PB alternative to the deterministic approach outlined in the standard. 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."

NFPA 805, Section 2.4.1, "Fire Modeling Calculations," specifically addresses the application requirements for using PB fire models as follows:

- NFPA 805, Section 2.4.1.2.1, "Acceptable Models," states the following:

> Only fire models that are acceptable to the authority having jurisdiction shall be used in fire modeling calculations.

- NFPA 805, Section 2.4.1.2.2, "Limitations of Use," states the following:

> Fire models shall only be applied within the limitations of that fire model.

- NFPA 805, Section 2.4.1.2.3, "Validation of Models," states the following:

> The fire models shall be verified and validated.

NFPA 805, Section 4.2.4.1, "Use of Fire Modeling," identifies the specific approach for use of fire modeling as a PB method, including the following required aspects: identify targets, establish damage thresholds, determine limiting condition(s), establish fire scenarios, protection of required nuclear safety success path(s), and operations guidance.

In addition, RG 1.205, Revision 1 (Reference 14), Regulatory Position C.4.2, and NEI 04-02, Revision 2 (Reference 21), Section 5.1.2, "Fire Modeling Considerations," provide guidance by identifying fire models that are considered acceptable for use by the NRC for plants transitioning to a RI/PB FPP in accordance with NFPA 805 and 10 CFR 50.48(c).

The NRC staff reviewed LAR Section 4.5.2, "Fire Modeling," which describes how the licensee used fire modeling as a part of the transition to NFPA 805 at ONS, and LAR Section 4.7.3, "Compliance with Quality Requirements in Section 2.7.3 of NFPA 805," which describes how the licensee complied with the NFPA 805 quality requirements for fire protection systems and features at ONS.

In LAR Section 4.5.2, the licensee stated that fire modeling analyses were used only to support development of the Fire PRA for use in performing FREs (i.e., in accordance with NFPA 805, Section 4.2.4.2, "Use of Fire Risk Evaluations"). The fire modeling PB method of NFPA 805, Section 4.2.4.1 was not used. The NRC staff reviewed the technical adequacy of the ONS Fire PRA, including the supporting fire modeling analyses, as documented in SE Section 3.4.3.

Because the fire modeling PB method of NFPA 805, Section 4.2.4.1, was not used, the NRC staff has not reviewed any such methods for acceptability in that context. Since the NRC staff has not reviewed any such fire modeling methods, the staff does not find any plant-specific fire modeling methods acceptable for use to support compliance with NFPA 805, Section 4.2.4.1, as a part of this licensing action supporting transition to RI/PB FPP.

3.4. Fire Risk Assessments

This section addresses the licensee's FRE PB method, which is based on NFPA 805, Section 4.2.4.2.

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

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

DID and SMs are discussed in SE Section 3.4.1. The acceptability of risk is discussed in SE Sections 3.4.2 through 3.4.6.

For those fire areas where the licensee used a PB approach to meet the nuclear safety performance criteria, the licensee used FREs in accordance with NFPA 805 Section 4.2.4.2 to demonstrate the acceptability of the plant configuration. Some VFDRs were resolved with plant modifications. Each remaining VFDR was evaluated for risk impact and maintenance of the philosophy of DID and maintenance of SMs associated with bringing the VFDR into the licensing basis instead of removing the VFDR by bringing the plant into compliance with the deterministic requirements.

3.4.1. Maintaining Defense-in-Depth and Safety Margins

When implementing the PB approach, the licensee followed the guidance contained in Section 5.3.5, "Acceptance Criteria," of NEI 04-02, which includes a detailed consideration of DID and SMs as part of the risk evaluation process. FREs were performed for each fire area, which includes a risk evaluation of each VFDR and a composite risk evaluation of the entire fire area. Each fire area FRE includes an assessment of DID systems and features and an assessment of how adequate the SM is maintained. The results of this assessment are summarized for each VFDR by fire area, in the LAR Attachment C Table B-3.

Defense-in-Depth (DID)

NFPA 805, Section 1.2, states the following:

> DID shall be achieved when an adequate balance of each of the following elements is provided:
>
> - Preventing fires from starting.
> - Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage.
> - Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

The NRC staff reviewed LAR Sections 4.2.4, "Fire Area-by-Fire Area Transition," Section 4.5.3, "NFPA 805 Fire Risk Evaluation Process," Table 4-3 "Fire Risk Evaluation Guidance Summary Table," and Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features," as well as the associated supplemental information, in order to determine whether the principles of DID were maintained in regard to the planned transition to NFPA 805.

Each fire area FRE includes an assessment of DID systems and features credited to maintain a balance amongst the DID attributes and identification of DID enhancements needed to disposition VFDRs and restore the balance among the DID attributes. LAR Attachment C Table B-3 (1) documents the existing balance of DID, (2) indicates whether or not the element needs to be strengthened by modifications (such as the installation of fire detection systems or other fire protection modification), and (3) documents the presence of automatic fire detection and suppression systems.

The licensee's process for evaluating fire suppression and detection systems also included a review of those systems credited to meet the NFPA 805 deterministic requirements. This review included the identification of automatic suppression and detection systems credited in NRC staff approved exemptions from the existing fire protection licensing basis and being carried forward into the RI/PB FPP as well as those credited by the licensee in EEEEs.

The NRC staff review finds 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 changes as proposed in its LAR adequately maintains DID against fires as required by NFPA 805.

Safety Margin (SM)

Although the appendices to NFPA 805 are not incorporated into the regulation, they may provide insight into what the authors of that Standard intended. Section A.2.4.4.3 of Appendix A to NFPA 805 provides the following background related to the meaning of the term "SMs:"

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

LAR Section 4.2.4, Section 4.5.3, and Table 4-3 stated that SMs were considered as part of the FRE process. Specifically, LAR Section 4.5.3.2 stated that the licensee evaluated each VFDR against the SM criteria of NEI 04-02.

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 SMs:

- Codes and Standards or their alternatives accepted for use by the NRC are met, and
- Safety analyses acceptance criteria in the licensing basis (e.g., FSAR and supporting analyses) are met, or provides sufficient margin to account for analysis and data uncertainty.

As described in SE Section 3.1, the licensee meets various codes and standards associated with fire protection. The licensee listed the SM attributes considered during its FRE for each fire area in LAR Attachment C, Table B-3. The SM attributes listed include the following:

- Fire modeling performed in support of the transition has been performed within the Fire PRA utilizing codes and standards developed by industry and endorsed by NRC.
- Plant system performance parameters were not modified as a result of the FREs.

- The bases for the application of the Fire PRA codes and standards were not altered in support of the FREs.

- In accordance with the requirements of 10 CFR 50.48(c)(2)(iii), the Fire PRA results, including cutsets for the scenarios of concern, were reviewed by the licensee and the licensee verified that the results presented do not rely solely on feed-and-bleed as the fire-protected SSD path for maintaining reactor coolant inventory, pressure control, and decay heat removal capability for each fire area.

In addition to the attributes listed above, the installation of the PSW and its associated risk reduction provides additional margin. The criteria described in NEI 04-02, Section 5.3.5.3 and the LAR are consistent with the criteria as described in RG 1.174 and therefore acceptable. The NRC staff finds that the licensee's approach has adequately addressed the issue of SM in the implementation process because the licensee used appropriate codes and standards (or NRC approved alternatives) and met the safety analyses acceptance criteria in the licensing basis (or, through the application of its Fire PRA in its FREs provided sufficient margin to account for analysis and data uncertainty).

Defense-in-Depth and Safety Margin Conclusions

The licensee's FRE process included a detailed review of fire protection DID and SMs. The individual FREs and LAR Table 4-4 and Attachment C Table B-3 document the results of the DID and SM review. The NRC staff finds the licensee's evaluation in regard to DID and SMs to be acceptable because the licensee's process and results followed the endorsed guidance in NEI 04-02, Revision 2 and are consistent with the NRC staff guidance in RG 1.205, Revision 1 and RG 1.174, Revision 1.

3.4.2. Fire Risk Evaluation

In accordance with the guidance in RG 1.205, Section C.2.2.4, "Risk Evaluations," risk increases or decreases for each fire area using FREs and for the overall plant should be provided. In LAR Attachment C, Table B-3, the licensee provided the results of the FRE for each individual VFDR that was not resolved with a modification. The risk increases and decreases associated with the VFDRs for each fire area and the total fire risk for each unit are provided in LAR Attachment W, Tables W-2, W-3, and W-4.

The tables in LAR Attachment W provide the estimated risk increase in each fire area associated with not modifying the facility to remove the VFDR as permitted by the RI/PB implementation of 10 CFR 50.48(c). Consistent with RG 1.174 (combined change request) and as stated in RG 1.205, it is acceptable for transition to credit selected non-fire related modifications to reduce the risk associated with retaining VFDRs. The licensee credited the risk reduction from the proposed PSW modification and provided the amount of risk reduction from PSW for each fire area. The licensee stated that the installation of the PSW system is expected to decrease the risk associated with hazards other than fire because the functions provided can be used to help mitigate other initiating events just as they are used to mitigate fire-initiated events. However, the licensee also indicated that they did not credit this additional risk reduction capability of the PSW for other hazards, but rather only estimated and provided the risk impact of the PSW installation on fire risk. The NRC staff has reviewed this process for FREs and the licensee's documented results and found it to be acceptable because it estimates the change in risk required by NFPA 805, as incorporated by reference in 10 CFR 50.48(c).

3.4.3. Quality of the Probabilistic Risk Assessment

NFPA 805 Section 2.4.3.2 states that the PSA approach, methods, and data shall be acceptable to the AHJ. In reviewing a RI LAR, the NRC staff evaluates the acceptability of the plant-specific PRA analyses and their proposed application using guidance from RG 1.174. RG 1.174 addresses PRA approach methods, and data and also provides additional guidance clarifying how acceptability should be determined. In RG 1.174, the objective of the PRA quality review is to determine whether the plant-specific PRA used in evaluating the proposed LAR is of sufficient scope, level of detail, and technical adequacy for the application. The scope, level of detail, and technical adequacy of the PRA are to be commensurate with the application for which it is intended and the role the PRA results play in the integrated decision process. The more emphasis that is put on the risk insights and on PRA results in the decisionmaking process, the more requirements that have to be placed on the PRA, in terms of both scope and how well the risk and the change in risk is assessed. Conversely, emphasis on the PRA scope, level of detail, and technical adequacy can be reduced if a proposed change results in a risk decrease or is very small.

In its LAR, the licensee estimates that the total change in risk associated with its proposed transition to 10 CFR 50.48(c) will be a substantial decrease in CDF and LERF. A RI application that can clearly be shown to result in a decrease in risk is considered to have satisfied the relevant risk-related principle of RI regulation (i.e., Principle 4 of RG 1.174). Therefore, the NRC staff's review of the quality of the ONS PRA described below focused on whether the PRA is adequate to support the conclusion that granting the proposed amendment is expected to result in an overall decrease in risk.

The licensee performed a Fire PRA to support this application. The scope and level of detail of the Fire PRA is consistent with the requested licensing action, which changes the FPP and includes FREs. The licensee has no seismic PRA, but the NRC staff has concluded that seismic-fire interaction is adequately addressed with the licensee's qualitative analysis. Other external events, such as external floods and high winds, are expected to be insignificant causes and contributors to fire risk based on the relatively low frequency of these other external events occurring with a coincidental or consequential fire. Consistent with RG 1.174, since the licensee's application is a risk decrease, there is no requirement to calculate the total CDF or LERF contribution from all hazards when considering risk acceptability. Therefore, the NRC staff concludes that the scope and level of detail of the Fire PRA analysis is acceptable for this application.

As described in RG 1.174 and RG 1.200, one approach the NRC staff uses to assess the technical adequacy of the licensee's base PRA is to consider the industry peer review process and the licensee's resolution of the findings from this process for the specific application. In accordance with RG 1.200, the performance of an industry peer review of a licensee's base PRA that meets NRC-endorsed PRA standards obviates the need for the NRC staff to perform a detailed review of the licensee's base PRA. The PRA standards identify major elements of a PRA and provide numerous supporting requirements (SRs) for each element. SRs identify individual evaluation steps and describe the technical attributes of the analysis or documentation required to properly meet each SR. The results of a review against the PRA standard are referred to as findings, observations, or facts and observations (F&Os). The NRC staff's assessment of the quality of a licensee's PRA uses these F&Os as the starting point for its review of the technical adequacy of the licensee's base PRA model being used for an application.

In this application, the industry peer review used the ASME RA-Sb-2005 PRA standard, "Addenda to ASME RA-Sb-2002 Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications" (Reference 47), as the basis for the review of the licensee's internal events PRA. The licensee's application is an NFPA 805 pilot application. Consistent with Regulatory Position C.4.3 of the initial issuance of RG 1.205, the NRC staff performed a pre-LAR review of the licensee's Fire PRA model in lieu of an industry peer review of the model as part of the pilot process. The NRC staff's review of the Fire PRA model is discussed further under "Fire PRA Model," below.

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 incorporated into the Fire PRA model for potential risk impact as part of the change evaluation process. Further, as described in SE Section 3.8.3, the licensee has a program for ensuring the developers and users of these models are appropriately trained and qualified. The licensee did not identify any of the following 1) known outstanding or imminent plant changes that would require a change to the Fire PRA model, or 2) any planned plant changes during the 10 CFR 50.48(c) implementation period that would significantly impact the Fire PRA model, beyond those identified and scheduled to be implemented as part of the transition to the RI/PB FPP, as set forth in Section 2.8 of this SE. Therefore, the NRC staff finds that the PRA program for developing, maintaining, and using the Fire PRA provides confidence that the licensee has satisfied the guidance in RG 1.174, RG 1.200, and RG 1.205 that the Fire PRA model will appropriately represent the as-built, as-operated and maintained plant for this specific application, within the limitations established in the license condition and once the committed modifications, SE Section 2.8, and the implementation items, SE Section 2.9 are completed.

Internal Events PRA Model

Revision 2 of the licensee's PRA was completed in December 1996 and peer reviewed using NEI 00-02, "Industry PRA Peer Review Process," by the B&W Owners Group in May 2001. By letter dated August 3, 2009 (Reference 8), the licensee summarized the changes made in November 2006 to update Revision 2 to Revision 3. An independent contractor for the licensee reviewed Revision 3 of the PRA using the ASME RA-Sb-2005 PRA standard (Reference 47) in June 2006. The contractor identified and commented on SRs in the ASME PRA Standard that were either "not met" or that did not meet the Capability Category (CC) II; the CC that RG 1.200 deemed as adequate for most applications. Using the results of the contractor review, the licensee made changes to the PRA, updating Revision 3 to Revision 3a in June 2008. In October 2008, the licensee reported that it performed a self-assessment on Revision 3a against the ASME PRA Standard, as modified by Revision 1 of RG 1.200.

In its LAR, the licensee stated that its PRA fully meets 242 of the 306 ASME PRA SRs. The licensee determined that 24 of the remaining 64 SRs were not applicable or did not need to meet a CC II to support this application. The LAR and responses to an NRC staff RAI (Reference 8) briefly described the 40 F&Os on the remaining SRs and assessed the impact of resolving these remaining observations on the reported risk of the transition to 10 CFR 50.48(c). The 40 internal events PRA F&Os are described in Attachment C.1, Table 3.4-1, along with the licensee's resolution for this application and the NRC staff's conclusions regarding the acceptability of the licensee's resolution. These F&Os cover numerous aspects of the internal events PRA (e.g., human reliability modeling, internal flooding modeling, and large early release frequency calculations). A number of the F&O resolutions have not been implemented and will involve PRA modifications. These modifications of the PRA are not expected to change this application from a risk decrease to a risk increase due to the significant risk reduction attributed to the PSW modification, and the licensee has committed to confirm a risk reduction after completion of all implementation items in Section 2.9 Table 2.9-1.

The NRC staff recognizes that the Fire PRA is developed from the internal events PRA model, and as such, the issues identified with the internal events PRA can impact the Fire PRA results. However, the NRC staff concludes that there would have to be major errors or inaccuracies in the ONS internal events PRA in order to change the substantial estimated fire risk decrease from the PSW modification into a risk increase. Previous reviews of the licensee's internal events PRA did not identify any such major errors or inaccuracies beyond the human reliability analysis weaknesses for which sensitivity studies have been completed demonstrating a minimal expected impact on risk evaluations supporting the LAR. The NRC staff evaluated the peer review results and the licensee's responses as summarized in Attachment C.1 of this SE, and concludes that changes to the PRA after resolving all the F&Os are unlikely to result in an increase in risk due to the significant risk reduction attributable to the PSW modification. Completion of all implementation items in Section 2.9 (including updating and revising the HRA) will further confirm that implementation of the RI/PB FPP will result in a risk decrease. Therefore, the NRC staff finds that the internal events PRA has sufficient technical adequacy that the results can be relied upon to support the determination that the transition to NFPA805 will result in a decrease in risk.

Fire PRA Model

The licensee developed its Fire PRA model using the guidance of NUREG/CR-6850/EPRI 1011989 (Reference 37). The model addresses both Level 1 (core damage frequency) and partial Level 2 (i.e., large early release frequency only) PRA during at-power operations. The licensee modified the internal events PRA to capture the effects of fire, both as the initiator of an event and to characterize the subsequent potential failure modes for affected circuits or individual plant SSCs (targets), including fire-affected human actions and new human actions necessary as the result of a fire. The Fire PRA was initially developed for Unit 3. A second model for Unit 2 was developed from the Unit 3 model. The licensee reported that a comparative analysis of the failures and ignition frequencies for comparable fire compartments between Units 1 and 2 was performed. The comparative analysis indicated that a separate Unit 1 fault tree and Fire PRA quantification file were not necessary because Units 1 and 2 are sufficiently similar. For the limited number of cases where the Unit 2 results were not considered to be bounding for Unit 1, the licensee adjusted the Unit 2 model to yield results applicable to Unit 1. The adaption of a single PRA model to each unit at a multi-unit site with reasonable symmetrical designs is a common method to support risk-informed applications. The NRC staff finds that the licensee's recognition and reporting on its process resulting in one PRA for Unit 3 and a common PRA for the symmetrical Units 1 and 2 is sufficient to conclude that the licensee has evaluated the impact of unit-specific differences and that the results are sufficiently unit-specific to support the LAR.

As stated above, the NRC staff performed a review of the licensee's Fire PRA model to determine the technical adequacy of the model because an industry peer review of the ONS Fire PRA had not been performed. The NRC staff conducted the review of the ONS Fire PRA model in March 2008 (Reference 33).

The NRC staff's review compared the licensee's Unit 3 Fire PRA characteristics against the SRs of the draft PRA standard ASME/ANS RA-Sa-2009, Part 4, "Fire PRA Technical Elements and Requirements" (Reference 34). The review also used the industry guidance set forth in draft NEI 07-12, "Fire Probabilistic Risk Assessment Peer Review Process Guidelines" (Reference

35). The NRC review identified that the ONS Unit 3 Fire PRA, representative of all three units, was incomplete, although all tasks but one had been started and many of the tasks had been completed. Therefore, the NRC staff's audit report concluded that a focused-scope peer review of those portions of the Fire PRA that changed substantially in the time between the NRC staff's review and the submittal of the plant's 10 CFR 50.48(c) LAR may be necessary. The licensee did not have a focused-scope peer review performed. Instead the licensee provided descriptions of its resolutions to the F&Os with respect to the application to transition to NFPA 805 to the NRC staff for review. These F&Os resolutions are provided in Table V-1 of Attachment V of the LAR. The Fire PRA F&Os are described in Attachment C. Table 3.4-2, along with the licensee's resolution for this application and the NRC staff's finding regarding the acceptability of the licensee's resolution.

Similar to the internal events PRA review, the NRC staff concludes that there would have to be major errors or inaccuracies in the ONS Fire PRA in order to change the substantial estimated fire risk decrease from the PSW modification into a risk increase. The NRC staff's audit of the Fire PRA and review of the current resolution of all F&Os from the audit (summarized in Attachment C.of this SE) did not identify any such major errors or inaccuracies. Completion of all implementation items in Section 2.9 (including completing a peer review of the fire PRA, resolving the findings from the review, and re-evaluating the change in risk from transition) will further confirm that implementation of the RI/PB FPP will result in a risk decrease. Therefore, the NRC staff finds that the Fire PRA has sufficient technical adequacy that the results can be relied upon to support the determination that the transition to NFPA-805 will result in a decrease in risk.

Fire Modeling in Support of Development of Fire PRA

Typically, the technical adequacy of the fire modeling that supports development of the base Fire PRA for a RI license application is determined by the PRA standards and associated peer review activities, with the NRC staff's review focused primarily on the licensee's resolution of peer review findings and the actual use of (i.e., changes made to) the PRA to address the risk impacts of the proposed LAR. However, since this LAR is a pilot application of the new 10 CFR 50.48(c) requirements, the NRC staff performed additional detailed reviews of the specific fire modeling used to support specific aspects of the Fire PRA in order to gain further assurance that these methods and approaches used for the application to transition to 10 CFR 50.48(c) are technically adequate. The following paragraphs discuss the NRC staff's additional review of these aspects of the licensee's fire modeling.

In LAR Section 4.5.2, "Fire Modeling," the application of fire modeling was intended to develop the ZOI around ignition sources in order to determine the thresholds at which a target would exceed the critical temperature or radiant heat flux. This approach provides a basis for the fire modeling treatment in the Fire PRA. By letter dated August 3, 2009 (Reference 8), the licensee stated that only the generic fire modeling methodology discussed below was used for the Fire PRA and no additional fire models (e.g., Consolidate Model of Fire Growth and Smoke Transport (CFAST)) or detailed fire modeling was performed.

The licensee's ZOI approach applied a generic fire modeling methodology to distinguish between fire scenarios that required further evaluation and those that did not require further evaluation. In general, this methodology developed conservative ZOIs for each type of ignition source by assuming that the maximum heat release rates (HRR) develops at time zero and extends for up to 60 minutes. The licensee assumed their armored cables were of limited combustibility and did not include an HRR contribution from ignition and combustion of adjacent cables while developing the ZOI. The licensee stated that since the poly-vinyl chloride (PVC) coating on the armored cable will not sustain propagation of fire along the armored cable for a significant distance, any horizontal propagation along cables is adequately captured within the target set of each scenario. In addition, in many scenarios the ZOI extended vertically to the ceiling.

The licensee also developed screening approaches to the potential for the generation of an HGL in the compartment or fire area being analyzed. These screening HGL approaches were used in the Fire PRA to further screen scenarios and compartments that would not be expected to generate an HGL. If it was determined that HGL formation was possible, the time to HGL formation was estimated and compared to the time required for fire brigade response. The licensee has committed to install detectors in certain areas to support HGL assumptions. However, because the licensee assumed their armored cables were of limited combustibility, the licensee did not include an HRR contribution from the cable material burning to their HGL formation. This resulted in lengthening the time for the HGL formation and screening out some target sets in fire compartments where HGL formation is a potential concern. While ONS has few "enclosed" compartments where HGL formation would be a concern, the NRC staff finds insufficient justification for limiting the combustibility of the armored cables and not accounting for the explicit contribution of their potential combustion to the ZOI development and HGL development. However, in its November 19, 2010, (Reference 52) submittal, the licensee reported HGL formation times of at least 33 minutes using these assumptions which can be compared to an expected brigade response time of no more than 20 minutes based on observed fire drills as described in the response to RAI 5-27. The NRC staff finds that including the contribution of the combustion of armored cables is not likely to expand the ZOI or accelerate the HGL formation to precede the fire brigade response to the extent that the substantive estimated fire risk decrease associated with the proposed transition to NFPA 805 will become a risk increase.

Qualified personnel performed a plant walk-down to identify ignition sources and surrounding targets or SSCs in compartments and applied the pre-solved empirical correlation screening tool to assess whether the SSCs were within the ZOI of the ignition source. Based on the fire hazard present, these generalized ZOIs were used to screen from further consideration those ONS-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 detailed Fire PRA used in support of the licensee's application further evaluated the ignition sources determined to adversely affect the operation of credited SSCs. The licensee adjusted the HRR values for a limited number of ignition source types (e.g., cabinets) based on fire modeling insights. For all transient fire HRRs, the 75th percentile HRR was used (Reference 8). Ignition sources determined to adversely affect the operation of credited SSCs were further evaluated in the detailed Fire PRA to support the LAR. In Reference 8, the licensee clarified that the only combustible fluids that require inclusion in the Fire PRA were lubricants.

NUREG-1824, "Verification and Validation of Selected Fire Models for Nuclear Power Plant Applications" (Reference 29), documents the verification and validation (V&V) of fire models used to support applications of RI/PB FPP at nuclear power plants. 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).

By letter dated August 3, 2009 (Reference 8), the licensee identified the use of several empirical correlations that are not addressed in NUREG-1824. The NRC staff reviewed the empirical correlation screening tool methodology, as well as the related material provided in the LAR in order to determine whether the licensee adequately demonstrated alignment with specific portions of the applicable NUREG-1824 guidance.

In addition, the licensee also responded with a detailed listing of the fire models and empirical correlations used in the screening tool including the specific versions of the software packages used. The response also provided detailed information regarding the correlations and fire models used to support transition, as well as a cross reference between major sections of American Society for Testing and Materials (ASTM) guidance document ASTM E 1355-05a, "Standard Guide for Evaluating Predictive Capability of Deterministic Fire Models" (Reference 30) and the correlations in terms of their applicability and validation. Included in the discussion was a summary of the treatment of ZOI of electrical panels.

For the fire modeling screening tool, documented in the LAR and associated RAI responses, the NRC staff reviewed the quality assurance process requirements of NFPA 805 Section 2.7.3 for performing V&V, limiting the application of acceptable methods and models to within prescribed boundaries, ensuring that personnel applying acceptable methods are qualified, and performing uncertainty analysis. The NRC staff assessed the acceptability of the application of each empirical correlation based on the adequacy of the V&V documentation and the correlation's applicability within its limits. Specifically, the NRC staff used the following criteria in assessing the acceptability of each correlation:

- the empirical correlation is included in a fire model for which V&V has been completed and documented in NUREG-1824, and the correlation is applied within the limits of its applicability; or

- the empirical correlation is widely accepted and utilized by fire protection engineering professionals, is documented in an authoritative publication of the Society of Fire Protection Engineers (SFPE) (e.g., *The SFPE Handbook of Fire Protection Engineering*), and is applied within the limits of its applicability; or

- the empirical correlation has been subjected to a peer review, is published in a widely recognized peer-reviewed journal article or in a conference report (e.g., *Fire Safety Journal*), and is applied within the limits of its applicability.

Based on these criteria, the NRC staff found the application of each of the empirical correlations in the Fire PRA application acceptable. Table 3.4-3 in SE Attachment C provides a summary of the correlations used, how each was applied in the Fire PRA, the V&V basis, and the NRC staff's evaluation.

The licensee indicated that, in general, the criteria and modeling techniques referenced in NUREG/CR-6850 and the empirical correlation screening tool have been the primary tools used for fire modeling in the development of the Fire PRA analysis. However, the licensee's fire modeling used for determining the ZOI of postulated fire scenarios and for the determination of the critical fire size needed for HGL formation in the compartments of interest were different from those referenced in NUREG/CR-6850. Reviews of those deviations from NUREG/CR-6850 are also addressed in Table 3.4-3, in SE Attachment C.

The NRC staff's evaluation finds the fire modeling employed by the licensee in the development of the Fire PRA utilized empirical correlations that provide bounding solutions for the ZOI or utilized conservative input parameters in the application of the correlations resulting in conservative results for the ZOI assuming timely manual suppression is successful. Although the assumption that the fire does not ignite any additional combustible material beyond the original ignition source can be non-conservative, the NRC staff concludes that there are only a few configurations that might be affected by refining the analysis to include this fire propagation and that these few configurations could not increase the change in risk estimates to change the substantial estimated risk decrease into a risk increase. Completion of all implementation items in Section 2.9 (including completing a peer review of the fire PRA, resolving the findings from the review, and revaluating the change in risk from transition) will further confirm that implementation of the RI/PB FPP will result in a risk decrease. Therefore, the NRC staff finds that this approach provides reasonable assurance that these aspects of the fire modeling used in the development of the fire scenarios in the Fire PRA is acceptable for use in this application.

PRA Quality Conclusions

The PRA models (internal events and fire) have been reviewed against the applicable PRA standards. All F&Os from the reviews have been investigated and addressed by the licensee for this application. Based on the NRC staff's review of the peer review results and the licensee's responses as summarized above and in Attachment C of this SE, the NRC staff concludes that changes to the PRA to resolve F&Os from the internal events and fire PRA reviews are not expected to change the substantial estimated risk decrease into a risk increase. Completion of all implementation items in Section 2.9 (including an industry fire PRA peer review, resolution of all peer review comments, and recalculation of the change in risk estimates) will further confirm that implementation of the RI/PB FPP will result in a risk decrease. Therefore, the NRC staff finds that the fire PRA has sufficient technical adequacy that the results can be relied upon to support the determination that the transition to NFPA 805 will result in a decrease in risk.

3.4.4. Additional Risk Presented by Recovery Actions

The NRC staff reviewed LAR Attachment C, "NEI 04-02 Table B-3 - Transition," Attachment G, "Operator Manual Actions Transition," and Attachment K, "Licensing Action Transition."

SE Section 3.2.4 describes the evaluation and transition of Operator Manual Actions (OMAs) to recovery actions. Each VFDR was evaluated to determine if a new recovery action would be relied on to disposition the VFDR.

For fire areas that utilized a previously approved SSF strategy, the licensee used the guidance in RG 1.205 Revision 1 to identify recovery actions. This included consideration of Primary Control Station (PCS) and the definition of recovery action as clarified in RG 1.205, Revision 1.

Based on the definition provided in RG 1.205, the ONS PCS actions are defined as:

- Actions inside the main control rooms,
- Actions inside the SSF control room,
- Actions inside the SSF facility to transfer control from the MCR to the SSF, and
- Actions inside the SSF facility to operate manual valves.

Any actions required to transfer control to, or operate equipment from the PCS, while required as part of the RI/PB FPP, were not considered recovery actions per the RG 1.205 guidance and any additional risk associated with these recovery actions need not be calculated. The only recovery actions the licensee identified as previously approved OMAs, were actions taken to deploy and operate the SSF submersible pump. By letter dated November 19, 2010, (Reference 52) the licensee estimated the increase in CDF associated with the fire induced loss of equipment requiring this recovery action as 2.3E-8/year. The staff finds that the guidance in RG 1.205 on how the risk of recovery actions should be evaluated has been met, and that this CDF increase is sufficiently small that the risk acceptance guidelines associated with pre-approved recovery actions have all been met. All other previously approved OMAs are associated with the main control room, the PCS, or transfer of control to a PCS.

The licensee established 12 new recovery actions that are relied on as part of the resolution of VFDRs. The three fire areas where new recovery actions were established are RB Unit 1, RB Unit 2, and RB Unit 3. These actions are described in LAR Table G-2.

As described in LAR Section G.4.2, the additional risk of a recovery action is conservatively taken as the CDF and LERF associated with the VFDR that resulted in the need for the recovery action. The additional risk of the new recovery actions is estimated to be negligible as reported in LAR Tables W-2, W-3, and W-4 (References 12). The licensee reviewed all of the recovery actions for adverse impact and dispositioned each action as stated in LAR Attachment G Section G.3.2. None of the OMA's listed in LAR Table G-2 that were identified as recovery actions were found to have an adverse impact on the Fire PRA.

The NRC staff has reviewed the licensee's evaluations for the additional risk of recovery actions and finds that the approach applied is acceptable because it utilizes the definition of recovery actions in NFPA 805 and RG 1.205, conservatively estimates the risk of previously approved and new recovery actions, and the risk associated with the new and the previously approved recovery actions are included appropriately in the change in risk estimates.

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

Alternatives to Compliance With NFPA 805, Section 50.48(c)(4):

> The final rule provides licensees the flexibility of requesting, via a license amendment, to use risk-informed or performance-based alternatives that deviate from compliance with NFPA 805. The NRC recognizes that licensees may propose acceptable approaches that are not encompassed by the criteria in NFPA 805. Therefore, the NRC is including a provision for requesting such approaches in the rule. However, to ensure adequate protection of public health and safety, the NRC is requiring that licensees obtain NRC

review and approval to use those methods, and is providing criteria in Section 50.48(c)(4) for review of their acceptability.

Final Rule, Voluntary Fire Protection Requirements for Light Water Reactors; Adoption of NFPA 805 as a Risk-Informed, Performance-Based Alternative, (**69 FR 33,543**) (June 16, 2004).

The licensee made no requests under 50.48(c)(4).

3.4.6. Cumulative Risk and Combined Changes

NFPA 805, Section 2.4.4.1, "Risk Acceptance Criteria," states the following:

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

The acceptability guidelines for changes to plant risk are described in RG 1.174. RG 1.205 further clarifies that changes in risk are to be judged on a fire-area by fire area-basis, as well as the total change in risk. RG 1.205 also clarifies that the additional risk from previously approved recovery actions may be reported separately and treated differently than changes in risk from other plant changes required to be estimated during transition to 10 CFR 50.48(c).

As allowed by RG 1.174 (combined change request) and RG 1.205 for transition, credit for selected non-fire related modifications (e.g., PSW modifications) that affect the Fire PRA results can be considered. [[

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D. Cumulative Changes in Risk

During transition, the cumulative risk is addressed by including all RI changes in the risk estimates. The licensee reported that its estimate of the internal events CDF and LERF for Unit 3 are [[ ]] per year and[[ ]] per year, respectively. Summing the internal events and Fire PRA risk estimates, crediting the PSW modification, yields CDF and LERF estimates for Unit 3 of [[ ]] per year and [[ ]] per year, respectively. As described above, implementation of NFPA-805 and installation of the PSW is expected to reduce risk. As stated in RG 1.174, if the application is clearly shown to result in a decrease in risk (i.e., CDF and LERF), then the change is considered to have satisfied the relevant principle of risk-informed regulation (Principle 4 of RG 1.174) regardless of the total plant risk. Therefore, the licensee did not need to provide estimates for the contribution from other external events such as seismic, external floods, high winds, and tornados.

The licensee reported the total change in fire CDF and fire LERF for each fire area with and without credit for the PSW in LAR Tables W-2, W-3, and W-4. Credit for the fire detection modifications is already included in all these results. The reported results include the risk increases associated with accepting the VFDRs without credit for the PSW in all fire areas and the risk decreases from credit for the PSW in all fire areas. By letter dated November 19, 2010, (Reference 52), the licensee identified a number of additional VFDRs in the AB that require an evaluation of their risk impact but that are not included in the LAR. These VFDRs occur if the licensee assumes that damage from fires can occur at any time, not only after 10-minutes as assumed in the current licensing basis. The licensee stated that these additional VFDRs would arise from possible spurious actions within the first 10-minutes of confirmation of an active fire.

The licensee reported that the initial results of its evaluation of potential risk increases from these additional VFDRs in Unit 3 are [[ ]] and [[ ]] for CDF and LERF respectively. These values are reflected in Tables 3.4 and 3.5 of this SE by increasing the appropriate estimates in the Units 1 and 2 and Unit 3 AB results. Due to the significant risk reduction attributed to the PSW modification, and based on the preliminary sensitivity analyses performed by the licensee, none of these additional VFDRs/recovery actions are expected to change this application from a risk decrease to a risk increase. In addition, the licensee has identified implementation items in Section 2.9 of this SE to ensure their resolutions will maintain this application as a risk decrease including, if necessary, to make plant modifications to address these VFDRs.

The reported results include the risk increases associated with accepting the VFDRs without credit for the PSW in all fire areas, and the risk decreases from credit for the PSW in all fire areas. The net change in risk including both VFDRs and the new PSW for each area except the reactor building meet the "very small" increase acceptance guidelines for CDF and LERF of [[ ]] and [[ ]], respectively. The increases in LERF from accepting VFDRs in the each unit's RB are about [[ ]] with or without credit for the PSW. These increases in LERF values[[ ]] the 1E-7/year "very small" LERF increase acceptance guideline in RG 1.174, but [[ ]] "small" LERF increase acceptance guideline (1E-6/year). The NRC staff finds that the licensee has reported the individual results required by

NFPA 805, RG 1.200, and RG 1.174 Section 2.1.1 on combined change requests, and finds the individual results acceptable in light of the overall risk decrease associated with the PSW modification.

The changes in total fire risk for each unit are provided in the following table. RG 1.174 directs that the total change in risk from all hazard events (internal, fires, external floods, etc.) associated with the proposed change be evaluated and compared to the acceptance guidelines. As shown in the table, there is an overall decrease in fire risk associated with the changes proposed in this LAR due to the significant decrease achieved by the installation of the PSW.

Table 3.4: Fire CDF and LERF for ONS

	Unit 1		Unit 2		Unit 3	
	Fire CDF	**Fire LERF**	**Fire CDF**	**Fire LERF**	**Fire CDF**	**Fire LERF**
Risk increase from accepting VFDRs	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
Risk decrease from PSW installation						
Total Change in Risk						
Final Total Fire Risk including PSW						

Therefore, the NRC staff finds that the combined change request is acceptable because of the overall fire risk decrease associated with the installation of the PSW modification.

3.4.7. Conclusion for Section 3.4

Based on the NRC staff's review of the information provided by the licensee in the LAR, Transition Report, and associated RAI responses, the NRC staff review finds:

1. 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 evaluation) is of sufficient quality to support the application to transition to NFPA-805, because the NRC staff concludes that the weaknesses and limitations, discussed in SE Attachment C, are not expected to change the substantial estimated risk decrease into a risk increase.

2. The licensee's resolution of numerous PRA review F&Os, discussed in SE Attachment C, are directed toward determining that resolving the issues would not change the substantial estimated risk decrease associated with transitioning to 10 CFR 50.48(c) into a risk increase. However, given the number of F&O resolutions that are not fully complete or have not been implemented and will involve PRA method and model changes, the licensee has committed to complete several implementation items identified in Sections 2.9 to further confirm that implementation of the RI/PB FPP will

result in a risk decrease. The following items are identified as implementation items in SE Section 2.9:

a. the licensee updates or upgrades its HRA methodology and estimates of human failure probability and, if needed, completes a peer review of this analysis, and resolves all the findings from this peer review,

b. the licensee upgrades its Fire PRA, completes a peer review of its upgraded Fire PRA, and resolves all the findings from this peer review,

c. The licensee includes in its PRA the as-built PSW system and completes the modeling of any additional VFDRs that are caused by assuming that damage from fires can occur at any time, not only after 10 minutes, and

d. the licensee confirms that there was a reduction in risk associated with transition to NFPA after completing the improvements to the PRA, modeling any additional VFDRs, and modeling the as-built PSW system in the PRA.

3. The plant change process included a detailed review of fire protection DID and SM. The evaluations provided by the licensee are acceptable because the licensee's process followed the endorsed guidance in NEI 04-02, Revision 2 and is consistent with the approved NRC staff guidance in RG 1.205, Revision 1.

4. The additional risk presented by the use of recovery actions was determined and provided in accordance with the guidance in RG 1.205 Revision 1 and NFPA 805 Section 4.2.4. The risk of those recovery actions was found to be acceptable since they were below the acceptance guidelines in RG 1.205, Revision 1, and RG 1.174.

5. The licensee did not request approval of any risk informed or performance-based alternatives to compliance to NFPA 805.

6. The licensee's application is a combined change, as defined by RG 1.205, Revision 1, which combines risk increases identified in the FREs with risk decreases due to other modifications (e.g., PSW). The combined change process is consistent with RG 1.174 and RG 1.205 and is acceptable.

7. The changes in risk (i.e., ΔCDF and ΔLERF) associated with the proposed alternatives to compliance with the deterministic criteria of NFPA 805 (plant change evaluations and FREs) are consistent with RG 1.205 and RG 1.174 guidelines and are acceptable.

3.5. Nuclear Safety Capability Assessment Results

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

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

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

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

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

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

3.5.1. Nuclear Safety Capability Assessment Results by Fire Area

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

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

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

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

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

In accordance with the above, the process defined in NFPA 805, Chapter 4 provides a framework to select either a deterministic or a PB approach to meeting 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 needed 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 the following:

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

This section of the SE evaluates the approach used to meet the 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 Section 4.2.4, "Fire Area-by-Fire Area Transition," Attachment C, "NEI 04-02 Table B-3 - Fire Area Transition," Attachment G, "Operator Manual Actions - Transition to Recovery Actions," Attachment S, "Plant Modifications" and Attachment W, "Fire PRA Insights" (Reference 11) during its evaluation of the ability of each fire area to meet the nuclear safety performance criteria of NFPA 805.

The ONS is a three-unit plant and is divided into 15 fire areas. Based on the information provided by the licensee in the LAR, as supplemented, the licensee performed the NSCA on a fire area basis for each of those fire areas. LAR Attachment C provides the results of these analyses on a fire area basis. For each fire area, the licensee documented the following:

- The approach used in accordance with NFPA 805 (i.e., the deterministic approach in accordance with NFPA 805, Section 4.2.3, or the PB approach in accordance with NFPA 805, Section 4.2.4).
- The SSCs required in order 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.
- The disposition of each VFDR using either modifications (completed or committed) or the performance of a FRE in accordance with NFPA 805, Section 4.2.4.2.

For a fire in the fire areas of the Units 1 and 2 Blockhouse (BH1/2), the Unit 3 Blockhouse (BH3), the RBs (RB1, RB2, and RB3), the SSF, the TB, the west penetration rooms (WP1, WP2, and WP3), or yard, safe and stable plant conditions are achieved utilizing the PSW pump and credited systems powered from the PSW power supply and controlled from the main control rooms. For a fire in Fire Areas CT4 Blockhouse (CT-4), Keowee Hydro Station (KEO), or PSW building, normal shutdown systems are not affected by the fire and the plant will be shutdown, if desired, using normal shutdown systems and operating procedures.

For a fire in the fire area AB the SSF will perform as a dedicated shutdown facility used to establish safe and stable plant conditions.

The licensee also performed a detailed analysis of fire protection DID with respect to fire detection and fire suppression systems for each fire area. LAR Section 4.8 includes a detailed listing of the fire areas, fire zones, and fire protection features necessary to meet the requirements of NFPA 805. LAR Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features," provides a detailed listing of the fire areas and fire zones at ONS, as well as an indication of whether automatic fire suppression and detection systems are required in these areas. This table identifies those fire areas/zones where automatic suppression and detection system modifications are required and list the regulatory and/or technical issue that makes the system required.

SE Table 3.5 identifies and briefly describes each fire area at ONS. This table is based on LAR Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features," which was provided by the licensee in LAR Section 4.8, "Summary of Results."

SE Table 3.5 also identifies the NFPA 805 compliance basis for each fire area, as well as the change in risk associated with CDF and LERF, as calculated by the licensee. The change in risk is broken down into three categories: (1) the risk increase due to VFDRs, (2) the risk decrease due to implementation of the PSW modification, and (3) the total change in risk resulting from summing the first two categories. The detailed discussion for each fire area, including the NRC staff's evaluation of the licensee's compliance with the applicable requirements, is contained in SE Attachment D, "Nuclear Safety Capability Assessment Results by Fire Area."

Table 3.5: ONS Fire Area and Compliance Strategy Summary

Fire Area	Fire Area Description	Licensing Actions Credited?	NFPA 805 Compliance Basis	Fire Risk Evaluation Delta Risk		PSW Modification Delta Risk		Total Delta Risk	
				ΔCDF	ΔLERF	ΔCDF	ΔLERF	ΔCDF	ΔLERF
Unit 1									
AB	Auxiliary Building	Yes	4.2.4.2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
BH12	Units 1 & 2 Block House	Yes	4.2.4.2						
BH3	Unit 3 Block House	Yes	4.2.4.2						
RB1	Unit 1 Reactor Building	Yes	4.2.4.2						
RB2	Unit 2 Reactor Building	Yes	4.2.4.2						
RB3	Unit 3 Reactor Building	Yes	4.2.4.2						
SSF	Standby Shutdown Facility	Yes	4.2.4.2						
TB	Turbine Building	Yes	4.2.4.2						
WP1	Unit 1 West Penetration Room	Yes	4.2.4.2						
WP2	Unit 2 West Penetration Room	Yes	4.2.4.2						
WP3	Unit 3 West Penetration Room	Yes	4.2.4.2						
YARD	Yard	Yes	4.2.4.2						
Unit 2									
AB	Auxiliary Building	Yes	4.2.4.2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
BH12	Units 1 & 2 Block House	Yes	4.2.4.2						
BH3	Unit 3 Block House	Yes	4.2.4.2						
RB1	Unit 1 Reactor Building	Yes	4.2.4.2						
RB2	Unit 2 Reactor Building	Yes	4.2.4.2						
RB3	Unit 3 Reactor Building	Yes	4.2.4.2						
SSF	Standby Shutdown Facility	Yes	4.2.4.2						
TB	Turbine Building	Yes	4.2.4.2						
WP1	Unit 1 West Penetration Room	Yes	4.2.4.2						
WP2	Unit 2 West Penetration Room	Yes	4.2.4.2						
WP3	Unit 3 West Penetration Room	Yes	4.2.4.2						
YARD	Yard	Yes	4.2.4.2	ε	ε	ε	ε	ε	ε
Unit 3									

Table 3.5: ONS Fire Area and Compliance Strategy Summary

Fire Area	Fire Area Description	Licensing Actions Credited?	NFPA 805 Compliance Basis	Fire Risk Evaluation Delta Risk		PSW Modification Delta Risk		Total Delta Risk	
				ΔCDF	ΔLERF	ΔCDF	ΔLERF	ΔCDF	ΔLERF
AB	Auxiliary Building	Yes	4.2.4.2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
BH12	Units 1 & 2 Block House	Yes	4.2.4.2						
BH3	Unit 3 Block House	Yes	4.2.4.2						
RB1	Unit 1 Reactor Building	Yes	4.2.4.2						
RB2	Unit 2 Reactor Building	Yes	4.2.4.2						
RB3	Unit 3 Reactor Building	Yes	4.2.4.2						
SSF	Standby Shutdown Facility	Yes	4.2.4.2						
TB	Turbine Building	Yes	4.2.4.2						
WP1	Unit 1 West Penetration Room	Yes	4.2.4.2						
WP2	Unit 2 West Penetration Room	Yes	4.2.4.2						
WP3	Unit 3 West Penetration Room	Yes	4.2.4.2						
YARD	Yard	Yes	4.2.4.2						
Other									
CT-4	CT-4 Block House	No	4.2.3.2	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]	[[ ]]
KEO	Keowee Hydro Station	No	4.2.3.2						
PSW	Protected Service Water	No	4.2.3.2						

Note: N/A – Not Applicable, applies to those fire areas that are deterministically compliant in accordance with NFPA 805, Section 4.2.3.

NI – Not Included, applies to the planned PSW structure, which was not included as a fire compartment in the Fire PRA; the licensee states that the additional risk from a PSW fire is insignificant.

ε – The delta risk is epsilon or negligible. Plant equipment associated with the VFDRs in the fire area were evaluated and not modeled in the Fire PRA and are not expected by the licensee to introduce any significant risk contributors to the risk being evaluated for the FREs.

SE Attachment D is broken down into those fire areas that were analyzed using the deterministic approach in accordance with NFPA 805, Section 4.2.3, and those using the PB approach in accordance with NFPA 805, Section 4.2.4. Each fire area 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.

SE Attachment D also addresses NRC staff-approved exemptions and other licensing actions that exempt the licensee from the existing deterministic licensing basis that the licensee desires to bring into the RI/PB FPP as allowed by NFPA 805, Section 2.2.7. The attachment includes a description of the previously approved exemption or other licensing action that exempts the licensee from the deterministic requirements, the basis and continuing validity of the exemption or other licensing action, and the NRC staff's evaluation of that exemption or licensing action. The licensee stated in LAR Section 4.2.3, "Licensing Action Transition," that the review of these existing licensing actions included a determination of the basis of acceptability and a determination that the basis of acceptability was still valid.

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 and systems addressed in these sections are only required when the analyses performed in accordance with NFPA 805, Chapter 4, indicate that the features and systems are required to meet the nuclear safety performance criteria.

Passive fire protection features address the fire barriers used to form fire area boundaries (and barriers separating SSD trains) that were previously reviewed and approved in accordance with the licensee's existing deterministic FPP. For its transition to NFPA 805, the licensee decided to retain most of the previously approved fire area boundaries as part of the RI/PB FPP.

The fire barrier fire resistance rating required for separation between fire areas under NFPA 805 (3 hours) is the same as that required under Appendix R (3 hours). Accordingly, based on the previously approved fire area boundaries continuing to meet the NFPA 805 fire barrier acceptance criteria, the NRC staff finds retaining these passive fire protection features acceptable.

For its transition to NFPA 805, the licensee has also decided to create two new fire areas, the TB and AB fire areas, from the previously approved balance-of-plant fire area, as part of the RI/PB FPP. The licensee plans to upgrade the fire barriers separating the TB from the AB and separating the AB from the west penetration room to have a 3-hour fire resistance rating as described in LAR Attachments A, C, and S. The licensee has also committed to upgrade the fire barriers between the purge inlet rooms and the spent fuel pool (SFP) area (AB Fire Area) to have a 3-hour fire resistance rating. Based on the licensee's commitment to upgrade these fire barriers to meet the NFPA 805 fire barrier acceptance criteria, the NRC staff finds these new passive fire protection features acceptable. The fire barriers being upgraded are described in SE Section 2.8.

The licensee's FREs identified the need to improve general area and/or hazard detection in several fire areas, either to support assumptions made in the Fire PRA or to provide DID. The licensee plans to upgrade and/or install new automatic fire detection systems in many fire zones throughout the plant, which are identified by fire area in SE Attachment D. In response to an NRC staff RAI (Reference 12), the licensee has stated that the upgraded and newly installed fire detection systems will be installed in accordance with NFPA 72, *National Fire Alarm Code*, as required by NFPA 805. Based on the licensee's commitment to upgrade existing fire detection systems and install new fire detection systems to meet the NFPA 805 criteria, the NRC staff finds these upgraded/new fire detection systems acceptable. SE Section 2.8 describes the fire detection system modification.

In addition to the above, SE Attachment D provides an evaluation of the credited recovery actions for each applicable fire area. As discussed in SE Section 3.2.4, the licensee credited recovery actions to satisfy the DID requirements of NFPA 805, Section 1.2, but are not needed to maintain the availability of a success path and do not adversely impact risk. Because the licensee has identified these recovery actions as being necessary to provide adequate DID, the NRC staff has evaluated them as a part of the RI/PB FPP. As such, future removal of these recovery actions would require a plant change evaluation in accordance with NFPA 805, Section 2.4.4.

Finally, as a part of the NSCA, the licensee evaluated fire detection and suppression systems on a fire zone basis. In SE Attachment D, the evaluation of each fire area includes a table that documents the licensee's review of these fire detection and suppression systems, as well as the NRC staff's evaluation of the review and its results.

As documented in SE Attachment D, for those fire areas that utilized a deterministic approach in accordance with NFPA 805, Section 4.2.3, the NRC staff finds that each of the fire areas analyzed using the deterministic approach meets the associated criteria of NFPA 805, Section 4.2.3.2. This conclusion is based on (1) the licensee's documented compliance with NFPA 805, Section 4.2.3.2; (2) the licensee's assertion that the success path will be free of fire damage without reliance on recovery actions; (3) an assessment that the suppression systems in the fire area will have no impact on the ability to meet the nuclear safety performance criteria; and, (4) the licensee's appropriate determination of the automatic fire suppression and detection systems required to meet the nuclear safety performance criteria.

In addition, for those fire areas that utilized the PB approach in accordance with NFPA 805, Section 4.2.4, the NRC staff finds that each fire area has been properly analyzed, and compliance with the NFPA 805 requirements demonstrated as follows:

- Exemptions and other licensing actions that exempt the licensee from the existing fire protection licensing basis were reviewed for applicability, as well as continued validity, and found acceptable.

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

- Recovery actions used to demonstrate the availability of a success path to achieve the nuclear safety performance criteria, or to provide DID, 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.

Accordingly, each fire area utilizing the PB approach was able to achieve and maintain the nuclear safety performance criteria, and the associated FREs meet the applicable NFPA 805 requirements for risk, DID, and SMs.

3.5.2. Fire Protection During Non-Power Operational Modes

NFPA 805 Section 1.1, "Scope," states:

> 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.

Thus, the nuclear safety goal of NFPA 805 requires the evaluation of the effects of a fire during any operational mode and plant configuration, including non-power operation (NPO) modes. In general, the underlying concerns are the differences between the functional requirements (i.e., a different (or additional) set of systems and components) and time dependencies on decay heat removal system operation during NPOs and full-power operations.

The NRC staff reviewed LAR Section 4.3, "Non-Power Operational Modes" and Attachment D, NEI 04-02, Table F-1, "Non-Power Operational Modes Compliance," to evaluate the licensee's treatment of potential fire impacts during NPOs. The licensee used the process from NEI 04-02 (Reference 21), for demonstrating that the nuclear safety performance criteria are met for Higher Risk Evolutions (HREs) during NPO modes.

To clarify the guidance from NEI 04-02, on providing "reasonable assurance that a fire during non-power operations will not prevent the plant from achieving and maintaining the fuel in a safe and stable condition." the NRC staff issued interim guidance in FAQ 07-0040, "Non-Power Operations Clarification," Revision 4. Specifically, FAQ 07-0040 clarifies the following:

- The process for selecting equipment and cabling to evaluate during NPO modes.
- Evaluation of HREs during NPO modes.
- The process for analyzing key safety functions (KSFs) in different plant operating states (POSs).

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

As discussed in FAQ 07-0040, protection of equipment during NPO modes includes a combination of the normal FPP DID actions and additional RI steps based on the availability of systems and equipment needed to support KSFs, and whether or not the plant is in an HRE. The licensee states that its strategy for control and protection of equipment during NPO modes includes a combination of normal fire protection DID actions, additional RI steps based on the availability of systems and equipment needed to support KSFs, and whether or not the plant is in an HRE.

The licensee defines KSFs as:

- decay heat removal,
- reactor coolant system (RCS) inventory control,
- reactivity control, and
- power availability, including support functions.

The licensee determined that containment closure was not a KSF since it does not directly support the nuclear safety goals of NFPA 805. However, the licensee identified the importance of demonstrating with high confidence that the equipment hatch can be closed prior to a release that would exceed the NFPA 805 radiological release criteria. Establishing high confidence would require the implementation of additional fire protection DID actions. Developing a process to evaluate the potential effects of fire on habitability and the impact of additional DID actions is an implementation Item 16.

As discussed in FAQ 07-0040, each plant may have a unique definition of what constitutes a higher risk evolution. However, the definition should consider the following:

- time to boil
- reactor coolant system and fuel pool inventory
- decay heat removal capability

In LAR Attachment D, the licensee defines an HRE as outage activities, plant configurations or conditions during shutdown where the plant is more susceptible to an event causing the loss of a KSF. The licensee further states that HREs include:

- Draining to reduced inventory when reactor coolant level is at or below the reactor vessel flange.
- Reactor coolant system at or below reduced inventory
- Midloop operation
- Any specific evolution determined by station management

Reduced Inventory is further defined by the licensee as a configuration with fuel in the reactor vessel and level less than 50" above the centerline of the reactor vessel hot leg. The licensee states that decay heat removal capability and time to boiling isaddressed in its shutdown risk management procedure (NSD-403) with the term Thermal Margin, which is the time to core boiling upon loss of decay heat removal.

The licensee states that it used the process from NEI 04-02, as clarified by FAQ 07-0040, to demonstrate that nuclear safety performance criteria of NFPA 805 are met for HREs during NPO modes. This process includes the following steps:

- Review the existing outage management processes.
- Identify equipment/cables.
- Review plant systems to determine success paths that support each of the DID KSFs.
- Identify cables required for the selected components and determine their routing.
- Perform fire area assessments (identify pinch points).
- Manage risk associated with fire-induced vulnerabilities during the outage.

The same basic methodology utilized for the nuclear capability safety assessment is used when assessing the impact of fire on nuclear safety during NPO modes. The licensee states that KSF are identified in Shutdown Risk Management Procedure NSD 403 and additional detail is provided in SD 1.3.5, Shutdown Protection Plan. Thus, the licensee's evaluation focused on those sets of systems, components and equipment that are required to ensure that the KSF's defined in these procedures can be maintained during potential HRE's. The licensee is revising Fleet Directive NSD-403 and SD 1.3.5, definition of "high risk evolution," to address NPO criteria and to reconcile thermal margin criteria with the criteria in FAQ 07-0040. These revisions are an implementation item (SE Section 2.9, Table 2.9-1, Item 15).

The process used by the licensee to identify the systems and equipment to be included in the NPO review began with the identification of the POSs that need to be considered. The POSs identified are those provided in FAQ 07-0040 for PWRs, which are consistent with those contained in Attachment 2 to Appendix G, "Phase 2 Significance Determination Process Template for PWR during Shutdown," of NRC Inspection Manual Chapter (IMC) 0609, "Significance Determination Process" (Reference 46). For other non-power conditions (e.g., defueled reactor vessel), normal FPP controls, processes and procedures will be used.

After identifying the plant-specific POSs that require additional equipment to be included in the NPO review, the licensee states that it performed the following:

1. Determined the KSFs that support the POS of concern.
2. Identified the equipment relied upon to provide the KSFs, including support functions, during the POS to be evaluated. This information was then entered into the Appendix R Database Management System (ARTRAK) SSD database to facilitate sorting of the component and cable information on a fire zone by fire zone basis. For those components not already in ARTRAK, cable selection and routing was performed as per the nuclear safety

methodology. The nuclear safety capability analysis methodology identified all required cables associated with a component and does not perform a circuit analysis until the area-by-area compliance assessment. As a result, according to the licensee, a conservative population of components and cables was identified for NPO. The resulting information was entered into ARTRAK and a series of NPO reports were developed within the software to allow the 'pinch point' analysis to be performed.

3. Utilizing the fire zone cable routing and equipment location information from ARTRAK, the licensee analyzed the KSF success paths on a fire zone by fire zone basis to assess the impact of a single fire.

4. Analyzed circuits of equipment not already credited (or credited in a different way, such as on versus off, open versus closed, etc.), in accordance with the nuclear safety methodology and identified additional cables to be included in the NPO review.

The licensee states that the current outage management procedures and site directives do not include all of the conditions applicable to the POSs reviewed in the NPO evaluation. To address this finding the licensee states the following activities will be performed:

- Develop procedural guidance to monitor BWST temperature before freezing occurs.

- Develop procedural controls to monitor lake levels and the availability of the reverse gravity condenser circulating water flow path during HREs.

- Develop procedural controls to use RCS wide-range pressure instruments, in lieu of reactor coolant (RC) low-range pressure, during HREs.

- Develop procedural controls to monitor the "A" Train BHUT level.

- Ensure capability for an operator to access motor-operated valves (MOVs) 1, 2, 3LP-21 (or 1, 2, 3LP-22) where 1, 2, 3DHR-GF1&2 success paths are credited.

- Ensure capability for an operator to access manual valves 1, 2, 3HP-363 and 1, 2, 3HP-78, where 1, 2, 3INVCTL3c success paths are credited.

Completion of each of these activities is an implementation item (SE Section 2.9, Table 2.9-1, Items 17 through 22, respectively).

Based on its review of the information provided in the LAR, the NRC staff concludes that the licensee used methods consistent with the interim 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.

Components that were identified as needed to support an NPO KSF but were not included on the post-fire SSD equipment list required additional circuit analysis. The licensee loaded that information into the ARTRAK database, which allowed sorting of the component and cable information on a fire zone by fire zone basis. Utilizing the fire zone cable routing and equipment location information from ARTRAK, the licensee's evaluation of NPO fire impacts focused on analyzing the KSF success paths on a fire zone by fire zone basis in order to assess the impact of a single fire. Those fire zones with KSF success path impacts were identified and categorized based on fire risk vulnerability. Recommendations to establish additional fire protection/fire prevention actions during HREs by fire zone were developed based on the assessed fire risk vulnerability. Due to the lack of rated fire barriers between all fire zones, additional fire prevention recommendations were made for fire zones where compartment to compartment interactions could potentially take place.

The licensee documented its analysis of the impact of a fire in each fire zone on the success paths for the KSFs, and recommendations of changes to fire risk and outage management procedures, in a site-specific calculation. Consistent with FAQ 07-0040, the recommendations of the site-specific NPO fire impact calculation apply only to those fire zones where fires could cause the complete loss of a KSF (pinch point). Fire modeling was not used to eliminate any fire zone from being a pinch point. Specific examples of recommendations include:

- Prohibition or limitation of hot work in fire zones during periods of increased vulnerability.
- Limitation of combustible materials in fire zones during periods of increased vulnerability.
- Plant configuration changes (e.g., removing power from equipment once it is placed in its desired position). The licensee states that it will develop procedures to realign and remove power from the MOVs in the unit-specific gravity feed flow paths prior to entering HREs to preclude spurious operation. Development of these procedures is an implementation item (SE Section 2.9, Table 2.9-1, Item 24).
- Provision of additional fire patrols at periodic intervals or other appropriate compensatory measures (such as surveillance cameras) during periods of increased vulnerability.
- Reschedule the work to a period with lower risk or higher DID.
- Crediting of committed modifications (e.g., PSW HPI System).

The licensee states that it will update the NPO evaluation of fire impacts on KSF success paths following the installation of the NFPA 805 committed modifications noted in SE Section 2.8.1. Completion of this update is an implementation item (SE Section 2.9, Table 2.9-1, Item 23).

The licensee states that it does not currently rely on the use recovery actions to restore KSFs. The licensee further states that recommendations resulting from its review will be incorporated into appropriate plant procedures prior to implementation of NFPA 805. Development of these procedure changes is an implementation item (SE Section 2.9, Table 2.9-1, Item 25).

In accordance with the method endorsed in NEI 04-02 and FAQ 07-0040, the primary mechanism being used to meet the nuclear safety performance criteria during NPO conditions is through the use of normal fire protection defense-in-depth (FP DID) actions to reduce the risk of fire. Specific examples include, but are not limited to, additional use of fire watch patrols or other appropriate measures (such as surveillance cameras) and establishing appropriate administrative controls to govern ignition sources, hot work and combustible materials. During HREs, this is achieved by implementing enhanced FP DID actions, specific examples of which are identified above, that reduce the frequency, severity or impact of fires such that the key pinch points are protected. During non-HREs, this is achieved by implementing the normal FP DID actions throughout the plant.

The NRC staff reviewed the breaker coordination study. In letters dated August 3, 2009 (Reference 8) and April 14, 2010 (Reference 11), the licensee agreed that the coordination study currently relied on in the LAR, as supplemented, requires further enhancement to meet Section 2.4.2.2.2 of NFPA 805 and that a revised breaker coordination study was underway. By letter dated September 13, 2010 (Reference 12) the licensee stated that the revised breaker coordination study had been completed and identified modifications to four breakers that have an overall risk increase due to their lack of coordination with the upstream protective device. The four breakers are being modified to maintain the Fire PRA risk profile reported in the LAR (see SE Section 3.4 for a more detailed discussion). The plant modifications are described in SE Section 2.8, Table 2.8.1-1.

In the cited supplementary information, the licensee stated that the revised study included the coordination of electrical protective devices associated with NPO/KSF power supplies and that any required modifications identified during the breaker coordination study were entered into its corrective action program and appropriate compensatory actions were implemented until the item is fully resolved. The licensee also stated that further analysis was performed for those feeders on the selected power supply, which were shown to be uncoordinated. The cables associated with these uncoordinated feeders were identified and routed by Fire Area in order to determine the impact to the associated Fire Areas/Scenarios. The cables associated with the uncoordinated feeders were documented in ARTRAK and will be utilized as an input to the NSCA, Fire PRA model, and the NPO Pinch Point Analysis.

In the LAR, the licensee also credited its original breaker coordination study to address common enclosure concerns. However, as discussed above, the original ONS coordination study does not satisfy applicable NFPA 805 or NEI 00-01 criteria. In Enclosure 3 of the LAR (Reference 11) the licensee states that the second phase of the revised coordination study included a review of the cable damage curves to determine if the electrical circuit design provides proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before ignition temperature is reached. The results of this review were entered into the ARTRAK database and analyzed in the Fire Area/Fire Zone impacts. All power supplies required by the NSCA, Fire PRA, and NPO Pinch Point Analysis, as identified on the associated equipment list, were included in the breaker coordination study scope of "SSD related" power supplies. The licensee further stated that the coordination study meets the requirements of NFPA 805, Section 2.4.2.2.2, for circuits that share a common enclosure with circuits required to achieve nuclear safety performance criteria. In addition, the licensee states that a review of recent modifications confirms that adequate electrical circuit protection has been maintained as part of the design change process. In addition, the licensee states that the results of the coordination study will be documented in the NSCA, NPO Pinch Point Analysis, and the Fire PRA.

Incorporating information related to cables associated with uncoordinated feeder breakers of credited power supplies into the NSCA and NPO Pinch Point Analysis and updating the Fire PRA model to include the results of the breaker coordination study is an implementation item (SE Section 2.9, Table 2.9-1, Item 33). Updating the breaker coordination study to include all new NFPA 805 SSEL-related power supplies (i.e., PSW) for power and non-power operations, and defining additional plant modification if necessary to ensure that the assumptions of the Fire PRA and NSCA remain valid, is an implementation item (SE Section 2.9, Table 2.9-1, Item 44).

The NRC staff also requested the licensee to provide an evaluation of spurious equipment actuations and/or mal-operations (including multiple spurious operations) during non-power operation modes. In its letter dated November 30, 2009 (Reference 10), the licensee states that site-specific NPO calculations had been revised to provide a greater level of detail in explaining how spurious equipment actuations and/or mal-operations including multiple spurious operations have been analyzed in the evaluation of pinch points for NPO. Specifically, the component and cable selection process was revised to include all component's with a potential for spurious operation including flow blockage and diversion and associated cables causing spurious operations. Any cable hit within a fire zone was considered an adverse impact on the component and any related KSF success path(s), which the licensee stated was conservative. In its September 13, 2010 (Reference 12) letter, the licensee states that any and all potential spurious actuations that may result from intra-cable shorting were considered. Such failures were considered to occur concurrently, regardless of number, in accordance with the guidance provided in NEI 00-01, Section 3.5.1.5[B]. The NRC staff finds that the licensee's overall approach conforms with the endorsed guidance.

Conclusion for Section 3.5.2

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 the following engineering analyses to demonstrate that it meets this requirement:

- Identified the KSFs required to support the nuclear safety performance criteria during NPOs.
- Identified the POSs where further analysis is necessary during NPOs.
- Identified the equipment required to meet the KSFs during the POSs analyzed.
- Identified the location of this equipment and their associated cables.
- Performed analyses on a fire zone basis to identify pinch points where one or more KSFs could be lost as a direct result of fire-induced damage.
- Planned modifications to appropriate station procedures in order to employ one or more fire protection strategies for reducing risk at these pinch points during HREs.

In addition, normal FP DID actions are credited for addressing the risk impact of those fires which potentially affect one or more trains of equipment that provide a KSF required during NPO modes, but would not be expected to cause the total loss of that KSF. 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 at ONS.

3.5.3. Conclusion for SE Section 3.5

The NRC staff reviewed the licensee's RI/PB FPP, as described in the LAR and its supplements, to evaluate the NSCA results. The licensee used a combination of the deterministic approach in accordance with NFPA 805, Section 4.2.3, and the PB approach in accordance with NFPA 805, Section 4.2.4, to perform this assessment at ONS.

For those fire areas that utilized a deterministic approach, the NRC staff confirmed the following:

- None of the exemptions from the existing fire protection licensing basis were credited to meet the deterministic requirements in any of the deterministic fire areas.

- 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.

- No recovery actions were relied on to meet the deterministic requirements in any of the fire areas.

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

The NRC staff found that each fire area utilizing the deterministic approach met the deterministic requirements of NFPA 805, Section 4.2.3.

For those fire areas that utilized the PB approach in accordance with NFPA 805 Section 4.2.4, the NRC staff confirmed that:

- Exemptions from the existing ONS fire protection licensing basis that were previously approved by the NRC, and which are being carried forward by the licensee into the RI/PB FPP, were evaluated and found to be valid and acceptable for meeting the deterministic requirements of NFPA 805 as allowed by NFPA 805, Section 2.2.7.

- Fire suppression effects were evaluated and found to have no adverse impact on the ability to achieve and maintain the nuclear safety performance criteria for each fire area.

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

- No recovery actions were necessary to demonstrate the availability of a success path.

- All recovery actions credited with providing DID were evaluated with respect to the additional risk presented by their use and found to be acceptable in accordance with NFPA 805, Section 4.2.4.

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

Accordingly, the NRC staff has reasonable assurance that the nuclear safety performance criteria will be met for each fire area utilizing the PB approach, in accordance with NFPA 805, Section 4.2.4. Furthermore, the associated FREs meet the requirements for risk, DID, and SMs.

The NRC staff's review of the licensee's analysis for, and outage management process during, NPO modes found that the licensee provided reasonable assurance that the nuclear safety performance criteria will be met during NPO modes and HREs. The staff's review also found that the normal FPP DID actions are credited for addressing the risk impact of those fires which potentially affect one or more trains of equipment that provide a KSF required during NPO modes, but would not be expected to cause the total loss of that KSF. The NRC staff finds this overall approach for fire protection during NPO modes acceptable.

3.6. Radioactive Release Performance Criteria

NFPA 805, Chapter 1 defines the radioactive release goals, objectives, and performance criteria that must be met by the FPP in the event of a fire at a nuclear power plant:

Radioactive Release Goal

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

Radioactive Release Objective

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

(1) Containment integrity is capable of being maintained.

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

Radioactive Release Performance Criteria

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

The NRC staff reviewed LAR Section 4.4, "Radioactive Release Performance Criteria," and Attachment E, "NEI 04-02 Table G-1 Radioactive Release Transition," to evaluate the engineering and procedural controls credited by the licensee to limit potential radioactive releases to unrestricted areas associated with fire fighting activities.

In letter dated April 14, 2010, (Reference 11), the licensee stated that the current operating license for ONS, delineated in Technical Specification 5.5.5b, permits a liquid effluent release limit of 10 times that of 10 CFR Part 20, Appendix B, Table 2, Column 2, and that this NRC-approved limit, (NRC Staff's SE dated January 6, 1993 (ADAMS Accession No. ML012040034) (Reference 55), is the radioactive release performance criteria for liquid effluent releases for ONS. Per the introductory text to Part 20, Table 2, the concentration values given in Column 2 are equivalent to the radionuclide concentrations which, if inhaled or ingested continuously over the course of a year, would produce a total effective dose equivalent (TEDE) of 50 mrem (or 0.5 mSv). A 1-year release having radionuclide concentrations a factor of 10 times the Table 2, Column 2 values would produce a TEDE of 500 mrem (or 5 mSv), the maximum allowed radiation dose limit to individual members of the public permitted by 10 CFR Part 20, Section 1301(d). Since the liquid effluent release limits permitted by the ONS operating license limits is equivalent to the 10 CFR Part 20 maximum permissible radiation dose to the general public, the NRC staff considers the licensee's liquid effluent release limit of 10 times the 10 CFR Part 20, Appendix B, Table 2, Column 2 equivalent to the NFPA 805 radioactive release performance criteria and therefore acceptable. In response to an NRC staff RAI, the licensee stated that the release limits for gaseous effluents at ONS, including smoke release, conform to the NFPA 805 radioactive release performance criteria (Reference 12).

In order to assess whether the ONS FPP to be implemented under NFPA 805 meets the above requirements, the licensee reviewed the existing ONS pre-fire plans and fire brigade training materials. Pre-fire plans that address fire areas where there is no possibility of radioactive materials being present were screened from further review. All other fire zone pre-fire plans were then evaluated to ensure that the locations that have the potential for radioactive release due to fire fighting activities are subject to specific steps for containment and monitoring of potentially contaminated smoke and fire suppression water. Engineering and procedural controls for water release and smoke were then reviewed to determine how effectively the specific steps in the pre-fire plans provide guidelines for the containment and monitoring for potentially contaminated smoke and fire suppression water.

The licensee's review determined the current FPP is compliant with the requirements of NFPA 805 and the guidance in RG 1.205, with the exception of the 10 CFR Part 20 limits for liquid effluent. As discussed above, the NRC staff considers the NRC-approved liquid effluent release limits in the ONS operating license to be in conformance with the maximum permissible dose limits of 10 CFR Part 20. In addition, the licensee stated that during the radioactive release review, a new fire brigade SOG-16 was developed to address smoke management and potentially contaminated water runoff when a fire involves potentially contaminated areas that may not be identified on the pre-fire plans (Reference 11). These areas may include other radioactively contaminated areas that have been established for short-term periods, such as outages and maintenance evolutions.

Table 3.6-1, "ONS Fire Areas and Their Compliance with the NFPA 805 Radioactive Release Performance Criteria," in Attachment E to this SE summarizes, for each fire pre-plan, (1) the fire zone included in the pre-plan, (2) the engineered controls used to minimize radioactive releases generated from the combustion of radioactive materials or from fire suppression activities, and (3) the NRC staff's evaluation of the adequacy of the licensee's methods of controlling and monitoring contaminated suppression agent runoff and combustion smoke.

The licensee also reviewed the fire brigade training materials to ensure they are consistent with the pre-fire plans in terms of containment and monitoring of potentially contaminated smoke and fire suppression water. The licensee's review determined that the existing fire brigade training materials are adequate. In addition, the new SOG-16 described above has been fully implemented into the ONS fire brigade training program.

NFPA 805 requires the licensee to address the nuclear safety and radioactive release goals, objectives and performance criteria in any operational mode. The licensee stated that the radioactive release review was not performed based on plant operating modes, since fire suppression activities, as defined in the pre-fire plans and fire brigade fire fighting instruction operating guidelines, are written for any plant operating mode. During non-power operational modes, the licensee stated that the fire pre-fire plans conservatively assume an "at power" entry condition and do not differentiate between operating and shutdown conditions. In addition, as described previously, a new fire brigade SOG-16 was developed to address smoke management and potentially contaminated water runoff when a fire involves potentially contaminated areas that have been established for short-term periods such as outages.

In letter dated April 14, 2010, (Reference 11), the licensee stated that the existing pre-fire plans adequately address containing and monitoring products of combustion generated from a fire in the RB while the equipment hatch is open.
In letter dated April 14, 2010, (Reference 11), the license stated that the pre-fire plans are controlled documents under the licensee's procedures and within the scope of the configuration

management process. The licensee also stated that the results of the radioactive release reviews will be maintained post-transition by the established ONS Configuration Management Program as described in LAR Section 4.7. (Note: SE Section 3.8 contains the NRC staff's review of the licensee's configuration management processes.)

Based on (1) the information provided in the LAR as supplemented, (2) the licensee's use of pre-fire plans, (3) the results of the NRC staff's evaluation of the identified engineered controls used to control suppression water and combustion products, (4) the development and implementation of a new fire brigade operating guideline regarding control of radiological release, and (5) fire brigade training on monitoring and controlling suppression water runoff and combustion smoke, the NRC staff concludes that the licensee's RI/PB FPP provides reasonable assurance that radiation releases to any unrestricted area due to the direct effects of fire suppression activities at ONS are as low as reasonably achievable and are not expected to exceed the radiological dose limits in 10 CFR Part 20 and the NRC-approved liquid effluent release limit of 10 times that of 10 CFR Part 20, Appendix B, Table 2, Column 2, allowed for in the ONS operating license. In conclusion, the NRC staff finds that the licensee's RI/PB FPP approach aligns with the goals, objectives, and criteria specified in NFPA 805 Sections 1.3.2, 1.4.2, and 1.5.2 and is acceptable.

3.7. Monitoring Program

For this section of the SE, the following requirements from NFPA 805, Section 2.6, are applicable to the NRC staff's review of the licensee's amendment request:

> Monitoring. A monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the FPP in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid.
>
> Availability, Reliability, and Performance Levels. Acceptable levels of availability, reliability, and performance shall be established.
>
> Monitoring Availability, Reliability, and Performance. Methods to monitor availability, reliability, and performance shall be established. The methods shall consider the plant operating experience and industry operating experience.
>
> Corrective Action. If the established levels of availability, reliability, or performance are not met, appropriate corrective actions to return to the established levels shall be implemented. Monitoring shall be continued to ensure that the corrective actions are effective.

The NRC staff reviewed the ONS NFPA 805 Monitoring Program described in LAR Section 4.6, "Monitoring Program" (Reference 11), that the licensee is developing to monitor availability, reliability, and performance of ONS FPP systems and features after the transition to NFPA 805. While the program was still under development at the time the LAR was submitted for review, the focus of the NRC staff's evaluation involved identifying the critical elements related to the program, including the selection of FPP systems and features to be included in the program, the attributes of those systems and features that will be monitored, and the methods for monitoring those attributes. Implementation of the program will occur on the same schedule as the NFPA 805 RI/PB FPP implementation. Completion of the ONS NFPA 805 Monitoring Program is an implementation item (SE Section 2.9, Table 2.9-1, Item 8).

The licensee is developing an ONS-specific calculation to document and describe the methodology and criteria used to select fire protection systems and features for inclusion in the ONS NFPA 805 Monitoring Program. By letter dated September 27, 2010 (Reference 13), the licensee provided a detailed description of the methodology and criteria and stated that a multi-disciplinary review team is being utilized to review and check the calculation. The licensee's review team includes representatives from operations, fire protection, PRA, and SSD, all of whom are experienced and qualified to the licensee's training program for their positions.

The scope of the licensee's monitoring program includes FPP Systems, Structures, and Components (SSCs) and FPP programmatic elements. Supporting engineering evaluations associated with the NFPA 805 transition effort were reviewed to identify SSCs and programmatic elements credited in these supporting evaluations for providing some functional role in reducing fire risk. All SSCs that perform functions or support assumptions credited in the NFPA 805 engineering evaluations to reduce fire risk are evaluated to determine if additional monitoring of the component is required.

The licensee stated that SSCs necessary to meet the NFPA 805 nuclear safety performance criteria are typically monitored as part of Maintenance Rule monitoring (as promulgated in 10 CFR 50.65). Accordingly, the NRC staff finds that the licensee may use the Maintenance Rule for the components covered by that program as a means to meet the requirements of the ONS NFPA 805 Monitoring Program. As such, these systems and equipment will not be included in the ONS NFPA 805 Monitoring Program. However, the licensee will review the NSCA SSCs credited in the NFPA 805 analyses to validate that availability and reliability is monitored as part of the Maintenance Rule and that the criteria is adequate to meet the needs of NFPA 805. If the criteria are not adequate, new Maintenance Rule functions will be created and applied to support NFPA 805 monitoring requirements (Reference 52). Those systems and equipment required to meet the nuclear safety performance criteria that are not included in the Maintenance Rule monitoring program will be reviewed for inclusion in the NFPA 805 Monitoring Program.

The SSCs and programmatic elements to be included in either monitoring program will be monitored for availability and reliability to ensure that the functions credited will be accomplished as assumed in the supporting engineering evaluations. Since a credited function may be performed by a number of individual components for a given fire area, the licensee is establishing Performance Monitoring Groups (PMGs) for each fire zone. PMGs are functional categories of fire protection systems and administrative controls. The table provided in the licensee's letter dated September 27, 2010 (Reference 13) provides the initial list of ONS PMGs.

The licensee has defined screening thresholds, which are being used to determine the most risk-significant fire compartments utilizing the results of the Fire PRA. Those fire compartments (and all PMGs within the compartments) that are determined to be risk significant will be brought into the scope of the ONS NFPA 805 Monitoring Program. In response to a supplemental NRC staff RAI (Reference 52), the licensee identified the following screening thresholds being used to determine either the fire compartments or components, or both, to be included in the scope of the ONS NFPA 805 Monitoring Program:

- CDF greater than or equal to 1.0E-07 per year (on a compartment basis)
- LERF greater than or equal to 1.0E-08 per year (on a compartment basis)
- risk achievement worth (RAW) greater than or equal to 2 (on a PMG)

The licensee has defined High Safety Significant (HSS) fire zones and SSCs as those that exceed the screening criteria. The licensee stated that all FPP SSCs that are in HSS fire zones, and all HSS FPP components, that are amenable to risk measurement will be included in the ONS NFPA 805 Monitoring Program.

The screening criteria being implemented at ONS in regard to the ONS NFPA 805 Monitoring Program are acceptable to the NRC staff based on the following: (1) the CDF and LERF criteria used to screen compartments into the program are consistent with the self approval limits under the RI/PB FPP license condition (see SE Section 4.0), and (2) the NRC staff has previously determined the RAW criteria used for screening individual PMG into the program to be acceptable for use in determining risk significant SSCs that must be monitored under the Maintenance Rule, as described in NUMARC 93-01, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (Reference 32).

The licensee also stated that it will establish criteria for acceptable levels of availability, reliability, and performance, or appropriate action levels, for each PMG. The intent is to establish conservative values of availability and reliability, such that assumptions made in the applicable supporting analyses are bounded. Target and action levels for availability will be primarily based on site-specific data reflecting expected out-of-service times to support maintenance and inspection activities, such that planned impairments with appropriate functional compensatory measures will not be assessed against availability criteria (Reference 52). Target and action values for reliability will be based primarily on industry guidance in EPRI Technical Report (TR) 1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features" (Reference 53), with adjustments to reflect site-specific operating experience, Fire PRA assumptions, and equipment types (and vendor data when available). However, in response to an NRC staff RAI (Reference 52), the licensee stated that availability and reliability targets for NFPA 805 monitored SSCs will be selected, reviewed, and maintained to ensure that the assumptions of the applicable supporting analyses (e.g., Fire PRA, NSCA, etc.) remain valid. Performance of programmatic elements such as fire brigade, fire-watches, and combustible controls will be evaluated using the existing ONS plant health process.

The NRC staff finds that establishing availability target and action levels using site-specific data, and reliability targets and action levels in accordance with EPRI TR 1006756, in conjunction with setting availability and reliability targets for NFPA 805 monitored SSCs to ensure that the assumptions made in the Fire PRA and other supporting analyses will remain valid, is acceptable. The method for establishing appropriate levels of availability, reliability, and performance because there will be margin between the value assumed in the Fire PRA for a given component or system and the action level used in the ONS NFPA 805 Monitoring Program to require corrective action.

The licensee further stated that inspection and test frequencies being used to gather data to assess the availability and reliability of PMGs will be those currently contained in the ONS Selected Licensee Commitments (SLC's), which are contained in ONS UFSAR Chapter 16. The licensee stated that as more performance data is obtained, frequencies may be adjusted using the PB process described in EPRI 1006756, the Nuclear Electric Insurance Limited (NEIL) underwriting guidelines, and applicable NFPA codes such as NFPA 72, "National Fire Alarm Code."

The NRC staff finds that establishing monitoring frequencies initially using those contained in SLCs is acceptable since the licensee did not identify any changes to ONS UFSAR Chapter 16 in the LAR, as supplemented. The staff also finds the PB methods for establishing monitoring frequencies described in EPRI 1006756, NEIL underwriting guidelines, and applicable NFPA codes acceptable for this NFPA 805 RI/PB FPP.

In addition, inspection and test acceptance criteria will be developed for each PMG that will determine when a system has failed to perform its required function. Initially, these criteria will be based on the system design, manufacturer criteria, and NFPA code requirements. However, the criteria may require adjustment by the system/program engineer or multidisciplinary review team as the program becomes established and monitoring data is gathered over a period of time. However, the values of availability and reliability data will be reviewed to ensure they remain bounding for the assumptions made in the applicable supporting analyses.

The licensee stated that a software program is being developed to collect applicable reliability and availability data and will provide alerts if target values are approached. Developing instructions for the software program is an implementation item (SE Section 2.9, Table 2.9-1, Item 37). The data will be periodically evaluated by the appropriate system or program engineers. Failure to meet availability and/or reliability criteria results in the initiation of the ONS Problem Investigation Process (PIP) to establish performance goals and corrective actions to return the component or PMG into compliance with the established criteria.

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 FPP 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 finds that the licensee's process provides reasonable assurance that the licensee will implement an effective program for monitoring risk-significant fire SSCs because the multi-disciplinary review team ensures that the ONS NFPA 805 Monitoring Program does the following:

- Establishes the appropriate performance monitoring groups to be monitored.
- Utilizes an acceptable screening process for determining the SSCs to be included in the PMGs.
- Establishes availability, reliability and performance criteria for the SSCs being monitored.
- Requires corrective actions when SSC availability, reliability, and performance criteria targets are exceeded in order bring performance back within the required range.

However, since the final values for availability and reliability, as well as the performance criteria for the SSCs being monitored, have not been established for the program as of the date of this SE, completion of the ONS NFPA 805 Monitoring Program is an implementation item, as noted previously. Completion of the program will occur on the same schedule as the implementation of NFPA 805, which the NRC staff finds acceptable. Accordingly, the NRC staff concludes that, upon successful closure of this implementation item, there is reasonable assurance that the licensee will meet the requirements specified in NFPA 805, Sections 2.6.1, 2.6.2, and 2.6.3 regarding a monitoring program.

3.8. Program Documentation, Configuration Control, and Quality Assurance

This section of the SE documents the NRC staff's review in regard to the appropriate content, configuration control, and quality of the documentation used to support the transition to NFPA 805 at ONS.

3.8.1. Documentation

The NRC staff reviewed LAR Section 4.7.1 (Reference 11) to evaluate the appropriateness of the content of the ONS FPP DBD and supporting documentation.

ONS's FPP design basis is a compilation of multiple documents (such as analyses, calculations and engineering evaluations), databases, and drawings that are identified in Figure 4-8 of the LAR. ONS has documented analyses to support compliance with 10 CFR 50.48(c). The licensee stated that analyses performed to support the NFPA 805 transition were performed in accordance with the licensee's processes for ensuring assumptions are clearly defined, that results be easily understood, clearly and consistently described, and that sufficient detail be provided to allow future review of the entire analyses, as required in NFPA 805, Section 2.7.1.

The ONS FPP DBD and necessary supporting documentation described in Section 2.7.1 of NFPA 805 will be revised as part of transition. Completion of the revisions to the ONS DBD and supporting documentation is an implementation item (SE Section 2.9, Table 2.9-1, Item 45).

The licensee stated in Section 4.7.1 of the LAR that documentation associated with the ONS RI/PB FPP will be maintained for the life of the plant and organized to facilitate review for accuracy and adequacy by independent reviewers and by the NRC staff. Based on the description of the content of the ONS FPP design basis and supporting documentation, and the licensee's plans to maintain this documentation throughout the life of the plant, the NRC staff finds that the licensee's approach meets the requirements of NFPA 805, Sections 2.7.1.1, 2.7.1.2, and 2.7.1.3 regarding adequate development and maintenance of the FPP DBD, and is therefore acceptable.

3.8.2. Configuration Control

The NRC staff reviewed LAR Section 4.7.2, (Reference 11). The licensee stated that program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) was subject to the licensee's configuration control processes that meet the requirements of Section 2.7.2 of NFPA 805. This includes the appropriate procedures and configuration control processes for ensuring that changes potentially impacting the FPP are reviewed.

In a letter dated September 27, 2010 (Reference 13), the licensee stated that configuration control of RI/PB documents before and during the transition period is managed using EC procedures. These procedures were modified to include evaluation criteria for implementing design changes that specifically relate to attributes that may impact NFPA 805. More detailed reviews would be required if these evaluation criteria indicate impact. These reviews would be conducted by qualified fire protection, SSD, and PRA personnel involved in the ongoing transition work for NFPA 805.

The plant processes described above, will be in place during the NFPA 805 transition to identify changes that may impact the FPP. Additionally, a comprehensive update of the NFPA 805 analyses is planned as part of the NFPA 805 implementation period to reflect the current plant configurations. The update will include review of plant configuration changes along with changes that may have occurred from RAI responses, updates from industry groups for MSO configurations, new or revised FAQ's, and development of the PSW modification. This final review will ensure current plant configurations are appropriately reflected and evaluated in the NFPA 805 documentation prior to full implementation of NFPA 805.

The licensee further stated that the ONS FPP change evaluation procedure will be updated during the implementation period to address the NRC-approved NFPA 805 change evaluation process and that configuration control processes and procedures will be updated during the transition period to manage configuration control of the NFPA 805 design/licensing-basis documents. Revision of these procedures is an implementation item (SE Section 2.9, Table 2.9-1, Item 27).

The NRC staff reviewed the licensee's description of the process for updating and maintaining the Fire PRA to reflect plant changes made after the transition to the NFPA has been completed in SE Section 3.4.1. Based on the licensee's description of the ONS configuration control process, that ONS RI/PB FPP design basis and supporting documentation are controlled documents, and that plant changes are reviewed for impact on the FPP, the NRC staff finds that the licensee has a configuration control process that aligns with the requirements of NFPA 805, Sections 2.7.2.1 and 2.7.2.2 for revising FPP DBDs, supporting documents, and applicable FPP documentation to reflect changes made to the RI/PB FPP after the NFPA 805 FPP has been implemented.

3.8.3. Quality

The NRC staff reviewed LAR Section 4.7.3, (Reference 11) to evaluate the quality of the engineering analyses used to support the transition to the NFPA 805 and to support post-transition FPP activities at ONS. During the transition to 10 CFR 50.48(c), the licensee performed work in accordance with the quality requirements of Section 2.7.3 of NFPA 805. Quality requirements from NFPA 805 that are not currently part of the licensee's processes will be revised to include any additional requirements. Revision of these quality requirements is an implementation item (SE Section 2.9, Table 2.9-1, Item 29).

NFPA 805 requires that each analysis, calculation, or evaluation performed be independently reviewed. The licensee stated that their analyses, calculations, and evaluations performed in support of compliance with 10 CFR 50.48(c) are performed in accordance with the licensee's procedures that require independent review. The licensee also stated that future changes to the FPP will follow the guidance outlined in RG 1.174 (Reference 15) which provides for 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.

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

Verification and Validation

NFPA 805 requires that each calculation model or numerical method used be verified and validated through comparison to test results or other acceptable models. The licensee stated in the LAR that calculation models and numerical methods used in support of compliance with 10 CFR 50.48(c) are verified and validated as required by Section 2.7.3.2 of NFPA 805.

The licensee also stated that it will revise the appropriate processes and procedures to include any additional NFPA 805 quality requirements that are not currently part of the licensee's processes for use during the performance of post-transition FPP changes. Revision of the applicable post transition processes and procedures to include the NFPA 805 requirements for verification and validation is an implementation item (SE Section, 2.9 Table 2.9-1, Item 29).

Based on the licensee's description of the ONS process for verification and validation of calculation models and numerical methods, the NRC staff finds the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.2 acceptable.

Limitations of Use

NFPA 805 requires that acceptable engineering methods and numerical models only be used for applications to the extent that these methods have been subject to verification and validation; and that they only be applied within the scope, limitations, and assumptions prescribed for that method. The licensee stated that the engineering methods and numerical models used in support of the transition to NFPA 805 were used subject to the limitations of use outlined in NFPA 805, Section 2.7.3.3, and that the engineering methods and numerical models used post-transition will be subject to these same limitations of use. As an example, in LAR Section 4.5.2, the licensee stated that the fire models developed to support the NFPA 805 transition at ONS fall within their verification and validation limitations. The licensee also stated that it will revise the appropriate processes and procedures to include any additional NFPA 805 quality requirements that are not currently part of the licensee's processes for use during the performance of post-transition FPP changes. Revision of the applicable post transition processes and procedures to include the NFPA 805 requirements for limitations of use is an implementation item (SE Section 2.9, Table 2.9-1, Item 29).

Based on the licensee's description of the ONS process for placing limitations on the use of engineering methods and numerical models, the NRC staff finds the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.3 acceptable.

Qualification of Users

NFPA 805 requires that personnel performing engineering analyses and numerical methods (e.g. fire modeling) shall be competent in that field and experienced in the application of these methods as they relate to nuclear power plants, nuclear power plant fire protection, and power plant operations. The licensee has stated that, during the transition to 10 CFR 50.48(c), work will be performed in accordance with the quality requirements of Section 2.7.3 of NFPA 805. It also stated that post-transition quality requirements from NFPA 805 that are not currently part of the licensee's processes will be revised to include any additional requirements. Revision of the applicable post transition processes and procedures to include any additional NFPA 805 requirements is an implementation item (SE Section 2.9, Table 2.9-1, Item 29). Also, the licensee stated that cognizant personnel who use and apply engineering analyses and numerical methods in support of compliance with 10 CFR 50.48(c) are competent and experienced as required by Section 2.7.3.4 of NFPA 805.

For personnel performing fire modeling or fire PRA development and evaluation, the licensee has qualification requirements for individuals assigned various tasks. Position specific guides will be developed to identify and document required training and mentoring to ensure individuals, both employees of the licensee and subcontractors, are appropriately qualified per the requirements of NFPA 805, Section 2.7.3.4 to perform assigned work. Development of these position-specific guides is an implementation item (SE Section 2.9, Table 2.9-1, Item 28).

Based on the licensee's description of the ONS procedures for ensuring that the personnel who use and apply engineering analyses and numerical methods, including those who develop the ONS Fire PRA and perform fire modeling calculations, are competent and experienced, the NRC staff finds the licensee's approach for meeting the requirements of NFPA 805, Section 2.7.3.4, acceptable.

Uncertainty Analysis

NFPA 805 requires that an uncertainty analysis be performed to provide reasonable assurance that the performance criteria have been met. (Note: 10 CFR 50.48(c)(2)(iv) states that an uncertainty analysis performed in accordance with NFPA 805, Section 2.7.3.5, is not required to support calculations used in conjunction with a deterministic approach.) When using the PB methods, the licensee stated that uncertainty analyses were performed for the analyses used in support of the transition to NFPA 805, and that uncertainty analyses will be performed for post-transition analyses. Based on the licensee's description of the ONS process for performing uncertainty analyses, the NRC staff finds the licensee's approach to meeting the requirements of NFPA 805, Section 2.7.3.5 acceptable.

Conclusion for Section 3.8.3

Based on the above discussions, the NRC staff finds that the RI/PB FPP quality assurance process adequately addresses each of the requirements of NFPA 805, Section 2.7.3: conducting independent reviews, performing verification and validation (V&V), limiting the application of acceptable methods and models to within prescribed boundaries, ensuring that personnel applying acceptable methods and models are qualified for performing uncertainty analyses. The NRC staff's evaluation of the application of the NFPA 805 quality assurance requirements in the licensee's LAR is provided in the individual sections of this SE, where appropriate.

3.8.4. Fire Protection Quality Assurance Program

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

> Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.

The licensee's Fire Protection Quality Assurance Program was established in accordance with the guidelines of Appendix A to Branch Technical Position (BTP) APCSB 9.5-1, Section C, "Quality Assurance Program," (Reference 51) and associated NRC guidance. In addition, the guidance in Appendix C to NEI 04-02 (Reference 21) suggests that the LAR include a description of how the existing fire protection quality assurance (QA) program will be transitioned to the new NFPA 805 RI/PB FPP, as discussed below.

The licensee stated in the LAR that it will maintain its current fire protection QA program after transition to the NFPA 805 RI/PB FPP. In response to an NRC staff RAI, the licensee further stated that it did not foresee any substantive changes to the existing FPP QA Program (Reference 10).

Based on the licensee's statement that the ONS FPP QA Program will be maintained after transition to the NFPA 805 RI/PB FPP, the NRC staff finds that the scope of the fire protection QA program will include the fire protection systems that are required by NFPA 805, Chapter 4, and is therefore acceptable.

3.8.5. Conclusion for Section 3.8

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

4.0 FIRE PROTECTION LICENSE CONDITION

In the April 14, 2010 LAR the licensee proposed a FPP license condition regarding transition to NFPA 805, in accordance with 10 CFR 50.48(c)(3)(i). The proposed license condition adopted parts of the standard fire protection license condition promulgated in RG 1.205, Revision 1, Regulatory Position C.3.1, (Reference 14). The licensee made plant-specific changes to the sample license condition. The proposed license condition also requested self-approval of quantitative risk-informed fire protection program changes. By letter dated December 22, 2010 (Reference 59), the licensee replaced the original proposed license condition with a new license condition. The new proposed license condition did not request self-approval of quantitative risk-informed fire protection program changes. The new proposed license condition requires the licensee to request NRC review and approval in accordance with 10 CFR 50.90 prior to being allowed to self approve quantitative risk-informed fire protection program changes except for those associated with the implementation items listed in Table 2.9-1 needing a plant change evaluation provided the overall transition risk remains a decrease.

The new proposed plant-specific FPP license condition is consistent with the standard fire protection license condition promulgated in RG 1.205; Revision 1. The NRC staff has reviewed the proposed license condition and finds that it incorporates all of the relevant features of the license condition published in RG 1.205 to allow transition to NFPA 805 at ONS. The NRC staff therefore finds the licensee's proposed license condition acceptable.

Implementation of the RI/PB FPP under 10 CFR 50.48(c) will be through the application of a new FPP license condition. As part of the implementation of this license amendment, the licensee shall complete all commitments in Tables 2.8.1-1 and 2.9-1 listed in Sections 2.8 and 2.9, respectively, of this SE. The NRC staff considered the above item in the tables as part of its evaluation, and finds the commitments appropriate for transitioning to the RI/PB FPP. The NRC staff has conditioned the implementation of the proposed transition on completion of the commitments in Tables 2.8.1-1 and 2.9-1. The new license condition also establishes the date by which full compliance with 10 CFR 50.48(c) will be achieved. In addition, the license condition also dictates the licensee's required actions and restrictions until the licensee is able to fully implement the new FPP in accordance with 10 CFR 50.48(c)(3)(i).

The new fire protection license condition will replace the existing fire protection license condition in each unit's license. As a result, the NRC will be reissuing license pages 2 through 11 in each unit's license because of pagination. The only changes to the license are the changes to the fire protection license condition.

5.0 SUMMARY

Based on the above evaluation of the licensee's application, as supplemented, the NRC staff finds, the transition to a risk-informed, performance-based FPP in accordance with the requirements established by 10 CFR 50.48(c) and NFPA 805 as incorporated therein is acceptable. The NRC staff concludes that the licensee's approach, methods, and data are acceptable to establish, implement, and maintain a RI/PB FPP in accordance with 10 CFR 50.48(c).

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the South Carolina State official was notified of the proposed issuance of the amendments. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to the installation or use of facility components located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding published in the *Federal Register* on October 28, 2010 (75 FR 66395). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

8.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner; (2) 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

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52. Letter from T. Preston Gillespie, Jr., Duke Energy, to the U. S. Nuclear Regulatory Commission Document Control Desk. November 19, 2010. Subject: "Duke Energy Carolinas, LLC, Oconee Nuclear Station Units 1, 2, and 3, Docket Numbers 50-269, 50-270 and 50-287, Request for Additional Information regarding the License Amendment Request to adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition), License Amendment Request (LAR) No. 2008-01. Seneca, SC. ADAMS Accession No. ML103300227.

53. EPRI Technical Report 1006756, "Fire Protection Surveillance Optimization and Maintenance Guide for Fire Protection Systems and Features," Electric Power Research Institute, Charlotte, NC, July 2003.

54. Letter from T. Preston Gillespie, Jr., Duke Energy, to the U. S. Nuclear Regulatory Commission Document Control Desk. October 14, 2010. Subject: "Duke Energy Carolinas, LLC, Oconee Nuclear Station Units 1, 2, and 3, Docket Numbers 50-269, 50-270 and 50-287, Request for Additional Information regarding the License Amendment Request to adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition), License Amendment Request (LAR) No. 2008-01. Seneca, SC. ADAMS Accession No. ML102910093.

55. Letter from Leonard A. Wiens, U.S. NRC to J.W. Hampton, Duke Power Company, LLC, January 6, 1993, Subject: Issuance of Amendment – Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. M84909, M84910, and M84911), Docket Nos. 50-269, 50-270, and 50-287. ADAMs Accession No. ML012040034.

56. NEI 00-01, *Guidance for Post-Fire SSD Circuit Analysis*, Revision 1, Nuclear Energy Institute (NEI), Washington, DC, January 2005. ADAMS Accession No. ML050310295.

57. NRC IN 92-18, Potential for Loss of Remote Shutdown Capability During a Control Room Fire, February 28, 1992. ADAMS Accession No. ML031200481.

58. Letter from Helen N. Pastis, U.S. NRC to, H. B. Tucker, Duke Power Company, LLC, June 7, 1988, Subject: Issuance of Amendment Nos. 166, 166, and 163 to Facility Operating Licenses DPR-38, DPR-47, and DPR-55 - Oconee Nuclear Station, Units 1, 2, and 3 (TAC Nos. 66090/66091/66092. ADAMs Accession No. ML012000214.

59. Letter from T. Preston Gillespie, Jr. Duke Energy, to the U. S. Nuclear Regulatory Commission Document Control Desk. December 22, 2010. Subject: "Duke Energy Carolinas, LLC, Oconee Nuclear Station Units 1, 2, and 3, Docket Nos. 50-269, 50-270 and 50-287, Request for Additional Information regarding the License Amendment Request to adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition), License Amendment Request (LAR) No. 2008-01." Seneca, SC. ADAMS Accession No.ML103620105.

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

This attachment contains Table 3.1-1, which provides the specific FPP elements and minimum design requirements from NFPA 805, Chapter 3, as appropriately modified by 10 CFR 50.48(c). In addition, the table describes each fundamental FPP element from NFPA 805, Chapter 3, and identifies which of the methods listed below the licensee used as the means for achieving compliance with the requirement. Table 3.1-1 also provides the NRC staff's evaluation of the licensee's compliance statement for each FPP element. LAR Attachment A, "NEI 04-02 Table B-1, Transition of Fundamental FPP and Design Elements (NFPA 805, Chapter 3)," provides further details regarding the licensee's compliance strategy for specific NFPA 805, Chapter 3, requirements, including references to where compliance is documented.

As part of the assessment of its compliance with the NFPA 805, Chapter 3, elements, the licensee reviewed each section and subsection against the existing ONS FPP and provided specific compliance statements for each NFPA 805, Chapter 3, attribute that contained applicable requirements. The methods used by the licensee 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, also called the B-1 Table, as "Comply." In assessing these statements, the NRC staff reviewed the provided information to ensure that it presented a reasonable basis for concluding that the existing FPP element was adequate to meet the NFPA 805, Chapter 3, element.

2. The existing FPP element complies through the use of an explanation or clarification; noted in the B-1 Table as "Complies with Clarification." In assessing these statements, the NRC staff reviewed the provided information to ensure that it presented a reasonable basis for concluding that the FPP element, as clarified by the supplemental information, was adequate to meet the NFPA 805, Chapter 3, element.

3. The existing FPP element complies with the requirement based on prior NRC approval of an alternative to the fundamental FPP attribute and the bases for the NRC approval remain valid; noted in the B-1 Table as "Complies by Previous NRC Approval." In assessing these statements, the NRC staff reviewed the information provided to ensure that the basis was still valid for concluding that the alternative was adequate to meet the NFPA 805, Chapter 3, element.

4. The existing FPP element complies through the use of an EEEE; noted in the B-1 Table as "Complies with the Use of EEEE." In assessing these statements, the NRC staff reviewed the provided information to ensure that it presented a reasonable basis for concluding that the existing FPP element was adequate to meet the NFPA 805, Chapter 3, element.

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

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.1	**General.** This chapter contains the fundamental elements of the FPP and specifies the minimum design requirements for fire protection systems and features. These FPP elements and minimum design requirements shall not be subject to the performance-based methods permitted elsewhere in this standard. Previously approved alternatives from the fundamental FPP attributes of this chapter by the AHJ take precedence over the requirements contained herein.		Individual Elements Reviewed Below
3.2	**Fire Protection Plan.**		Individual Elements Reviewed Below
3.2.1	**3.2.1 Intent.** A site-wide fire protection plan shall be established. This plan shall document management policy and program direction and shall define the responsibilities of those individuals responsible for the plan's implementation. This section establishes the criteria for an integrated combination of components, procedures, and personnel to implement all FPP activities.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.2	**Management Policy Direction and Responsibility.** A policy document shall be prepared that defines management authority and responsibilities and establishes the general policy for the site FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.2.1	The policy document shall designate the senior management position with immediate authority and responsibility for the FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.2.2	The policy document shall designate a position responsible for the daily administration and coordination of the FPP and its implementation.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.2.2.3	The policy document shall define the fire protection interfaces with other organizations and assign responsibilities for the coordination of activities. In addition, this policy document shall identify the various plant positions having the authority for implementing the various areas of the FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.2.4	The policy document shall identify the appropriate AHJ for the various areas of the FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update the design basis specification (DBS) to include the statement the NRC is the AHJ for fire protection changes requiring approval (SE Section 2.9, Table 2.9-1, Item 1).
3.2.3	**Procedures.** Procedures shall be established for implementation of the FPP. In addition to procedures that could be required by other sections of the standard, the procedures to accomplish the following shall be established:		Individual Elements Reviewed Below
3.2.3.(1)	Inspection, testing, and maintenance for fire protection systems and features credited by the FPP	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.3.(2)	Compensatory actions implemented when fire protection systems and other systems credited by the FPP and this standard cannot perform their intended function and limits on impairment duration	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.2.3.(3)	Reviews of FPP — related performance and trends	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to complete the development of the NFPA 805 Monitoring Program including surveillance frequencies (SE Section 2.9, Table 2.9-1, Item 8).
3.2.3.(4)	Reviews of physical plant modifications and procedure changes for impact on the FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.3.(5)	Long-term maintenance and configuration of the FPP.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.2.3.(6)	Emergency response procedures for the plant industrial fire brigade.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3	**Prevention.** A fire prevention program with the goal of preventing a fire from starting shall be established, documented, and implemented as part of the FPP. The two basic components of the fire prevention program shall consist of both of the following:		Individual Elements Reviewed Below
3.3.(1)	Prevention of fires and fire spread by controls on operational activities.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.(2)	Design controls that restrict the use of combustible materials.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.1	**Fire Prevention for Operational Activities.** The fire prevention program activities shall consist of the necessary elements to address the control of ignition sources and the use of transient combustible materials during all aspects of plant operations. The fire prevention program shall focus on the human and programmatic elements necessary to prevent fires from starting or, should a fire start, to keep the fire as small as possible.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.1	**General Fire Prevention Activities.** The fire prevention activities shall include but not be limited to the following program elements:	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.1.(1)	Training on fire safety information for all employees and contractors including, as a minimum, familiarization with plant fire prevention procedures, fire reporting, and plant emergency alarms.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.1.(2)	Documented plant inspections including provisions for corrective actions for conditions where unanalyzed fire hazards are identified.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.1.(3)	Administrative controls addressing the review of plant modifications and maintenance to ensure that both fire hazards and the impact on plant fire protection systems and features are minimized.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.2	**Control of Combustible Materials.** Procedures for the control of general housekeeping practices and the control of transient combustibles shall be developed and implemented. These procedures shall include but not be limited to the following program elements:	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.1.2.(1)	Wood used within the power block shall be listed pressure- impregnated or coated with a listed fire-retardant application. *Exception: Cribbing timbers 6 in. by 6 in. (15.2 cm by 15.2 cm) or larger shall not be required to be fire-retardant treated.*	Submit For NRC Approval	The NRC staff finds the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.1.
3.3.1.2.(2)	Plastic sheeting materials used in the power block shall be fire-retardant types that have passed NFPA 701, *Standard Methods of Fire Tests for Flame Propagation of Textiles and Films*, large-scale tests, or equivalent.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to complete the development of procedural controls for plastic sheeting (SE Section 2.9, Table 2.9-1, Item 2).
3.3.1.2.(3)	Waste, debris, scrap, packing materials, or other combustibles shall be removed from an area immediately following the completion of work or at the end of the shift, whichever comes first.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.2.(4)	Combustible storage or staging areas shall be designated, and limits shall be established on the types and quantities of stored materials.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.2.(5)	Controls on use and storage of flammable and combustible liquids shall be in accordance with NFPA 30, *Flammable and Combustible Liquids Code*, or other applicable NFPA Standards.	Complies with Clarification	The NRC staff finds that the licensee's explanation of their method of compliance with these requirements acceptable based on the information provided in the LAR B-1 Table element.

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.1.2.(6)	Controls on use and storage of flammable gases shall be in accordance with applicable NFPA standards.	Complies with Clarification	The NRC staff finds that the licensee's explanation of their method of compliance with these requirements acceptable based on the information provided in the LAR B-1 Table element.
3.3.1.3	**Control of Ignition Sources.**		Individual Elements Reviewed Below
3.3.1.3.1	A hot work safety procedure shall be developed, implemented, and periodically updated as necessary in accordance with NFPA 51B, *Standard for Fire Prevention During Welding, Cutting, and Other Hot Work,* and NFPA 241, *Standard for Safeguarding Construction, Alteration, and Demolition Operations.*	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.3.2	Smoking and other possible sources of ignition shall be restricted to properly designated and supervised safe areas of the plant.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.1.3.3	Open flames or combustion-generated smoke shall not be permitted for leak or airflow testing.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update appropriate station procedure(s) for leak or air flow testing to preclude the use of open flames or combustion generated smoke (SE Section 2.9, Table 2.9-1, Item 3).

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.1.3.4	Plant administrative procedure shall control the use of portable electrical heaters in the plant. Portable fuel-fired heaters shall not be permitted in plant areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from a fire.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update the directives to prohibit the use of fuel-fired heaters in plant areas with equipment important to nuclear safety or in areas where there is the potential for radiological release due to fire (SE Section 2.9, Table 2.9-1, Item 35).
3.3.2	**Structural.** Walls, floors, and components required to maintain structural integrity shall be of noncombustible construction, as defined in NFPA 220, *Standard on Types of Building Construction*.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.3	**Interior Finishes.** Interior wall or ceiling finish classification shall be in accordance with NFPA *101®*, *Life Safety Code®*, requirements for Class A materials. Interior floor finishes shall be in accordance with NFPA *101* requirements for Class I interior floor finishes.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update the coatings program directives to include Class A and Class I specifications for interior finishes (SE Section 2.9, Table 2.9-1, Item 4).
3.3.4	**Insulation Materials.** Thermal insulation materials, radiation shielding materials, ventilation duct materials, and soundproofing materials shall be noncombustible or limited combustible.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.5	**Electrical.**		Individual Elements Reviewed Below

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.5.1	Wiring above suspended ceiling shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers.	Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.3.
3.3.5.2	Only metal tray and metal conduits shall be used for electrical raceways. Thin wall metallic tubing shall not be used for power, instrumentation, or control cables. Flexible metallic conduits shall only be used in short lengths to connect components.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update documentation for the use of electrical raceway construction limits (SE Section 2.9, Table 2.9-1, Item 5).
3.3.5.3	Electric cable construction shall comply with a flame propagation test as acceptable to the AHJ. [Note: This entry modified per 10 CFR 50.48(c)(2)(v)]	Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.4.
3.3.6	**Roofs.** Metal roof deck construction shall be designed and installed so the roofing system will not sustain a self-propagating fire on the underside of the deck when the deck is heated by a fire inside the building. Roof coverings shall be Class A as determined by tests described in NFPA 256, *Standard Methods of Fire Tests of Roof Coverings*.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.7	**Bulk Flammable Gas Storage.** Bulk compressed or cryogenic flammable gas storage shall not be permitted inside structures housing systems, equipment, or components important to nuclear safety.	Comply	The NRC staff finds the licensee's statement of compliance acceptable

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.7.1	Storage of flammable gas shall be located outdoors, or in separate detached buildings, so that a fire or explosion will not adversely impact systems, equipment, or components important to nuclear safety. NFPA 50A, *Standard for Gaseous Hydrogen Systems at Consumer Sites*, shall be followed for hydrogen storage.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Comply	The NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.2.
3.3.7.2	Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.7.3	Flammable gas storage cylinders not required for normal operation shall be isolated from the system.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.8	**Bulk Storage of Flammable and Combustible Liquids.** Bulk storage of flammable and combustible liquids shall not be permitted inside structures containing systems, equipment, or components important to nuclear safety. As a minimum, storage and use shall comply with NFPA 30, *Flammable and Combustible Liquids Code*.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.9	**Transformers.** Where provided, transformer oil collection basins and drain paths shall be periodically inspected to ensure that they are free of debris and capable of performing their design function.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update transformer deluge system flow test procedures (SE Section 2.9, Table 2.9-1, Item 6).
3.3.10	**Hot Pipes and Surfaces.** Combustible liquids, including high flashpoint lubricating oils, shall be kept from coming in contact with hot pipes and surfaces, including insulated pipes and surfaces. Administrative controls shall require the prompt cleanup of oil on insulation.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.11	**Electrical Equipment.** Adequate clearance, free of combustible material, shall be maintained around energized electrical equipment.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.12	**Reactor Coolant Pumps.** For facilities with non-inerted containments, RCPs with an external lubrication system shall be provided with an oil collection system. The oil collection system shall be designed and installed such that leakage from the oil system is safely contained for off normal conditions such as accident conditions or earthquakes. All of the following shall apply.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.12.(1)	The oil collection system for each RCP shall be capable of collecting lubricating oil from all potential pressurized and nonpressurized leakage sites in each RCP oil system.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.
3.3.12.(2)	Leakage shall be collected and drained to a vented closed container that can hold the inventory of the RCP lubricating oil system.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.3.12.(3)	A flame arrestor is required in the vent if the flash point characteristics of the oil present the hazard of a fire flashback.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.12.(4)	Leakage points on an RCP motor to be protected shall include but not be limited to the lift pump and piping, overflow lines, oil cooler, oil fill and drain lines and plugs, flanged connections on oil lines, and the oil reservoirs, where such features exist on the RCPs.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.3.12.(5)	The collection basin drain line to the collection tank shall be large enough to accommodate the largest potential oil leak such that oil leakage does not overflow the basin.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4	**Industrial Fire Brigade.**		Individual Elements Reviewed Below
3.4.1	**On-Site Fire-Fighting Capability.** All of the following requirements shall apply.		Individual Elements Reviewed Below
3.4.1.(a)	A fully staffed, trained, and equipped fire-fighting force shall be available at all times to control and extinguish all fires on site. This force shall have a minimum complement of five persons on duty and shall conform with the following NFPA standards as applicable:	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. Note that the licensee states that they will comply with NFPA 600, 2005 Edition.
3.4.1.(a).(1)	NFPA 600, *Standard on Industrial Fire Brigades* (interior structural fire fighting)	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. Note that the licensee states that they will comply with NFPA 600, 2005 Edition.
3.4.1.(a).(2)	NFPA 1500, *Standard on Fire Department Occupational Safety and Health Program*		The licensee stated it complies with NFPA 600, 2005 Edition: see subsection 3.4.1.(a).(1) above.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.4.1.(a).(3)	NFPA 1582, *Standard on Medical Requirements for Fire Fighters and Information for Fire Department Physicians*		The licensee stated it complies with NFPA 600, 2005 Edition: see subsection 3.4.1.(a).(1) above.
3.4.1.(b)	Industrial fire brigade members shall have no other assigned normal plant duties that would prevent immediate response to a fire or other emergency as required.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.1.(c)	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. *Exception to (c): Sufficient training and knowledge shall be permitted to be provided by an operations advisor dedicated to industrial fire brigade support.*	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.1.(d)	The industrial fire brigade shall be notified immediately upon verification of a fire.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.1.(e)	Each industrial fire brigade member shall pass an annual physical examination to determine that he or she can perform the strenuous activity required during manual firefighting operations. The physical examination shall determine the ability of each member to use respiratory protection equipment.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.2	**Pre-Fire Plans.** Current and detailed pre-fire plans shall be available to the industrial fire brigade for all areas in which a fire could jeopardize the ability to meet the performance criteria described in Section 1.5.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.4.2.1	The plans shall detail the fire area configuration and fire hazards to be encountered in the fire area, along with any nuclear safety components and fire protection systems and features that are present.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update the pre-fire plans (SE Section 2.9, Table 2.9-1, Item 9).
3.4.2.2	Pre-fire plans shall be reviewed and updated as necessary.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.2.3	Pre-fire plans shall be available in the control room and made available to the plant industrial fire brigade.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update the pre-fire plans (SE Section 2.9, Table 2.9-1, Item 10).
3.4.2.4	Pre-fire plans shall address coordination with other plant groups during fire emergencies.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3	**Training and Drills.** Industrial fire brigade members and other plant personnel who would respond to a fire in conjunction with the brigade shall be provided with training commensurate with their emergency responsibilities.		Individual elements reviewed below
3.4.3.(a)	*Plant Industrial Fire Brigade Training.* All of the following requirements shall apply.		Individual elements reviewed below
3.4.3.(a).(1)	Plant industrial fire brigade members shall receive training consistent with the requirements contained in NFPA 600, *Standard on Industrial Fire Brigades*, or NFPA 1500, *Standard on Fire Department Occupational Safety and Health Program*, as appropriate.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. Note that the licensee states that they comply with NFPA 600, 2005 Edition.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.4.3.(a).(2)	Industrial fire brigade members shall be given quarterly training and practice in fire fighting, including radioactivity and health physics considerations, to ensure that each member is thoroughly familiar with the steps to be taken in the event of a fire.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(a).(3)	A written program shall detail the industrial fire brigade training program.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(a).(4)	Written records that include but are not limited to initial industrial fire brigade classroom and hands-on training, refresher training, special training schools attended, drill attendance records, and leadership training for industrial fire brigades shall be maintained for each industrial fire brigade member.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(b)	*Training for Non-Industrial Fire Brigade Personnel.* Plant personnel who respond with the industrial fire brigade shall be trained as to their responsibilities, potential hazards to be encountered, and interfacing with the industrial fire brigade.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(c)	*Drills.* All of the following requirements shall apply.		Individual elements reviewed below
3.4.3.(c).(1)	Drills shall be conducted quarterly for each shift to test the response capability of the industrial fire brigade.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(c).(2)	Industrial fire brigade drills shall be developed to test and challenge industrial fire brigade responses, including brigade performance as a team, proper use of equipment, effective use of pre-fire plans, and coordination with other groups. These drills shall evaluate the industrial fire brigade's abilities to react, respond, and demonstrate proper fire-fighting techniques to control and extinguish the fire and smoke conditions being simulated by the drill scenario.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.4.3.(c).(3)	Industrial fire brigade drills shall be conducted in various plant areas, especially in those areas identified to be essential to plant operation and to contain significant fire hazards.	Comply	The NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to update fire brigade training documentation (SE Section 2.9, Table 2.9-1, Item 7).
3.4.3.(c).(4)	Drill records shall be maintained detailing the drill scenario, industrial fire brigade member response, and ability of the industrial fire brigade to perform as a team.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.3.(c).(5)	A critique shall be held and documented after each drill.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.4	**Fire-Fighting Equipment.** Protective clothing, respiratory protective equipment, radiation monitoring equipment, personal dosimeters, and fire suppression equipment such as hoses, nozzles, fire extinguishers, and other needed equipment shall be provided for the industrial fire brigade. This equipment shall conform with the applicable NFPA standards.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.5	**Off-Site Fire Department Interface.**		Individual elements reviewed below.
3.4.5.1	**Mutual Aid Agreement.** Off-site fire authorities shall be offered a plan for their interface during fires and related emergencies on site.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.5.2	**Site-Specific Training.** Fire fighters from the off-site fire authorities who are expected to respond to a fire at the plant shall be offered site-specific training and shall be invited to participate in a drill at least annually.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.4.5.3	**Security and Radiation Protection.** Plant security and radiation protection plans shall address off-site fire authority response.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.4.6	**Communications.** An effective emergency communications capability shall be provided for the industrial fire brigade.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5	**Water Supply.**		Individual elements reviewed below
3.5.1	A fire protection water supply of adequate reliability, quantity, and duration shall be provided by one of the two following methods. (a) Provide a fire protection water supply of not less than two separate 300,000-gal (1,135,500-L) supplies. (b) Calculate the fire flow rate for 2 hours. This fire flow rate shall be based on 500 gpm (1892.5 L/min) for manual hose streams plus the largest design demand of any sprinkler or fixed water spray system(s) in the power block as determined in accordance with NFPA 13, *Standard for the Installation of Sprinkler Systems,* or NFPA 15, *Standard for Water Spray Fixed Systems for Fire Protection.* The fire water supply shall be capable of delivering this design demand with the hydraulically least demanding portion of fire main loop out of service.	Comply	The NRC staff finds that the licensee's explanation of their PB method of compliance using the HPSW system with these requirements acceptable to meet the intent of this subsection.

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Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.5.2	The tanks shall be interconnected such that fire pumps can take suction from either or both. A failure in one tank or its piping shall not allow both tanks to drain. The tanks shall be designed in accordance with NFPA 22, *Standard for Water Tanks for Private Fire Protection.* *Exception No. 1: Water storage tanks shall not be required when fire pumps are able to take suction from a large body of water (such as a lake), provided each fire pump has its own suction and both suctions and pumps are adequately separated.* *Exception No. 2: Cooling tower basins shall be an acceptable water source for fire pumps when the volume is sufficient for both purposes and water quality is consistent with the demands of the fire service.*	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.3	Fire pumps, designed and installed in accordance with NFPA 20, *Standard for the Installation of Stationary Pumps for Fire Protection*, shall be provided to ensure that 100 percent of the required flow rate and pressure are available assuming failure of the largest pump or pump power source.	Complies by Previous NRC Approval	The NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP. The NRC staff has accepted the use of the HPSW pumps as fire pumps in NRC SE, Section 4.3.1.2, dated August 11, 1978. Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
		Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.6, 3.1.3.7, 3.1.3.8, and 3.1.3.11.
3.5.4	At least one diesel engine-driven fire pump or two more seismic Category I Class IE electric motor-driven fire pumps connected to redundant Class IE emergency power buses capable of providing 100 percent of the required flow rate and pressure shall be provided.	Complies by Previous NRC Approval	For use of the HPSW pumps as fire pumps to protect ONS Power Block: the NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP in NRC SE Section 4.3.1.2, dated August 11, 1978 (Reference 26). Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
		Submit for NRC Approval	For use of a single fire pump at Keowee Hydro Station, the NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.11.
3.5.5	Each pump and its driver and controls shall be separated from the remaining fire pumps and from the rest of the plant by rated fire barriers.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.6	Fire pumps shall be provided with automatic start and manual stop only.	Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.8.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.5.7	Individual fire pump connections to the yard fire main loop shall be provided and separated with sectionalizing valves between connections.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.9.
3.5.8	A method of automatic pressure maintenance of the fire protection water system shall be provided independent of the fire pumps.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.9	Means shall be provided to immediately notify the control room, or other suitable constantly attended location, of operation of fire pumps.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.10	An underground yard fire main loop, designed and installed in accordance with NFPA 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances*, shall be installed to furnish anticipated water requirements.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.9.
3.5.11	Means shall be provided to isolate portions of the yard fire main loop for maintenance or repair without simultaneously shutting off the supply to both fixed fire suppression systems and fire hose stations provided for manual backup. Sprinkler systems and manual hose station standpipes shall be connected to the plant fire protection water main so that a single active failure or a crack to the water supply piping to these systems can be isolated so as not to impair both the primary and backup fire suppression systems.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.5.12	Threads compatible with those used by local fire departments shall be provided on all hydrants, hose couplings, and standpipe risers. *Exception: Fire departments shall be permitted to be provided with adapters that allow interconnection between plant equipment and the fire department equipment if adequate training and procedures are provided.*	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.13	Headers fed from each end shall be permitted inside buildings to supply both sprinkler and standpipe systems, provided steel piping and fittings meeting the requirements of ANSI B31.1, *Code for Power Piping*, are used for the headers (up to and including the first valve) supplying the sprinkler systems where such headers are part of the seismically analyzed hose standpipe system. Where provided, such headers shall be considered an extension of the yard main system. Each sprinkler and standpipe system shall be equipped with an outside screw and yoke (OS&Y) gate valve or other approved shutoff valve.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.5.14	All fire protection water supply and fire suppression system control valves shall be under a periodic inspection program and shall be supervised by one of the following methods: (a) Electrical supervision with audible and visual signals in the MCR or other suitable constantly attended location. (b) Locking valves in their normal position. Keys shall be made available only to authorized personnel. (c) Sealing valves in their normal positions. This option shall be utilized only where valves are located within fenced areas or under the direct control of the owner/operator.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.5.15	Hydrants shall be installed approximately every 250 ft (76 m) apart on the yard main system. A hose house equipped with hose and combination nozzle and other auxiliary equipment specified in NFPA 24, *Standard for the Installation of Private Fire Service Mains and Their Appurtenances,* shall be provided at intervals of not more than 1000 ft (305 m) along the yard main system. *Exception: Mobile means of providing hose and associated equipment, such as hose carts or trucks, shall be permitted in lieu of hose houses. Where provided, such mobile equipment shall be equivalent to the equipment supplied by three hose houses.*	Complies by Previous NRC Approval	The NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP. The NRC staff has accepted the use of the installed hydrants in NRC SE, Section 4.3.1.3, dated August 11, 1978 (Reference 26). Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
		Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.9.
3.5.16	The fire protection water supply system shall be dedicated for fire protection use only. *Exception No. 1: Fire protection water supply systems shall be permitted to be used to provide backup to nuclear safety systems, provided the fire protection water supply systems are designed and maintained to deliver the combined fire and nuclear safety flow demands for the duration specified by the applicable analysis.* *Exception No. 2: Fire protection water storage can be provided by plant systems serving other functions, provided the storage has a dedicated capacity capable of providing the maximum fire protection demand for the specified duration as determined in this section.*	Complies by Previous NRC Approval	The NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP. The NRC staff has accepted the use of the water supply in NRC SE, Sections 4.3.1.2 and 4.3.1.4, dated August 11, 1978 (Reference 26). Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
		Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.7 and 3.1.3.10.
3.6	**3.6 Standpipe and Hose Stations.**		Individual elements reviewed below

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.6.1	For all power block buildings, Class III standpipe and hose systems shall be installed in accordance with NFPA 14, *Standard for the Installation of Standpipe, Private Hydrant, and Hose Systems*.	Complies Via Previous NRC Approval	The NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP. The NRC staff approved the design of the standpipe and fire hose systems including modification required to the RB hose stations in the NRC SE, Section 4.3.1.4, dated August 11, 1978 (Reference 26) and the NRC SER dated June 7, 1988 (Reference 57). Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
		Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.7.
3.6.2	A capability shall be provided to ensure an adequate water flow rate and nozzle pressure for all hose stations. This capability includes the provision of hose station pressure reducers where necessary for the safety of plant industrial fire brigade members and off-site fire department personnel.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
		Submit For NRC Approval	The NRC staff finds that the licensee's proposed PB method to demonstrate compliance is acceptable as described in SE Section 3.1.3.7.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.6.3	The proper type of hose nozzle to be supplied to each power block area shall be based on the area fire hazards. The usual combination spray/straight stream nozzle shall not be used in areas where the straight stream can cause unacceptable damage or present an electrical hazard to fire-fighting personnel. Listed electrically safe fixed fog nozzles shall be provided at locations where high-voltage shock hazards exist. All hose nozzles shall have shutoff capability and be able to control water flow from full open to full closed.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.6.4	Provisions shall be made to supply water at least to standpipes and hose stations for manual fire suppression in all areas containing systems and components needed to perform the nuclear safety functions in the event of a SSD earthquake (SSE). [Note: This entry modified per 10 CFR 50.48(c)(2)(vi)]	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.6.5	Where the seismic required hose stations are cross-connected to essential seismic non-fire protection water supply systems, the fire flow shall not degrade the essential water system requirement.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.7	**Fire Extinguishers.** Where provided, fire extinguishers of the appropriate number, size, and type shall be provided in accordance with NFPA 10, *Standard for Portable Fire Extinguishers*. Extinguishers shall be permitted to be positioned outside of fire areas due to radiological conditions.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8	**Fire Alarm and Detection Systems.**		Individual elements reviewed below.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.8.1	**Fire Alarm.** Alarm initiating devices shall be installed in accordance with NFPA 72, *National Fire Alarm Code®*. Alarm annunciation shall allow the proprietary alarm system to transmit fire-related alarms, supervisory signals, and trouble signals to the control room or other constantly attended location from which required notifications and response can be initiated. Personnel assigned to the proprietary alarm station shall be permitted to have other duties. The following fire-related signals shall be transmitted:	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.(1)	Actuation of any fire detection device	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.(2)	Actuation of any fixed fire suppression system	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.(3)	Actuation of any manual fire alarm station	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.8.1.(4)	Starting of any fire pump	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.(5)	Actuation of any fire protection supervisory device	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.(6)	Indication of alarm system trouble condition	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.1	Means shall be provided to allow a person observing a fire at any location in the plant to quickly and reliably communicate to the control room or other suitable constantly attended location.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.2	Means shall be provided to promptly notify the following of any fire emergency in such a way as to allow them to determine an appropriate course of action:		Individual elements reviewed below.
3.8.1.2.(1)	General site population in all occupied areas.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.8.1.2.(2)	Members of the industrial fire brigade and other groups supporting fire emergency response.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.8.1.2.(3)	Off-site fire emergency response agencies. Two independent means shall be available (e.g., telephone and radio) for notification of off-site emergency services.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.8.2	**Detection.** If automatic fire detection is required to meet the performance or deterministic requirements of Chapter 4, then these devices shall be installed in accordance with NFPA 72, *National Fire Alarm Code*, and its applicable appendixes.	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to complete the development of compliance calculations for fire detection (SE Section 2.9, Table 2.9-1, Item 11). MODIFICATION - The licensee evaluated the fire detection coverage and additional detection is required (SE Section 2.8, Table 2.8.1-1, Item 5).
3.9	**Automatic and Manual Water-Based Fire Suppression Systems.**		Individual elements reviewed below.
3.9.1	If an automatic or manual water-based fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be installed in accordance with the appropriate NFPA standards including the following:		Individual elements reviewed below.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.9.1.(1)	NFPA 13, *Standard for the Installation of Sprinkler Systems*	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to complete the hydraulic calculations for all required automatic or manual water-based suppression systems (SE Section 2.9, Table 2.9-1, Item 12).
3.9.1.(2)	NFPA 15, *Standard for Water Spray Fixed Systems for Fire Protection*	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. IMPLEMENTATION ITEM – The licensee identified an action to complete the hydraulic calculations for all required automatic or manual water-based suppression systems (SE Section 2.9, Table 2.9-1, Item 12).
3.9.1.(3)	NFPA 750, *Standard on Water Mist Fire Protection Systems*		The licensee has not credited any of these systems in LAR Table 4-4.
3.9.1.(4)	NFPA 16, *Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems*		The licensee has not credited any of these systems in LAR Table 4-4.
3.9.2	Each system shall be equipped with a water flow alarm.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.9.3	All alarms from fire suppression systems shall annunciate in the control room or other suitable constantly attended location.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.9.4	Diesel-driven fire pumps shall be protected by automatic sprinklers.		The licensee has stated that there are no diesel-driven fire pumps installed at ONS.
3.9.5	Each system shall be equipped with an OS&Y gate valve or other approved shutoff valve.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.9.6	All valves controlling water-based fire suppression systems required to meet the performance or deterministic requirements of Chapter 4 shall be supervised as described in 3.5.14.	Comply	The NRC staff finds the licensee's statement of compliance acceptable.
3.10	**Gaseous Fire Suppression Systems**		Individual elements reviewed below
3.10.1	If an automatic total flooding and local application gaseous fire suppression system is required to meet the performance or deterministic requirements of Chapter 4, then the system shall be designed and installed in accordance with the following applicable NFPA codes:		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.1.(1)	NFPA 12, *Standard on Carbon Dioxide Extinguishing Systems*		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.1.(2)	NFPA 12A, *Standard on Halon 1301 Fire Extinguishing Systems*		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.1.(3)	NFPA 2001, *Standard on Clean Agent Fire Extinguishing Systems*		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.2	Operation of gaseous fire suppression systems shall annunciate an alarm in the control room or other constantly attended location identified.		The licensee has not credited any of these systems in LAR Table 4-4.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.10.3	Ventilation system design shall take into account prevention from over-pressurization during agent injection, adequate sealing to prevent loss of agent, and confinement of radioactive contaminants.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.4	In any area required to be protected by both primary and backup gaseous fire suppression systems, a single active failure or a crack in any pipe in the fire suppression system shall not impair both the primary and backup fire suppression capability.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.5	Provisions for locally disarming automatic gaseous suppression systems shall be secured and under strict administrative control.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.6	Total flooding carbon dioxide systems shall not be used in normally occupied areas.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.7	Automatic total flooding carbon dioxide systems shall be equipped with an audible pre-discharge alarm and discharge delay sufficient to permit egress of personnel. The carbon dioxide system shall be provided with an odorizer.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.8	Positive mechanical means shall be provided to lock out total flooding carbon dioxide systems during work in the protected space.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.9	The possibility of secondary thermal shock (cooling) damage shall be considered during the design of any gaseous fire suppression system, but particularly with carbon dioxide.		The licensee has not credited any of these systems in LAR Table 4-4.
3.10.10	Particular attention shall be given to corrosive characteristics of agent decomposition products on safety systems.		The licensee has not credited any of these systems in LAR Table 4-4.

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Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.11	**Passive Fire Protection Features.** This section shall be used to determine the design and installation requirements for passive protection features. Passive fire protection features include wall, ceiling, and floor assemblies, fire doors, fire dampers, and through fire barrier penetration seals. Passive fire protection features also include electrical raceway fire barrier systems (ERFBS) that are provided to protect cables and electrical components and equipment from the effects of fire.		Individual elements reviewed below
3.11.1	**Building Separation.** Each major building within the power block shall be separated from the others by barriers having a designated fire resistance rating of 3 hours or by open space of at least 50 ft (15.2 m) or space that meets the requirements of NFPA 80A, *Recommended Practice for Protection of Buildings from Exterior Fire Exposures.* Exception: *Where a performance-based analysis determines the adequacy of building separation, the requirements of 3.11.1 shall not apply.*	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.
3.11.2	**Fire Barriers.** Fire barriers required by Chapter 4 shall include a specific fire-resistance rating. Fire barriers shall be designed and installed to meet the specific fire resistance rating using assemblies qualified by fire tests. The qualification fire tests shall be in accordance with NFPA 251, *Standard Methods of Tests of Fire Endurance of Building Construction and Materials*, or ASTM E 119, *Standard Test Methods for Fire Tests of Building Construction and Materials.*	Complies with Use of EEEE	The licensee has stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable.

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.11.3	**Fire Barrier Penetrations.** Penetrations in fire barriers shall be provided with listed fire-rated door assemblies or listed rated fire dampers having a fire resistance rating consistent with the designated fire resistance rating of the barrier as determined by the performance requirements established by Chapter 4. *(See 3.11.3.4 for penetration seals for through penetration fire stops.)* Passive fire protection devices such as doors and dampers shall conform with the following NFPA standards, as applicable: (1) NFPA 80, *Standard for Fire Doors and Fire Windows* (2) NFPA 90A, *Standard for the Installation of Air-Conditioning and Ventilating Systems* (3) NFPA *101*, *Life Safety Code* *Exception: Where fire area boundaries are not wall-to-wall, floor-to-ceiling boundaries with all penetrations sealed to the fire rating required of the boundaries, a performance-based analysis shall be required to assess the adequacy of fire barrier forming the fire boundary to determine if the barrier will withstand the fire effects of the hazards in the area. Openings in fire barriers shall be permitted to be protected by other means as acceptable to the AHJ.*	Complies with Clarification	The NRC staff finds that the licensee's explanation of their method of compliance with these requirements acceptable based on the information provided in the associated LAR B-1 Table element.
		Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. MODIFICATION – The licensee has evaluated the fire barriers separating the power block buildings using Code Conformance Reviews. The licensee identified some modification of fire doors are required (SE Section 2.8, Table 2.8.1-1, Item 3).
3.11.4	**Through Penetration Fire Stops.** Through penetration fire stops for penetrations such as pipes, conduits, bus	Comply	The NRC staff finds the licensee's statement of compliance acceptable.

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
	ducts, cables, wires, pneumatic tubes and ducts, and similar building service equipment that pass through fire barriers shall be protected as follows: (a) The annular space between the penetrating item and the through opening in the fire barrier shall be filled with a qualified fire-resistive penetration seal assembly capable of maintaining the fire resistance of the fire barrier. The assembly shall be qualified by tests in accordance with a fire test protocol acceptable to the AHJ or be protected by a listed fire-rated device for the specified fire-resistive period.	Complies by Previous NRC Approval	The NRC staff has previously approved an alternative to this requirement that the licensee is carrying forward into the RI/PB FPP. The NRC staff has approved specific deviations regarding the fire penetration stops in NRC SE dated August 21, 1989, Reference 27.. Based on the licensee's justification of continued validity, the NRC staff finds the licensee's statement of compliance acceptable.
	(b) Conduits shall be provided with an internal fire seal that has an equivalent fire-resistive rating to that of the fire barrier through opening fire stop and shall be permitted to be installed on either side of the barrier in a location that is as close to the barrier as possible. *Exception: Openings inside conduit 4 in. (10.2 cm) or less in diameter shall be sealed at the fire barrier with a fire-rated internal seal unless the conduit extends greater than 5 ft (1.5 m) on each side of the fire barrier. In this case the conduit opening shall be provided with noncombustible material to prevent the passage of smoke and hot gases. The fill depth of the material packed to a depth of 2 in. (5.1 cm) shall constitute an acceptable smoke and hot gas seal in this application.*	Complies with Use of EEEE	The licensee stated that compliance has been demonstrated through the use of an EEEE. Based on the licensee's justification of continued validity and evaluation quality, the NRC staff finds the licensee's statement of compliance acceptable. MODIFICATION - Modification of penetrations seals are required (SE Section 2.8, Table 2.8.1-1, Items 2 and 3).

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
3.11.5	**Electrical Raceway Fire Barrier Systems (ERFBS).** ERFBS required by Chapter 4 shall be capable of resisting the fire effects of the hazards in the area. ERFBS shall be tested in accordance with and shall meet the acceptance criteria of NRC Generic Letter 86-10, Supplement 1, "Fire Endurance Test Acceptance Criteria for Fire Barrier Systems Used to Separate SSD Trains Within the Same Fire Area." The ERFBS needs to adequately address the design requirements and limitations of supports and intervening items and their impact on the fire barrier system rating. The fire barrier system's ability to maintain the required nuclear safety circuits free of fire damage for a specific thermal exposure, barrier design, raceway size and type, cable size, fill, and type shall be demonstrated.		The licensee has stated that there are no ERFBS credited at ONS.

Attachment A, NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Table 3.1-1 NFPA 805, Chapter 3, Fundamental Elements Compliance Matrix

Element	NFPA 805 Requirement	ONS Compliance Statement	NRC Staff's Evaluation
	Exception No. 1: When the temperatures inside the fire barrier system exceed the maximum temperature allowed by the acceptance criteria of Generic Letter 86-10, "Fire Endurance Acceptance Test Criteria for Fire Barrier Systems Used to Separate Redundant SSD Training Within the Same Fire Area," Supplement 1, functionality of the cable at these elevated temperatures shall be demonstrated. Qualification demonstration of these cables shall be performed in accordance with the electrical testing requirements of Generic Letter 86-10, Supplement 1, Attachment 1, "Attachment Methods for Demonstrating Functionality of Cables Protected by Raceway Fire Barrier Systems During and After Fire Endurance Test Exposure." *Exception No. 2: ERFBS systems employed prior to the issuance of Generic Letter 86-10, Supplement 1, are acceptable providing that the system successfully met the limiting end point temperature requirements as specified by the AHJ at the time of acceptance.*		

Attachment B, Nuclear Safety Capability Assessment Method Review

Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," Revision 1, endorses, with certain exceptions, NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based FPP Under 10 CFR 50.48(c)," Revision 2, and Chapter 3 of NEI 00-01, Revision 2, "Guidance for Post-fire SSD Circuit Analysis", and promulgates the method outlined in NEI 04-02 for conducting an NSCA. This NRC-endorsed method documents in a table format (i.e., NEI 04-02 Table B-2, "NFPA 805 Chapter 2 - Nuclear Safety Transition - Methodology Review") the licensee's comparison of its post-fire SSD analyses to the guidance in NEI 00-01, Chapter 3, which has been determined to address the related requirements of NFPA 805, Section 2.4.2, "Nuclear Safety Capability Assessment."

This attachment contains Table 3.2-1, which identifies each applicable NEI 00-01 guidance section, documents whether the licensee stated that it met the guidance, or provided justification for meeting the intent of that guidance or not meeting the guidance, and presents the staff's evaluation of each NEI 00-01, Chapter 3 attribute for which the licensee stated its process/justification for meeting the intent of the guidance or not meeting the guidance.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.0	Deterministic Methodology	Aligns with intent - ONS states that it conforms to NEI 00-01, Revision 1, with certain exceptions, as noted in the individual paragraph or section comparisons below.	By letter dated July 30, 2010 (Reference 43), the staff requested the licensee to clarify differences in Alignment Bases statements of the April 2010 LAR (Reference 11) and the October 2008 LAR (Reference 2) and, where necessary, provide additional information to readily conclude that each sub-criterion of the NEI 00-01 section has been satisfied. In its response, dated September 13, 2010 (Reference 12), the licensee identifies 55 sections of NEI 00-01 where the Alignment Basis was found to differ between submittals. Of these, changes made to 13 sections, it was determined to have resulted in a lack of detail needed to confirm alignment with the NEI guidance. The licensee states that for each of these 13 sections, additional details will be added back to the alignment basis. The specific Sections of NEI 00-01 requiring revision and proposed changes are identified in Reference 12 and are identified as an implementation item (SE Section 2.9, Table 2.9-1, Item 38). As documented in the detailed discussions for each of the table attributes below, the NRC staff finds that the licensee has, in most instances, achieved either alignment with the NEI 00-01 guidance document, or alignment with the intent of the guidance. For those attributes that do not align or align with intent, the licensee has described the process and actions being taken to bring the attribute into alignment. The NRC staff finds this approach acceptable.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1 [A] Intro	SSD Systems and Path Development	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1 [B] Goals	SSD Systems and Path Development	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1 [C] Spurious Operation	SSD Systems and Path Development	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1	Criteria/Assumption	Detailed alignment discussed below.	Individual elements reviewed below.
3.1.1.1	GE BWR Paths	N/A	N/A
3.1.1.2	SRVs / LP Systems	N/A	N/A
3.1.1.3	PWR Pressurizer Heaters	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1.4	Alternative Shutdown Capability	Aligns with intent. The transfer of control to the SSF isolates required systems and equipment from the effects of a fire for the fire areas of concern. The intent of the guidance is that dedicated cables and equipment is independent of the fire area of concern. Following transfer of control to the SSF, the dedicated equipment credited for an SSF shutdown meets the intent of the guidance.	The NRC staff agrees that the intent of the guidance is to ensure that following transfer of control to the SSF, cables and equipment credited for shutdown (alternative or dedicated) are independent of the fire area of concern. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1.5	Initial Conditions	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1.6	Other Events in Conjunction with Fire	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1.1.7	Offsite Power	Aligns With Intent. Oconee relies upon Keowee hydro station to provide emergency onsite power. The cascading power supply analysis determines fire impact to offsite power sources and is utilized in the analysis of fire areas for SSD functions to determine availability of all credited power sources. The adverse consequences of offsite power being available is considered in the NSCA.	By letter dated July 30, 2010 (Reference 43), the licensee clarified an apparent discrepancy between the information presented in Table B-2 of the October 2008 LAR (Reference 2) and the April 2010 LAR (Reference 11). Specifically, Section 3.1.1.7 of the October 2008 LAR B-2 Table states that offsite power has not been analyzed or demonstrated to be free of fire damage for redundant shutdown. However, the April 2010 LAR Alignment Basis indicates that the availability of offsite power has, in fact, been analyzed. In its response, dated September 13, 2010 (Reference 12), the licensee states that the credited power supplies are the Keowee Hydro Station (KHS) and the SSF DG and neither the KHS nor the SSF DG requires offsite power. The licensee also states that the adverse consequences of offsite power being available are considered in the NSCA. To ensure clarity, the licensee has created an action item in the ONS corrective action program (CAP) to revise calculation OSC- 9291, NFPA 805 Transition B-2 Table, Section 3.1.1.7 to reword the alignment basis to clearly state that offsite power is not credited for the deterministic analysis and therefore not analyzed for its availability in the deterministic analysis. The licensee also states that alignment statement will also be revised to ensure the proper relationship with the alignment basis. Completion of the CAP item is an implementation item (SE Section 2.9, Table 2.9-1, Item 47). Based on the response provided in the September 13, 2010 letter (Reference 12), the staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1.1.8	Safety Related Equipment	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1.9	72-Hour Coping	The licensee stated that the approach used aligns with the intent of the 72-hour coping criterion specified in the NEI 00-01 guidance. NFPA 805 does not have any explicit requirements to achieve cold shutdown within 72 hours; therefore, the NFPA-805 criteria for nuclear safety performance goals have been applied to ensure the fuel is maintained safe and stable.	See NRC staff evaluation for SE Section 3.2.1, Item 5. Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach to demonstrate the capability to maintain the plant in a safe and stable condition following a fire is acceptable.
3.1.1.10	Manual / Automatic Initiation of Systems	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion. Manual initiation of components and systems from either the MCR or Emergency (local) control stations have been credited as acceptable compliance strategies where permitted by current regulations or approved by NRC. Automatic initiation of components is not credited.	Manual initiation of components and systems may provide an acceptable compliance strategy where permitted by current regulations or approved by NRC. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.1.11	Multiple Affected Units	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion. Oconee shares some equipment between units. Fire impacts at the component level have been evaluated for impact on each unit.	Although Oconee shares some equipment between units, the evaluation considered the impact of fire damage on each unit. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2	Shutdown Functions	Detailed alignment discussed below.	See NRC staff's evaluation for individual subsections below.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1.2.1	Reactivity Control	Aligns	The NRC staff's finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.2	Pressure Control Systems	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion. Pressure control is accomplished utilizing reactor makeup from the SSF or injection from HPI in conjunction with pressurizer heaters, safety relief valves, PORV's, RCS loop high point vent valves, or reactor head vent valves and controlling decay heat removal rates. An assured success path is determined during the Fire Area Analysis.	The systems discussed in this section of NEI 00-01 are examples of systems that can be used for pressure control. This does not restrict the use of other systems for this purpose. The licensee states that an assured success path is determined for each fire area. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.3	Inventory Control	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.4	Decay Heat Removal	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion. SG(s) are fed from emergency feed water to remove decay heat under natural circulation conditions. Main steam safety valves are utilized for decay heat removal in hot standby. NFPA 805 does not have any explicit requirements to achieve cold shutdown, therefore the NFPA 805 criteria for the Nuclear Safety Performance Goals have been applied to ensure the fuel is maintained in a safe and stable condition.	The NRC staff agrees that NFPA 805 does not have any explicit requirements to achieve cold shutdown, therefore the NFPA 805 criteria for the Nuclear Safety Performance Goals have been applied to ensure the fuel is maintained in a safe and stable condition. In addition, there is no restriction on the use of systems other than those identified in this criterion. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1.2.5	Process Monitoring	The licensee stated that the approach used to address the process monitoring function aligns with the intent of the NEI 00-01 criterion because an exemption for boron sampling in lieu of neutron source range monitoring instrumentation has been granted for the SSF, therefore neutron source range instrumentation has not been provided or analyzed for the SSF. SG pressure instruments are also not provided in the SSF and was accepted by the NRC. Both Neutron Instrumentation and SG pressure indication are provided in the Main Control Room.	NEI 00-01 refers to Information Notice 84-09 Attachment 1, Section IX, which specifies the process variables, deemed necessary to perform and control the reactor shutdown functions. The specific instruments provided may be based on operator preference, SSD procedural guidance strategy (symptomatic vs. prescriptive), and systems and paths selected for SSD. In its letter dated April 14, 2010, (Reference 11), the licensee identified the process monitoring and diagnostic instrumentation credited for SSD from both the MCR and SSF. The licensee further states that the variances from the process variables identified in Information Notice (IN) 84-09 are consistent with its license basis to the extent that the use of boron sampling and a lack of SG pressure instruments at the SSF have been previously approved in an SE dated August 31, 1983 (Reference 40). This approved exemption is being carried forward into the RI/PB FPP and the licensee's statement that the original basis for the exemption remains valid was found acceptable by the NRC staff (see SE Attachment D). Based on the above, the NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.6	Support Systems	Detailed alignment discussed below.	See NRC staff evaluation for subsections below.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.1.2.6.1	Electrical Systems	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion. Emergency onsite power is provided from the KHS on the site.	In lieu of using emergency DGs, the licensee will rely upon the KHS as an emergency onsite power source. In addition, the licensee states that alternating current (AC) and direct connect (DC) power supplies have been analyzed and both the SSF and credited plant battery chargers will be available. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.6.2	Cooling Systems [Main Section]	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.2.6.2	Cooling Systems [HVAC]	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.3	Methodology for Shutdown System Selection	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.1.3.1	Identify SSD Functions	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.3.2	Identify Combinations of Systems that Satisfy Each SSD Function	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.3.3	Define Combinations of Systems for Each SSD Path	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.1.3.4	Assign Shutdown Paths to Each Combination of Systems	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.2	SSD Equipment Selection	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.2.1	Criteria/ Assumptions	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.2.1.1	Primary Secondary Components	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion.	The licensee stated that there was no segregation of SSEL components. Some of the components defined as secondary were captured by the cable selection process and others are captured within the cascading interlocks analysis as pseudo-components. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.1.2	Fire Damage to Mechanical Components (not electrically supervised	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.1.3	Manual Valve Positions	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.1.4	Check Valves	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.1.5	Instrument Failures	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.2.1.6	Spurious Components	The licensee stated that the approach used aligns with the intent of the NEI 00-01 criterion that states that equipment that could spuriously operate should be identified in the equipment selection phase and that Bin 1 of RIS 2004-03 should be considered during the equipment identification process.	The licensee states that spurious operation was considered in identification of SSEL components. RIS 2004-03 (Reference 45) was referenced, however no initial effort was made to 'bin' the types of potential spuriously operating components or their cables. Spurious operation was considered later during compliance assessment when circuit analysis was performed to determine if potential spurious operation was a concern requiring mitigating actions or other compliance strategies. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.1.7	Instrument Tubing	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.2	Methodology for Equipment Selection	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.2.2.1	Identify the System Flow Path for Each Shutdown Path	The licensee stated that the approach used to identify the system flow paths of SSD components aligns with the intent of the requirement since the piping and instrument drawings (P&ID's) were marked up to determine flow and diversion paths which were then translated into SSD Success path logic diagrams. These logic diagrams were then used to identify potential SSEL components.	Flow and diversion paths were identified and translated into SSD Success path logic diagrams. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

Attachment B, Nuclear Safety Capability Assessment Method Review

Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.2.2.2	Identify the Equipment in Each SSD System Flow Path Including Equipment That May Spuriously Operate and Affect System Operation	The licensee stated that the approach used to identify the system flow paths of SSD components aligns with the intent of the requirement since P&ID's and electrical one lines were marked up to determine flow and diversion paths for SSD functions and to identify potential SSEL components including spurious operations. SSD success paths were then translated into SSD Logic Diagrams.	The approach used is functionally equivalent to that specified in NEI 00-01. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.2.3	Develop a SSD Equipment List and Assign the Corresponding System and SSD Path(s) to Each	The licensee stated that the approach used to develop the SSD Equipment List (SSEL) aligns with the intent of this requirement. P&ID's were marked up to determine flow and diversion paths for SSD functions and to identify potential SSEL components including spurious operations. An iterative process was utilized to arrive at the final SSEL based on additional support components identified during the cable selection process. NEI 00-01 Attachment 3 was not utilized, since the SSD database has its own data entry format, which provides the necessary equipment information.	The approach implemented by the licensee is functionally equivalent to that specified in NEI 00-01. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.2.2.4	Identify Equipment Information Required for the SSD Analysis	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.2.2.5	Identify Dependencies Between Equipment, Supporting Equipment, SSD Systems and SSD Paths	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3	SSD Cable Selection and Location	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.3.1	Criteria/ Assumptions	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.3.1.1	Cable Selection	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.1.2	Cables Affecting Multiple Components	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.1.3	Isolation Devices	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.1.4	Identify "Not Required" Cables	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.1.5	Identification of Power Supplies	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.3.1.6	Engineered safety features actuation system (ESFAS) Initiation	The licensee stated that the approach used aligns with the intent of this requirement. Automatic initiation logic was not credited for performance of SSD functions. Manual operation of components from the Main Control Room, SSF or locally were identified during the fire area compliance assessment task as needed. To preclude adverse impact from automatic initiation logic circuits or control logic circuits where multiple components receive signals from common control logic, the control logic was analyzed as a primary component and a pseudo component was created for the logic with cables selected accordingly. This same methodology was used for similar circuit scenarios such as common power supplies. In this way the effects of a fire-induced failure causing spurious component operation were fully evaluated.	The approach used to evaluate the impact of the ESFAS initiation is functionally equivalent to NEI 00-01. The licensee states that VFDRs were identified and evaluated in the FREs to assess the impact of the VFDR and any necessary recovery actions to mitigate the effects of the VFDR. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.1.7	Circuit Coordination	**Does Not Align.** Proper coordination of common power supplies for all circuits was an assumption of the analysis. ONS existing coordination study does not include all SSEL related power supplies. The coordination study needs to be updated with the additional power supplies to ensure that the assumptions of the EIR remain valid.	Based on the LAR, as supplemented, the NRC staff finds that the licensee's approach addressed the issue of inadequate breaker coordination is acceptable and that the licensee states ONS will comply with the requirements of NFPA 805, Section 2.4.2.2.2.(a) upon completion of the committed plant modifications and implementation items. (see SE Sections 2.8, Table 2.8.1-1 and 2.9, Table 2.9-1).
3.3.2	Associated Circuit Cables	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.3.2 [A]	Associated Circuit Cables – Cables Whose Failure May Cause Spurious Actuations	Not a review criterion - Generic paragraph. This section only describes spurious actuation concern. Section 3.3.3 below addresses the methodology for selecting cables whose failure may cause spurious operations.	See NRC staff's evaluation for Section 3.3.3 below.
3.3.2 [B]	Associated Circuit Cables – Common Power Source Cables	Not a review criterion - Generic paragraph. Provides brief description of common power source concern. Section 3.5.2.4 below addresses the methodology for analyzing circuit failures due to inadequate circuit coordination.	See NRC staff's evaluation for Section 3.5.2.4 below.
3.3.2 [C]	Associated Circuit Cables – Common Enclosure Cables	Not a review criterion - Generic paragraph. Provides brief description of common enclosure concern. Section 3.5.2.5 below addresses the methodology for analyzing circuit failures due to common enclosure concerns.	See NRC staff's evaluation for Section 3.5.2.5 below.
3.3.3	Methodology for Cable Selection and Location	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.3.3.1	Identify Circuits Required for the Operation of the SSD Equipment	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.3.3.2	Identify Interlocked Circuits and Cables Whose Spurious Operation or Mal-operation Could Affect Shutdown	The licensee stated that the approach used aligns with the intent of this requirement; for control logic circuits where multiple components receive signals from common control logic or interlocks, the control logic was analyzed as a primary component and a pseudo component was created on the SSEL for the logic with cables selected accordingly. Pseudo-components whose associated cabling can affect another primary component based on common power were identified in the cable selection for the affected component as an interlocked primary component. The cascading power supply and cascading interlocks analyses evaluate these interlocked components.	The approach used by the licensee addresses interlocked components. The staff finds this acceptable based on the description of the process used, where a deviation from the endorsed guidance occurs, a conservative assumption was used in the circuit analysis (interlocked contact or relay will be assumed to be in the worst-case position). The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.3.3.3	Assign Cables to the SSD Equipment	**Does not Align** Coordination of power supplies was assumed when assigning cables to the SSD equipment; Oconee does not meet the intent of the guidance since it did not consider inadequate breaker coordination when selecting cables.	Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach has adequately addressed the issue of inadequate breaker coordination and that, upon completion of the modifications and implementation items, ONS will align with the endorsed guidance for this attribute. (See SE Section 3.2.1).
3.4	Fire Area assessment and Compliance Assessment	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.4.1	Criteria/Assumption	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.4.1.1	Number of Postulated Fires	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.2	Damage to Unprotected Equipment and Cables	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.3	Assess Impacts to Required Components	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.4	Manual Actions	The licensee stated that the approach used aligns with the intent of this requirement; the least impacted SSD success path was analyzed and Variances from the Deterministic Requirements (VFDRs) were identified. Mitigating strategies to address the VFDRs in a PB FRE were developed and documented. One of the potential mitigating strategies is procedural action (recovery action) to mitigate the operational effects from fire damage.	The licensee states variances from the nuclear safety performance criteria deterministic approach were evaluated as a FRE per Section 4.2.4.2 of NFPA 805. If the FRE meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the PB approach is acceptable per Section 4.2.4.2 of NFPA 805. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.5	Repairs	The licensee stated that the approach used aligns with the intent of this requirement; The least impacted SSD success path was analyzed and Variances from the Deterministic Requirements (VFDRs) were identified. Mitigating strategies to address the VFDRs in a PB FRE were developed and documented. One of the potential mitigating strategies is repairs (recovery action) to mitigate the operational effects from fire damage.	The licensee states variances from the nuclear safety performance criteria deterministic approach were evaluated as a FRE per Section 4.2.4.2 of NFPA 805. If the FRE meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the PB approach is acceptable per Section 4.2.4.2 of NFPA 805. NFPA 805 does not have any explicit requirements to achieve cold shutdown. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.4.1.6	Assess Compliance with Deterministic Criteria	The licensee stated that the approach used aligns with the intent of this requirement; The least impacted SSD success path was analyzed and Variances from the Deterministic Requirements (VFDRs) were identified. Mitigating strategies to address the VFDRs in a PB FRE were developed and documented. The methods described above are options to satisfy the deterministic criteria to preclude identification of VFDRs.	The licensee states variances from the nuclear safety performance criteria deterministic approach were evaluated as a FRE per Section 4.2.4.2 of NFPA 805. If the FRE meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the PB approach is acceptable per Section 4.2.4.2 of NFPA 805. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.7	Consider Additional Equipment	The licensee stated that the approach used aligns with the intent of this requirement; the least impacted SSD success path was analyzed and Variances from the Deterministic Requirements (VFDRs) were identified. Mitigating strategies to address the VFDRs in a PB FRE were developed and documented. The methods described above are options to satisfy the deterministic criteria to preclude identification of VFDRs.	The licensee states variances from the nuclear safety performance criteria deterministic approach were evaluated as a FRE per Section 4.2.4.2 of NFPA 805. If the FRE meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the PB approach is acceptable per Section 4.2.4.2 of NFPA 805. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.1.8	Consider Instrument Tubing Effects	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.2	Methodology for Fire Area Assessment	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.4.2.1	Identify the Affected Equipment By Fire Area	Aligns	The NRC staff finds the licensee's statement of alignment to the guidance acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.4.2.2	Determine the Shutdown Paths Least Impacted by a Fire in Each Fire Area	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.2.3	Determine SSD Equipment Impacts	The licensee stated that the approach used aligns with the intent of this requirement; The SSEL and logics were developed based on potential spurious operations and other plant impacts by their selection from a functional basis. The fire area analysis methodology assumes multiple fire-induced failures and multiple spurious actuations, based on the SSD cables and components present in the fire area of concern. All postulated SSD cable and component failures were identified and a resolution provided at the cable or component level for the credited train.	The staff was concerned that the MSO analysis was limited to only SSD cables and components. In its October 14, 2010 response (Reference 54), the licensee states that the methodology assumed multiple fire-induced failures and multiple spurious actuations based on the cables and components present in the fire area of concern, and was not limited to SSD cables and components. Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's basis for alignment to Section 3.4.2.3 of NEI 00-01 acceptable.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.4.2.4	Develop a Compliance Strategy or Disposition to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable	The licensee stated that the approach used aligns with the intent of this requirement; the SSD success paths were analyzed and potential impacts identified. These potential impacts were resolved such that the least impacted SSD success path could be identified. Variances from the Deterministic Requirements (VFDRs) were identified. Mitigating strategies to address the VFDRs in a PB FRE were developed and documented. Credit for existing features and exemptions was taken wherever possible and procedural (recovery) action specified as a last resort.	The approach used to determine SSD equipment impacts is functionally equivalent to NEI 00-01. The licensee states variances from the nuclear safety performance criteria deterministic approach were evaluated as a FRE per Section 4.2.4.2 of NFPA 805. If the FRE meets the acceptance criteria, this is confirmation that a success path effectively remains free of fire damage and that the PB approach is acceptable per Section 4.2.4.2 of NFPA 805. The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.4.2.5	Document the Compliance Strategy or Disposition Determined to Mitigate the Effects Due to Fire Damage to Each Required Component or Cable	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5	Circuit Analysis and Evaluation	Not Required. Generic paragraph. Detailed alignment discussed below.	See NRC staff's evaluation for individual subsections below.
3.5.1	Criteria/Assumption	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.5.1.1	Circuit Failure Types and Impact	The licensee stated that the approach used aligns with the intent of this requirement; All combinations of circuit failures except inter-cable hot shorts are considered and evaluated to determine if spurious component actuation can occur. Intercable hot shorts were not considered due to the use of armored cable at Oconee.	Based on the information provided in Reference 10, the NRC staff finds that the licensee's approach has adequately addressed the issue of grounding of armored cable to preclude inter-cable shorts. (See SE Section 3.2.1).
3.5.1.2	Circuit Contacts and Operational Modes	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.1.3	Duration of Circuit Failures	**Does not Align, but has previous NRC approval.** Previous design considerations did not assume spurious actuations or hot shorts due to a fire for the first 10 minutes of the event. This was stated in the referenced SER for the SSF. Other spurious operations beyond this assumption were postulated to occur until mitigating actions are taken.	In a letter dated November 19, 2010 (Reference 52) the licensee agreed to eliminate the "*10 minute free of fire damage*" assumptions and to perform an evaluation using NFPA 805 risk-informed processes. Specifically, the licensee states that it will utilize a risk-informed approach to evaluate conflicts that previously relied upon the 10-minute prior approval. This will involve a thorough review of existing analyses to identify new VFDRs. Changes to the FPP, as a result of these VFDRs, will be resolved using the change evaluation process. Upon completion of this activity, all applicable FRE(s) will be updated and compliance will be demonstrated consistent with NFPA 805, Section 4.2.4.2. (See SE Section 3.2.1, Item 2, for more detail) Based on the information provided, the NRC staff finds that the licensee's process for eliminating the "*10-minute free of fire damage assumption*" provides reasonable assurance that the safety objectives of NFPA 805 will be satisfied.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.5.1.4	Cable Failure Configurations	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.1.5 [A]	Circuit Failure Risk Assessment Guidance	Detailed alignment discussed below.	See NRC staff evaluation for subsections below.
3.5.1.5 [B]	Cable Failure Modes	The licensee stated that the approach used aligns with the intent of this requirement; Oconee has armored sheathing - cable-to-cable hot shorts are not postulated for armor-jacketed cables.	See NRC staff's evaluation for SE Section 3.2.1. Based on the information contained in the licensee's letter dated September 13, 2010 (Reference 12), the NRC staff finds that the licensee's approach has adequately addressed the issue.
3.5.1.5 [C]	Likelihood of Undesired Consequences	**Does not Align** Treatment of multiple spurious actuations is being resolved through transition to NFPA-805.	The NRC staff was concerned that the MSO analysis was limited to only SSD cables and components. In its October 14, 2010 letter (Reference 54), the licensee states that the methodology assumed multiple fire-induced failures and multiple spurious actuations based on the cables and components present in the fire area of concern, and was not limited to SSD cables and components. Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's basis for alignment to Section 3.5.1.5[C] of NEI 00-01 acceptable.
3.5.2	Types of Circuit Failures	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.5.2.1	Circuit Failures Due to Open Circuits	**Does not Align** Open circuits are analyzed as shown on the referenced figures from NEI 00-01 except CT circuits.	By letter dated September 13, 2010 (Reference 12), the licensee states, in part, that the assumption associated with the secondary CT circuits is being removed to ensure that ONS has properly evaluated the effects of an open secondary CT as prescribed in NFPA 805, Section 2.4.2 and guided in NEI 00-01, Section 3.5.2.1. Completion of this action is an implementation item (SE Section 2.9, Table 2.9-1, Item 40). (See SE Section 3.2.1) The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.2.2	Circuit Failures Due to Shorts to Ground [General]	Detailed alignment discussed below.	See NRC staff's evaluation for subsections below.
3.5.2.2	Circuit Failures Due to Shorts to Ground [A, Grounded Circuits]	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.2.2	Circuit Failures Due to Shorts to Ground [B, Ungrounded Circuits]	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.2.3	Circuit Failures Due to a Hot Short [General]	Detailed alignment discussed in subsequent paragraphs below.	See NRC staff's evaluation for subsections below.

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
3.5.2.3	Circuit Failures Due to a Hot Short [A, Grounded Circuits]	The licensee stated that the approach used aligns with the intent of this requirement intra-cable conductor-to-conductor hot shorts are analyzed. The external hot short is not considered credible at ONS due to the armored cable configuration.	Based on the information contained in the licensee's letter dated September 13, 2010 (Reference 12), the NRC staff finds that the licensee's approach has adequately addressed the issue. (See SE Section 3.2.1, Item 1).
3.5.2.3	Circuit Failures Due to a Hot Short [B, Ungrounded Circuits]	Aligns	The NRC staff finds the licensee's statement of alignment to the endorsed guidance acceptable.
3.5.2.4	Circuit Failures Due to Inadequate Circuit Coordination	**DOES NOT ALIGN** Proper coordination of common power supplies for all circuits was an assumption of the analysis. The licensee's existing coordination study does not include all SSEL related power supplies.	Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach has adequately addressed the issue of inadequate breaker coordination at ONS and that, upon completion of the modifications and implementation items, ONS will align with the endorsed guidance for this attribute. (see SE Section 3.2.1, Item 1)
3.5.2.5	Circuit Failures Due to Common Enclosure Concerns	**DOES NOT ALIGN** The electrical circuit design for ONS is assumed to provide proper circuit protection in the form of circuit breakers, fuses and other devices that are designed to isolate cable faults before ignition temperature is reached. Adequate electrical circuit protection and cable sizing were included as part of ONS plant electrical design. However, as discussed in Section 3.3.3.3, the breaker coordination study for ONS does not include all SSD equipment and the analysis is required to be updated to ensure coordination exists. Due to the uncertainty	Based on the information provided in the LAR, as supplemented, the NRC staff finds that the licensee's approach has adequately addressed the Common Enclosure Associated Circuit concern at ONS and that ONS complies with the requirements of NFPA 805, Section 2.4.2.2.2.(b). (See SE Section 3.2.1, Item 4).

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Table 3.2-1: Nuclear Safety Capability Assessment Method Review

NEI 00-01 Section	Section Title	ONS Alignment Basis	NRC Staff's Evaluation
		of breaker coordination, ONS does not meet the intent of the guidance.	

Attachment C, Fire Risk Evaluation Tables

The licensee evaluated the technical adequacy of the portions of its internal events probabilistic risk assessment (PRA) model used to support development of the Fire PRA model by first performing a peer review of the ONS internal events PRA model. Subsequently, a contractor review and a licensee self-assessment were performed. Table 3.4-1, "Internal Events PRA Findings and Observations Resolution," summarizes the NRC staff's review of the licensee's resolution of the internal events PRA findings and observations (F&Os), which demonstrates the technical adequacy for this application.

Since ONS is an industry pilot for NFPA 805, consistent with RG 1.205, Revision 0, the NRC staff performed the review of the licensee's Fire PRA model to determine its technical adequacy because an industry peer review of the ONS Fire PRA model had not yet been performed. In addition, because a full-scope industry peer review of the ONS Fire PRA was not performed, the NRC staff reviewed a number of aspects of the Fire PRA model in detail. Table 3.4-2, "Fire PRA Findings and Observations Resolution," summarizes the NRC staff's review of the licensee's resolution of findings from the NRC staff's review (including both F&Os as well as supporting requirements (SRs) evaluated as less than Capability Category II without any specific F&O). Table 3.4-2 includes both the licensee's reported Topic, and the NRC staff's text from the NRC staff's F&Os. This evaluation establishes the technical adequacy of the ONS Fire PRA for this application.

The licensee provided detailed information regarding the correlations and fire models used to support implementation of NFPA 805 at ONS, as well as a cross reference between major sections of American Society for Testing and Materials (ASTM) guidance document ASTM E 1355-05a, "Standard Guide for Evaluating Predictive Capability of Deterministic Fire Models," and the associated correlations in terms of their applicability and validation. Table 3.4-3, "V&V Basis for Fire Modeling Correlations Used at ONS," identifies the empirical correlations and models used in the screening tool, the basis for acceptability with respect to verification and validation (V&V), and the NRC staff's evaluation of that basis.

Attachment C, Fire Risk Evaluation Tables

Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
AS-B3 The AS [accident sequence] notebooks document phenomenological conditions created by the accident progression. For example, sump temperature is examined when considering the potential for LPI [low pressure injection] pump cavitation, and influences from ambient conditions are examined for high energy line breaks. However, as noted for supporting requirements SY-B8 and SY-B15, SSCs [structures, systems, and components] that may be required to operate in conditions beyond their environmental qualifications are not completely identified and/or documented. Examples include: a) LOCA [loss of coolant accident] inside containment with failure of the Reactor Building Cooling System would expose SG [steam generator] instrumentation to a harsh environment; b) Steam line breaks in the TB [turbine building] could expose equipment other than just the 4 kV switchgear and EFW control panel to an adverse environment; c) Clogging of the RBES [reactor building emergency sump] is not discussed.	Accident sequence notebooks and system model notebooks should identify those environmental effects of the initiating event and the impact on mitigation systems.	The NRC staff does not accept that the F&O is resolved solely by documentation. However, in response to RAI 5-9i (Reference 8) and 5-53 (Reference 9), the licensee described the possible impact of the environment in the three example accident scenarios in the comment. Based on the disposition of the examples, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
DA-B1 As documented in calculation OSC-8796, mechanical components are grouped according to component type and system (HPI, EFW, SW, etc.). Electrical components are grouped based on component type and voltage level. However, since a generic database (NUREG/CR-6928) that separates standby and operating component failure rates is now available, consider separating these groups.	Revise the data calc. to segregate standby and operating component data. Segregate components by service condition to the extent supported by the data. This is a refinement to the equipment failure rates. However, since most components are grouped appropriately, the overall impact should be small.	Based on the type of changes to the PRA identified by the licensee, the NRC staff finds that changes are expected to be very small and thus this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
DA-D4 There is no evidence in the documentation that the specific checks required by this SR have been performed on the Bayesian-updated data to ensure that the data is appropriate. However, a verification of the proper operation of the software within the expected data range (item d of the SR) was performed. A quick review of the current data did not reveal any unusual or unexpected results, however.	Enhance the documentation to include a discussion of the specific checks performed on the Bayesian-updated data, as required by this SR.	The NRC staff does not accept that the F&O is resolved solely by documentation. However, the NRC staff finds that a Bayesian update is not likely to cause the estimated transition risk decrease to become a risk increase.
DA-D6 Plant-specific CCF [common cause failure] failure documentation (OSC-8797) was reviewed to ensure that the generic CCF estimates were consistent with plant operating experience. However no evidence is provided to show that the component boundaries used in the CCF generic estimates are consistent with the component boundaries assumed for the PRA.	Provide documentation in SAAG 637 of the comparison of the component boundaries assumed for the generic CCF estimates to those assumed in the ONS PRA to ensure that these boundaries are consistent.	The NRC staff does not accept that the F&O is resolved solely by documentation. However, the licensee provided a review comparing the ONS CCF events with generic electric system CCFs in its response to RAI 5-9e (Reference 8). The review identified differences in the boundaries but concluded that plant operating experience is properly monitored for potential CCF events based on ONSs modeling. Therefore, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
HR-A2 No documentation was found of a review of procedures and practices to identify calibration activities that if performed incorrectly can have an	Enhance the HRA [human reliability analysis] to consider the potential for calibration errors. Based on preliminary	Based on the assessment that calibration (human error probabilities (HEPs) are not

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
adverse impact on the automatic initiation of standby safety equipment. The Oconee PRA assumes a Type A screening value for each standby PRA system train, as well as a common cause Type A screening value where identified to be appropriate. A review of procedures and practices would provide a worthwhile cross-check of this approach, to ensure that no Type A events have been overlooked. For example, the review may identify a tank level instrument calibration that would prevent the operation of redundant pump trains due to interlocks, which may not have been previously captured as part of assigning common cause Type A screening values.	evaluations using the EPRI HRA calculator, calibration errors that result in failure of a single channel are expected to fall in the low 10^{-3} range. Calibration errors that result in failure of multiple channels are expected to fall in the low 10^{-5} range. Relative to post-initiator HEPs], equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability.	expected to contribute significantly to overall equipment unavailability, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
<u>HR-A3</u> The Oconee PRA identified common cause Type A HFEs [human failure events] that effect different trains of redundant systems where considered to be appropriate. No documentation was found of a review of procedures and practices to identify maintenance and calibration activities that could render equipment unavailable if performed incorrectly. Such a review of procedures and practices would provide a worthwhile cross-check of this approach, to ensure that no Type A events have been overlooked that simultaneously affect equipment in either different trains of a redundant system or diverse systems.	Identify maintenance and calibration activities that could simultaneously affect equipment in either different trains of a redundant system or diverse systems. Relative to post initiator HEPs, equipment random failure rates and maintenance unavailability, calibration HEPs are not expected to contribute significantly to overall equipment unavailability. See the Expected Impact on Applications for requirement HR-A2 above.	Based on the assessment that calibration HEPs are not expected to contribute significantly to overall equipment unavailability, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
<u>HR-D6</u> The Type A HEPs are not identified to be mean values and error factors are not provided in the summary table of the HR [human reliability] notebook	Develop mean values for pre-initiator HEPs. Pre-initiator HEPs are generally set to relatively high screening values.	Based on the sensitivity study provided in the September 29, 2009 RAI response, the NRC staff

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
(Table 2)	Thus the suggested data refinement is not expected to have a significant impact on this application.	finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
<u>HR-G3</u> The human cognitive reliability [HCR] model or the caused-based approach was used to quantify cognition errors, and an abbreviated version of THERP to quantify execution errors. These evaluation of cognitive errors explicitly included considerations of plant-specific and scenario-specific performance shaping factors noted by this supporting requirement. Execution errors were not computed if cognition error probabilities dominated the human error probabilities, but more detailed documentation would be desirable to support such conclusions. In many instances execution errors were assigned bounding screening values. These screening value assignments took into account a simplified set of scenario specific performance shaping factors. It is not clear that full consideration of performance shaping factors has been adequately taken into account.	Document in more detail the influence of performance shaping factors on execution of human error probabilities.	The NRC staff does not accept that this F&O is resolved solely by documentation. The NRC staff finds that enhancing the human reliability analysis (HRA) models to address this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
<u>HR-G4</u> The total time available to complete actions was generally obtained from thermal-hydraulic calculations for the accidents of interest (e.g., from MAAP analyses, hand calculations, or other sources). In many cases, human interactions were assessed to not be time critical, and no estimate of the time available was needed. In other instances, adequate estimates on event timing were not available to	Enhance HRA documentation accordingly.	See NRC staff finding in HR-G3.

Attachment C, Fire Risk Evaluation Tables

Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
support HCR assessment and the caused-based decision tree approach was used. More detailed documentation to support these decisions and conclusions is desirable.		
HR-G6 The PRA notebooks do not document a review of the HFEs and their final HEPs relative to each other to check reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	Document a review of the HFEs and their final HEPs relative to each other to confirm their reasonableness given the scenario context, plant history, procedures, operational practices, and experience.	See NRC staff finding in HR-G3.
HR-G9 The Oconee post-initiator HRA does not provide mean HEP values for use in the quantification of the PRA results.	Develop mean values for post-initiator HEPs. The use of mean values for HEPs instead of lower probability median values can affect the PRA results. The fire analysis will include a sensitivity study to evaluate the use of different HEPs if the calculated risk is close to the threshold.	See NRC staff finding in HR-D6.
HR-H2 The HR notebook documents that some operator recovery actions are credited in the Oconee PRA for which no procedures are available. More detailed documentation would be desirable of the cues that alert the operator to perform the recovery actions, relevant performance shaping factors, and availability of sufficient manpower to perform the action.	Develop more detailed documentation of operator cues, relevant performance shaping factors, and availability of sufficient manpower to perform the action.	See NRC staff finding in HR-G3.
IF-B3 The current analysis identifies the capacities of various tanks in the plant and identifies the flow rates for the defined initiating event definitions. The current analysis does not address spray events, does	Enhance the Internal Flood analysis to address the potential for spray, jet impingement, and pipe whip failures. Additionally, document how these	The licensee stated in its LAR that accident scenarios initiated by internal flooding are not expected to impact the results of the fire

Attachment C, Fire Risk Evaluation Tables

Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
not identify the potential volume of water that can be lost through a pipe rupture (limited source versus an unlimited source), and does not use system pressures to calculate potential flow rate from a pipe rupture. Such information is necessary to satisfy the requirements of the ASME PRA Standard.	failures are included in the quantification. Internal flooding modeling issues do not impact fire risk.	analysis. The NRC staff finds that only flooding as a result of fire-fighting could impact the fire results, and such flooding is addressed independently of the internal events PRA as described in Section 4.2.4 of the LAR. Therefore, the NRC concludes that this deficiency will not cause the estimated transition risk decrease to become a risk increase.
IF-C2c For those flood areas addressed in the current flooding analysis, equipment important to accident mitigation and the associated critical flood heights are identified. However, given the expected increase in number of flood areas to be explicitly addressed, additional equipment will need to be identified and discussed in order to meet the requirements of the ASME PRA Standard. The current flooding analysis does not discuss flood mitigative features and this will have to be corrected to satisfy the requirements of the ASME PRA Standard.	Given the expected increase in number of flood areas needed to satisfy requirement IF-A1, additional equipment will need to be identified and discussed in order to meet the requirements of the ASME PRA Standard. The current flooding analysis does not discuss flood mitigative features and this will have to be corrected to satisfy the requirements of the ASME PRA Standard. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.
IF-C3 The current flooding analysis identifies the submergence failure height of the equipment important to accident mitigation, but never addresses the impact of spray. Spray as a failure mechanism needs to be addressed in the analysis or a note	The current flooding analysis identifies the submergence failure height of the equipment important to accident mitigation, but never addresses the impact of spray. Spray as a failure	See NRC staff finding in IF-B3.

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
made explaining why it was omitted.	mechanism needs to be addressed in the analysis or a note made explaining why it was omitted. Internal flooding modeling issues do not impact fire risk.	
IF-C3b Discussion of propagation in the current flooding analysis is simply that water from any pipe break in the AB will eventually drain down to the basement where it will begin to accumulate. The mechanisms by which water will propagate are not discussed. Given the expected increase in number of flood areas, additional propagation paths will likely be identified and the mechanisms by which water will propagate will have to be added to the discussion.	Provide more analysis of flood propagation flowpaths. Address potential structural failure of doors or walls due to flooding loads and the potential for barrier unavailability. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.
IF-E5 Some discussion of human errors is provided in the current flooding analysis, however the details that impact the development of performance shaping factors and the timelines that are essential to any human reliability analysis are not included.	Develop and document any HEPs in a manner comparable to that used in developing HEPs for the internal events PRA. Use of EPRI HRA method will address this SR. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.
IF-E5a Some discussion of human errors is provided in the current flooding analysis; however the details that impact the development of performance shaping factors and the timelines that are essential to any human reliability analysis are not included.	Develop and document any HEPs in a manner comparable to that used in developing HEPs for the internal events PRA. Use of EPRI HRA method will address this SR. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.
IF-E6b The current flooding analysis addresses the submergence failure heights of various equipment, however no discussion of spray, jet impingement, or pipe whip failures is included. Additionally, due to the lack of quantification information it is unknown how	Address potential indirect effects. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
any of these failures were included in the quantification. A complete discussion of the quantification in needed to satisfy the requirements of the ASME Standard		
IF-F2 The flood analysis documentation does not address all of the items identified in this requirement.	Need to document how the analysis addressed all of the items identified in this requirement. Internal flooding modeling issues do not impact fire risk.	See NRC staff finding in IF-B3.
LE-C6 The only mention of operator actions credited in the Level 2 analysis was in Section 7.3 of SAAG #818. However, the contribution to LERF [large early release frequency] from failure to perform this action is assumed to be negligible compared to other LERF sequences.	The only operator action expected to be important is RCS [reactor coolant system] depressurization for small LOCAs. However, the current analysis lacks a formal dependency analysis for this action. The result is expected to be insensitive to this impact given that the SGTR [steam generator tube rupture] so dominates the result.	See NRC staff finding in HR-D6.
LE-F2 The assumptions in the model development are presented in Section 5 of SAAG #818. However, their impact on the resulting LERF has not been evaluated. A parametric uncertainty analysis of the LERF results is presented in the Integration Notebook. Sensitivity analyses and their insights have not been documented.	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	In response to RAIs 5-55 and 5-56 (Reference 9), the licensee reported re-evaluating various assumptions used in the LERF estimates. Based on this re-evaluation of the more significant LERF pathways, the NRC staff concludes that a sensitivity study, including developing the appropriate documentation, would not cause the estimated transition risk decrease to become a risk increase. Therefore, the NRC

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
		staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
LE-F3 Tables 4.5.8-2 d and e of the ASME PRA Standard include requirements such as documenting a review of the dominant contributors to LERF, comparing the overall LERF and LERF dominant contributors to similar plants, and evaluating the overall LERF uncertainty intervals. The Integration Notebook presents the uncertainty band around the mean LERF, but these other requirements have not been performed.	Compare LERF results and uncertainties to similar plants and include in the LERF documentation.	Comparing LERF results with similar plants is not likely to significantly change the results. The NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
LE-G3 SAAG #818 describes the development of the Oconee LERF models, and the Integration Notebook presents the point estimate and uncertainty results. The documentation does not assess the relative contribution of the PDSs [plant damage states], etc. to the LERF.	Evaluate the relative contribution of the various contributors to the total LERF	Documenting the relative contribution of the PDSs is less important than properly linking the PDS frequencies with the conditional LERF. The review included no findings on inappropriate linking. The NRC staff finds that developing the appropriate documentation would not cause the estimated transition risk decrease to become a risk increase.
LE-G4 The assumptions in the model development are presented in Section 5 of SAAG #818. However, their impact on the resulting LERF has not been evaluated. The mean, 5th and 95th percentile LERF uncertainty values are presented in the Integration	Perform and document sensitivity studies to determine the impact of the assumptions and sources of model uncertainty on the LERF results.	This observation is made on an SR that identifies documentation requirements instead of technical requirements. The NRC staff finds that developing the

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
Notebook. Sensitivity analyses and their insights have not been documented.		appropriate documentation would not cause the estimated transition risk decrease to become a risk increase.
LE-G5 No evaluation of the limitations in the LERF analysis that could impact applications was presented in the documentation.	Include in the LERF documentation an assessment that identifies the limitations in the LERF analysis that could impact applications.	The NRC staff does not accept that the F&O is resolved solely by documentation. However, in response to RAI 5-55 (Reference 9), the licensee reported that previously screened containment penetrations and Interfacing System Loss of Coolant Accident pathways (which are the most likely non-SGTR LERF pathways) were re-evaluated including the potential for fire induced multiple spurious operations. Therefore, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
LE-G6 The ASME PRA Standard defines "large, early" as: "the rapid, unmitigated release of airborne fission products from the containment to the environment occurring before the effective implementation of offsite emergency response and protective actions such that there is a potential for early health effects". SAAG #818 presents similar wording in Section 7.1.	Provide a discussion of the significant cut sets and sequences.	The NRC staff finds that this observation is an observation that a documentation SR is not met (as opposed to a technical element). Significant fire LERF sequences are reported in the LAR. Based on the assessment that this is a documentation issue

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
		only, the NRC staff finds that developing the appropriate documentation would not cause the estimated transition risk decrease to become a risk increase.
QU-D3 The Integration Notebook does not include a comparison of results between the ONS PRA and other similar plants.	Perform and document a comparison of results between the ONS PRA and other similar plants.	The NRC staff finds that a comparison of the internal events results with results at similar plants is not likely to cause the estimated transition risk decrease to become a risk increase.
QU-E4 Although general modeling assumptions are provided in the PRA Modeling Guidelines (XSAA-115) and specific assumptions related to system design, operation, and modeling are documented in the various PRA notebooks, the sensitivity of the results to model uncertainties and assumptions has not been thoroughly evaluated.	Perform and document a set of sensitivity cases to determine the impact of the assumptions and sources of model uncertainty on the results. Perform and document sensitivity analyses to determine the impact of the assumptions and sources of model uncertainty on the Fire PRA results.	The NRC staff finds that the Fire PRA results are dominated by assumptions regarding the fire caused failures, and that sensitivity to assumptions not directly related to fires is unlikely to cause the large estimated risk decrease to become a risk increase.
QU-F2 The model integration process and basic quantification results are documented in the Integration Notebook. However, there is no discussion of sensitivity analyses or of some of the other expected contents. Note that although the Modeling Guidelines (XSAA-115) specify that the Integration Notebook is to include an overall description of Train A vs. Train B model asymmetries and a discussion of the impact on results, this	Expand the documentation of ONS PRA model results to address all required items.	This observation is made on an SR that identifies documentation requirements instead of technical requirements. Based on the assessment that this is a documentation issue only, the NRC staff finds that developing the appropriate documentation would not cause the estimated

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
discussion is not included in SAAG 517.		transition risk decrease to become a risk increase.
QU-F6 Section 12.0 of the Integration Notebook notes that the CDF result is dominated by external events tornado, fire and flood. There is no discussion of a specific quantitative definition for significant basic events, cutsets, accident sequences or functional failures.	Document the required definitions.	This observation is made on an SR that identifies documentation requirements instead of technical requirements. Based on the assessment that this is a documentation issue only, the NRC staff finds that developing the appropriate documentation would not cause the estimated transition risk decrease to become a risk increase.
SC-B5 Review of the PRA Modeling Guidelines (XSAA-115), the success criteria (SC) documentation provided (LPI with no LPSW SAAG 569, and HPI for Small and Medium LOCAs SAAG 213), and samples of the AS (ATWS [anticipated transient without scram], Transients - SAAG 671, LOCAs - SAAG 241) and SY [systems] (SSF, EFW SAAG 259, and HPI/CC) notebooks did not indicate that the TH [thermal-hydraulic] results and results of other SC evaluations were consistently check for reasonableness and acceptability. Only SAAG 569 described a cross-disciplinary check of results where the Oconee LPI system engineer reviewed the RB sump temperatures and RB pressures calculated by MAAP to confirm no NPSH [net positive suction head] problems would occur with operating the LPI pumps.	Provide evidence that an acceptability review of the TH analyses is performed.	In its response to RAI 5-9a and 5-9b (Reference 8) about AS-A9 (since closed by the licensee), the licensee described the update of its thermal-hydraulic calculations and provided a new table clearly identifying the SC based on the update. The NRC finds that the update is sufficient to close AS-A9 and that any slight modification that might arise by comparing with other studies will not cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
SC-C1 Only a portion of the SC documentation was provided for review (LPI with no LPSW SAAG 569, and HPI for Small and Medium LOCAs SAAG 213). SC information was also found in some of the AS notebooks reviewed (e.g., LOCA SAAG 241), which included additional SC runs in the appendices; other AS and SY notebooks included general statements of SC and associated references (not all of which were reviewed). In general, the SC documentation is somewhat scattered and dated; a consolidated SC document would facilitate upgrades and review.	Improve the documentation on the TH bases for all safety function SC for all initiators.	See NRC staff finding in SC-B5.
SC-C2 The PRA Modeling Guidelines (XSAA-115), the SC documentation provided (LPI with no LPSW SAAG 569, and HPI for Small and Medium LOCAs SAAG 213), and samples of the AS (ATWS, Transients - SAAG 671, LOCAs - SAAG 241) and SY (SSF, EFW SAAG 259, and HPI/CC) notebooks all contain documentation of some aspect of the process to develop overall PRA SC. According to the HRA notebook, the total time available for performing operator actions was generally obtained from thermal-hydraulic calculations for the accidents of interest (e.g., from MAAP analyses, hand calculations, or other sources). Review of the HRA toolbox spreadsheets indicates that sometimes no reference for time available is provided, or sometimes only a reference number, which were not provided for review. The documentation reviewed is not consistently thorough in presenting input, methods, and results, and not all of the aspects listed above	Improve the documentation on the TH bases for all safety function SC for all initiators.	See NRC staff finding in SC-B5.

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
are clearly documented.		
SY-A14 Some failure modes are excluded in a qualitative fashion rather than by using quantitative criteria. Examples: the containment isolation system write-up notes that "electrical penetrations are not modeled due to their low probability of failure;" the High Pressure Service Water (HPSW) system write-up uses a redundancy argument for excluding inadvertent isolation of the main headers; the RCS write-up states that "transfer failure events for motor-operated valves (MOVs), manual valves and check valves with 24 hr exposure times are not modeled unless probabilistically significant with respect to 'neighboring' basic events; the RPS write-up uses a diversity argument for excluding common mode failure of sensors or instrument strings that generate a reactor scram signal.	Provide quantitative evaluations for screening.	In response to RAI 5-9g (Reference 8), the licensee reviewed the qualitative screening process and concluded that changing to a quantitative approach is not expected to change the results of the qualitative screening. This was verified specifically for the HPSW and MOV examples identified in the F&O. Based on the results of the review of the qualitative screening process, the NRC staff finds that this deficiency is not likely to cause the estimated risk decrease to become a risk increase.
SY-A4 Some of the notebooks include a walk down checklist and system engineer review, others do not.	Enhance the system documentation to include an up-to-date system walkdown checklist and system engineer review for each system. Consider revising workplace procedure XSAA-106 to require that such documentation be revisited with each major PRA revision.	The NRC staff does not accept that the F&O is resolved solely by documentation. However, the results from the Fire PRA used to support the transition are dominated by scenarios where the fire-caused SSC failures are followed by a few random SSC failure events and/or human error events. The NRC staff finds that location-specific attributes, other than affects of fire damage, which

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
		might not be fully reflected in the PRA will not cause the estimated transition risk decrease to become a risk increase.
SY-A8 Component boundaries are consistent with those identified in the data analysis (OSC-8796). For example, breakers that supply a single load are generally included within the boundary of the load. (An exception is the 4 kV circuit breakers.) However, boundaries are not discussed. Interlocks are explicitly modeled in the system models.	Enhance systems analysis documentation to discuss component boundaries. In response to RAI 5-9h (Reference 8), the licensee clarified that the component boundaries are consistent with source documents, such as NUREG/CR-6928, and that the reviewer did not identify any technical issues with the assessment.	The NRC staff does not accept that the F&O is resolved solely by documentation. In response to RAI 5-9h (Reference 8), the licensee clarified that the component boundaries are consistent with source documents, such as NUREG/CR-6928, and that the reviewer did not identify any technical issues with the assessment. Based on the results of the review and subsequent completion of the breaker co-ordination evaluation reported elsewhere in the SE and Attachment C, Table 3.4-2, F&O CS-B1-1, the NRC staff finds that any changes to the PRA that might be caused by adjusting component boundaries will not cause the estimated transition risk decrease to become a risk increase.
SY-B8 Duke's PRA modeling guidelines (XSAA-115) include a walkdown checklist for documenting spatial dependencies for modeled equipment such as	Per Duke's PRA modeling guidelines, ensure that a walkdown/system engineer interview checklist is included	The NRC staff does not accept that the F&O is resolved solely by documentation. However, the

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
inadvertent sprinkler operation, missiles, high temperatures, flooding, fire, close proximity of equipment, and dependencies on HVAC that could significantly degrade the equipment. However, some of the system notebooks do not include this checklist.	in each system notebook. Based on the results of the system walkdown, summarize in the system write-up any possible spatial dependencies or environmental hazards that may impact system operation.	spatial dependencies related to fires clearly dominates the results of the Fire PRA and the potential impact of suppression activation was evaluated separately as described in Section 4.2.4 in the LAR. The NRC staff finds that spatial dependencies not associated with the fire or fire suppression is not likely, if identified, to cause the estimated transition risk decrease to become a risk increase.
<u>SY-B15</u> SSCs that may be required to operate in conditions beyond their environmental qualifications are not identified. Examples include: LOCA inside containment with failure of the RB cooling system would expose SG instrumentation to a harsh environment steam line breaks in the TB could expose equipment other than just the 4 kV switchgear and EFW control panel to an adverse environment; clogging of the RBES is not discussed and is not included in the system models.	Cut set review during applications should address this. Suggest adding this guidance to workplace procedure XSAA-103.	The NRC staff does not accept that the F&O is resolved solely by documentation. Further, the NRC staff does not accept that application cutset reviews resolve this issue. However, in response to RAI 5-9 (Reference 8) and 5-53 (Reference 9), the licensee described the possible impact of the environment in three example accident scenarios in the F&O for AS-B3. In the response, the licensee determined that the failure modes and/or loss of the identified equipment would have an insignificant impact on the PRA results. Based on the disposition

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Table 3.4-1, Internal Events PRA F&O Resolution

Facts and Observations (F&Os)	Licensee's Disposition	NRC Staff's Findings
		of the examples cited in the F&O for AS-B3, the NRC staff finds that this deficiency is not likely to cause the estimated transition risk decrease to become a risk increase.
SY-C2 The system notebooks contain much of the information listed in this SR. However, system model documentation should be enhanced to comply with all ASME PRA Standard requirements.	Enhance system model documentation to comply with all ASME PRA Standard requirements. In response to RAI 5-9j (Reference 8), the licensee noted that, prior to the advent of the ASME PRA Standard, the quality of the ONS PRA had been demonstrated to be adequate for supporting applications and clarified that the next version of the ONS PRA model (Revision 4) will have significantly enhanced documentation to bring it into compliance with the ASME PRA Standard supporting requirements. The licensee also noted that the reviewer did not identify any technical issues with the assessment.	The NRC staff finds that this F&O is made on an SR that identifies documentation requirements instead of technical requirements. Based on the assessment that this is a documentation issue only, the NRC staff finds that developing the appropriate documentation would not cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O CS-B1-1 Breaker coordination study incomplete (self identified). NRC staff F&O text: Plant personnel self-identified some issues with over current coordination during the inspection. The plant is working to resolve issues with molded-case circuit breaker instantaneous over current tripping coordination.	Breaker coordination issue not yet resolved [at the time of licensee's LAR submittal]; PIP O-08-2444 has been generated to track completion. The top 50% risk contributing scenarios involve loss of 4KV power and reliance on SSF mitigation which are not significantly impacted by additional failures due to improper breaker coordination.	In response to RAI 5-57, the licensee clarified that the draft breaker coordination study confirmed that coordination exists at higher voltage levels and that therefore lack of coordination at lower voltage levels are not expected to be significant. Subsequently, the licensee completed its breaker coordination study and therefore the licensee has appropriately resolved this F&O (Reference 12).
F&O CS-C4-1 Breaker coordination documentation. NRC staff F&O text: No specific documentation of overcurrent coordination was provided during the inspection.	This item is the documentation component of the breaker coordination issue discussed in CS-B1-1.	The licensee completed its breaker coordination study and therefore the licensee has appropriately resolved this F&O (Reference 12).
ES-B4 The NRC audit found that this supporting requirement was met at capability category I based on identifying the 4 Main Coolant Pump seal return isolation valves as additions to the equipment list based on fire-induced spurious operations leading to an ISLOCA [interfacing systems loss-of-coolant accident]..	The licensee evaluated this supporting requirement as meeting capability category III since no limit was placed on the number of fire-induced spurious operations in consideration of potential ISLOCA or containment bypass scenarios.	Based on the licensee statement that there was no limit to the number of spurious operations considered in the ISLOCA analysis, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
CS-A10 The NRC audit found that this supporting requirement was met at capability category I based on cable location information not being available in the ARTRAK database for "Y2" components and cable routing information not being available for "Y3" components.	The licensee evaluated this supporting requirement as meeting capability category III since cable location information is available in ARTRAK at the fire area, fire zone, and raceway level. While compliance is on an area basis, the Fire PRA scenarios included identification of targets (including raceways), where applicable. It is noted that credit by exclusion was only applied to low risk components (designated as 'Y3') for which cable routing information was not assembled.	In response to RAI 5-29, the licensee further clarified that (1) an evaluation of "Y1" and "Y2" components determined that these components more closely met capability category III than I and (2) the impact of the use of the "Y3" components on the Fire PRA quantification results was evaluated in a sensitivity analysis in the ONS NFPA 805 Fire PRA Application Calculation. Based on identifying cable locations at the raceway level, with the exception of low risk components, and the evaluation of low risk components in a sensitivity analysis, the NRC staff finds that this F&O has been resolved by the licensee for this application.
F&O FQ-A2-1 Initiating events not defined for all fire scenarios. NRC staff F&O text: Fire scenario frequencies are documented and reported based on plant analysis unit by unit and/or ignition source. The corresponding failed components are identified, but there is no identification of which initiating event is initiated by the fire.	Loss of Condenser Vacuum was the default initiator for the Fire PRA but some scenarios assumed a different initiator as discussed in the Fire PRA Model Development Report. The applied initiator has been added to information provided in the quantification results summary table (Fire PRA Summary Report, Appendix A).	Based on the identification of corrective actions being completed, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FQ-B1-1 Demonstrate convergence for the selected truncation limit (see IEPRA-2 for related FPIE [full power internal event] issue). NRC staff F&O text: Internal events QU-B3 is incorporated by reference and is not met: An iterative demonstration of convergence versus truncation level has not yet been performed.	Reference IEPRA-2 below for closure of FPIE truncation issue. The Fire PRA solves for conditional core damage probability (CCDP) (prior to application of ignition frequency) at one order of magnitude greater than the FPIE. Since a typical scenario frequency is typically much less than 0.1, there is not a truncation (convergence) issue for Fire PRA.	Based on the licensee's assessment that there is no truncation issue, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O FQ-C1-1 Use of nominal HEP values may result in loss of cutsets before application of recoveries/multipliers. NRC staff F&O text: Using nominal HEPs during quantification can result in cutsets being truncated. Rule Based Recovery will not correct this, since the cutsets are not present in the results.	To address the retention of additional cutsets, the HEP values were set to 0.1 for the initial solve prior to the application of recoveries (as described in the Fire PRA Model Development Report).	In response to RAI 5-58 (Reference 9), the applicant indicated that they use the larger of 0.1 or the nominal HEP for initial cutset generation, retaining hundreds of joint human errors in the cutsets. While using 1.0 as the screening HEP would increase the number of retained cutsets, the licensee did not expect this would introduce new important combinations, but merely burden the quantification effort unduly. Based on the licensee's conclusion that no new important human error combinations are expected, the NRC finds that reducing the screening HEP will not cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FQ-E1-1 Identification of significant contributors. NRC staff F&O text: The chosen method (see PRM-FQ-A) does not produce different significant contributor categories to support results review. As of audit review, limited review of available results had been performed.	Risk insights from the risk significant fire initiating events with identification of significant contributors have been included in the Fire PRA Application Calculation.	Based on the identification of corrective actions being completed, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application.
F&O FQ-F1-1 Improve LERF documentation. NRC staff F&O text: It is not possible to see the initiating event assigned to each scenario unless one looks in the cutset output files and deduces the initiating event based on the failed equipment. Method used by the licensee can produce the correct numerical results without meeting the standard.	Documentation concerns were largely confined to LERF. Accordingly, the insights section in the Fire PRA Summary Report was expanded to address LERF. Inconsistencies between LERF and CDF have been reconciled. Additionally, the risk insights section of the Fire PRA Application Calculation and the insights section in the Fire PRA Summary Report have been expanded to address LERF. Also, the applied initiator has been added to information provided in the quantification results summary as discussed in response to FQ-A2-1.	Based on the identification of corrective actions being completed, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-A5-1 Horizontal propagation for cables with PVC jackets (self identified). NRC staff F&O text: Horizontal propagation outside zone of influence (ZOI) not done yet, as the effects of PVC jacket (typically TP [thermoplastic] on horizontal spread is an open item self identified by the licensee.	PVC jacketing impact on horizontal fire spread is discussed in the Fire Scenario Report. While armored cables are generally considered to be noncombustible (refer to NUREG/CR-6850 Section R.4.1.4), the armored cables at ONS have a PVC jacketing. The justification concluded that the PVC coating will not sustain propagation of fire along the armored cable for a significant distance; any horizontal fire propagation along cable trays is adequately captured within the target set of each scenario involving overhead tray failures.	The NRC staff finds that the licensee's evaluation that targets associated with horizontal fire propagation along armored cables having PVC coating has not been completely addressed. However, it is unlikely that this deficiency would cause the estimated transition risk decrease to become a risk increase for this application.
FSS-C2 The NRC audit found that this supporting requirement was met at capability category I based on the peak heat release rate being assumed at t=0 when establishing the HGL threshold and for development of scenarios for individual ignition sources.	The licensee dispositions capability category I acceptable for the application given that the results are conservative and that no changes to conclusions are anticipated if time dependent fire growth profiles are assumed given the conservatism inherent in the NUREG/CR-6850 maximum heat release rates and available fire growth profiles relative to the ignition frequencies.	Based on the use of conservative assumptions for HRR for the initial ZOI and the evaluation of possible failure of manual suppression, the NRC staff finds that this observation has been investigated and addressed for this application appropriately because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-C5-1 Potential for PVC pooling may impact assumed damage threshold. NRC staff F&O text: For those cases with potentially pooling thermoplastic (TP), TP characteristics for failure should be attributed to ONS cable.	PVC pooling is addressed in the Fire Scenario Report. The justification centers on the expectation that cables would tend to have the PVC jacket melt and flow away creating voids for the flow of melting materials from other cables. In addition, the ridges that are characteristic of the armor jacketing provide additional free space for the flow of material. As such, it is not expected that pooling of PVC within cable trays is likely to occur even if multiple layers of armored cables exist.	The NRC staff finds that the licensee's evaluation that PVC pooling within cable trays is not expected to occur has not been completely addressed, but any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O FSS-D5-1 Justify use of 75% HRR for transient fires. NRC staff F&O text: The licensee has provided a weak basis for applying only the 75% fire HRR for transient combustibles, excluding the larger HRR.	Justification in the Fire Scenario Report is expanded. Use of the 75% HRR for transients is considered realistic treatment (more characteristic of actual transient fire scenarios identified in the fire events database) and appropriate for PRA application. Use of a trash bag as the transient fuel package provides a bounding characterization of the behavior of observed transient fire ignition sources, but ignores the ignition element. Consequently, while administrative controls are factored into the development of transient ignition frequencies for each compartment, the actual transient combustible loading that may be allowed or present does not directly impact the numerical results of the Fire PRA.	Based on the information provided by the licensee, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O FSS-D6-1 Justify fire brigade response time with respect to formation of HGL. NRC staff F&O text: A probability should be	The licensee developed support for fire brigade response time of 20 minutes from the review of actual fire brigade drill performance, which is included in the Fire Scenario Report. Specifically, the applicant reviewed 100 plant	The topic identified by the licensee does not completely address the NRC staff's finding that a manual suppression failure probability should be developed for the fire brigade intervening to prevent damage from the

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
developed for manual suppression for the fire brigade intervening to prevent damage from the HGL.	drills, in which the time exceeded 20 minutes only once, and only slightly in that one instance. In lieu of a sensitivity analysis, the applicant bounded the effect of delayed brigade response by assuming the more conservative phenomenology for damage due to ZOI effects vs. HGL formation, showing that the compartments screened for the former and, therefore, bounded the latter. The applicant further cited an inherent conservatism that, while the brigade might not initiate suppression activities within 20 minutes, credit has not been taken for the brigade taking the simple action to open a door to prevent HGL formation in a shorter time.	HGL, or upward propagation through successive cable trays. In response to RAI 5-27 (Reference 9), the license summarized the site experience used to support the 20 minute response time. Based on the licensee's documentation supporting its assumed brigade response time the NRC staff finds that a high likelihood of response within 20 minutes has been established. However, the assumptions used by the licensee in the ZOI determination, while conservative in the HRR attributed to the ignition source, are non-conservative in that the fire never propagates by igniting additional combustible material (and thereby increasing the HRR) beyond the original ignition source.. The applications of conservative initial ZOI with the non-conservative assumption that the fire never propagates to combustibles beyond the original ignition source yields an indeterminate result. However, it is unlikely that deficiency would cause the estimated transition risk decrease for this application.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
FSS-D9 The NRC audit found that this supporting requirement was met at capability category I based on smoke damage to Fire PRA equipment not being considered.	The licensee evaluated this supporting requirement to meet capability category II/III based on the potential for smoke damage to equipment not already failed by fire affects being added to the Fire Scenario Report.	Based on the licensee evaluating the potential for smoke damage to Fire PRA equipment when identifying targets, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
FSS-E3 The NRC audit found that this supporting requirement was met at capability category I based on the uncertainty analyses presented in the "Generic Fire Modeling Treatments" document.	The licensee dispositions capability category I acceptable for the application given that parameters for modeling fire scenarios (such as heat release rates and severity factors) are taken from NUREG/CR-6850, which is the consensus method for Fire PRA development.	Based on the use of fire modeling parameters from NUREG/CR-6850, the NRC staff finds that capability category I is acceptable for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
FSS-F2 The NRC audit found that this supporting requirement was met at capability category I based on no criterion being established or justified for structural collapse due to exposure of structural steel to a fire.	The licensee dispositions this supporting requirement as not applicable since no scenarios were selected for quantification (structural steel damage required no further quantitative treatment).	The NRC staff finds that the justification that FSS-F2 is not applicable to ONS is acceptable.
FSS-F3 The NRC audit found that this supporting requirement was met at capability category I based on the qualitative assessment that the MFW oil fire scenario bounds the CCDP of a structural collapse of the TB.	The licensee dispositions this supporting requirement as not applicable since no scenarios were selected for quantification (structural steel damage required no further quantitative treatment).	The NRC staff finds that the justification that FSS-F3 is not applicable to ONS is acceptable.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-G1-1 Multi-compartment analysis incomplete. NRC staff F&O text: Multi-compartment analysis does not include a range of potential multi-compartment fire scenarios.	Addressed via multi-compartment screening analysis added as Attachment D to the Fire Scenario Report.	Based on the licensee's multi-compartment evaluation to identify targets in adjacent compartments that are within the zone of influence for a given fire scenario and including these targets in the set of equipment that are damaged by the fire, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O FSS-G2-1 Multi-compartment analysis screening criteria not defined. NRC staff F&O text: No screening criteria for multi-compartment fires have been defined.	Screening criteria added to multi-compartment discussion in the Fire Scenario Report.	See NRC staff finding on F&O FSS-G1-1.
F&O FSS-G3-1 Multi-compartment analysis screening incomplete; no MCA scenarios defined for quantification. NRC staff F&O text: The analysis has not screened potential multi-compartment combinations of interest nor defined any multi-compartment fire scenarios beyond those that are inherently captured in the treatment of fire scenarios for the TB fire zones (see PP).	Addressed via multi-compartment screening analysis added as Attachment D to the Fire Scenario Report. The screening criteria were applied to all analyzed compartments with the end result being that all compartments (physical analysis units) screened.	See NRC staff finding on F&O FSS-G1-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-G4-1 Multi-compartment analysis did not consider potential for barrier failure. NRC staff F&O text: Multi-compartment analysis is incomplete and has not included an assessment of credits given to passive fire barrier features.	Potential for fire barrier failure addressed via multi-compartment screening analysis added as Attachment D to the Fire Scenario Report.	See NRC staff finding on F&O FSS-G1-1.
F&O FSS-G5-1 Multi-compartment analysis did not assess active fire barriers. NRC staff F&O text: Active fire barriers have been credited in partitioning but not assessed per this SR.	No active fire barriers have been credited in the Fire PRA for limiting the zone of influence. The active fire barrier between BH12 and CT4 is only credited for the deterministic fire area boundary definition.	Based on the licensee's statement that no active fire barriers are being credited in the Fire PRA, the NRC staff finds that this F&O has been appropriately resolved by the licensee for this application.
F&O FSS-G6-1 Assessment of multi-compartment analysis scenarios relative to fire risk not performed. NRC staff F&O text: No assessment of a range of potential multi-compartment scenarios has been provided.	Addressed via multi-compartment screening analysis added as Attachment D to the Fire Scenario Report. No additional scenarios were identified for quantification.	See NRC staff finding on F&O FSS-G1-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-H2-1 document resolution of PVC pooling issue. NRC staff F&O text: Treatment of PVC jacket on cable failure is not addressed and is an open item. The cable jacket affects Oconee conclusion that TS failure criteria should be used. Oconee is collecting information regarding the PVC jacket to establish the nature of the PVC; however, PVC is typically thermoplastic material.	Addressed in Fire Scenario report: while the PVC jacket is thermoplastic, the cable insulation within the armor is consistent with thermoset (flame retardant cross-linked polyethylene). See FSS-C5-1.	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has not been completely addressed, but any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
FSS-H6 The NRC audit found that this supporting requirement was met at capability category I based on the lack of documentation supporting the contention that using the area ratio is adequate and justifying the use of the 75th percentile of the NUREG/CR-6850 heat release rate for transient combustibles.	The licensee evaluated this supporting requirement to meet capability category II/III based on the use of conservative scoping fire modeling criteria from NUREG/CR-6850 and that, other than the Bayesian update of fire frequencies and consideration of actual fire brigade response times, no plant specific updates were applied.	Based on the Bayesian updating of fire frequencies and the consideration of ONS-specific fire brigade response times, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application because any changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O FSS-H8-1 Multi-compartment analysis documentation incomplete. NRC staff F&O text: The multi-compartment analysis remains incomplete (see FSS-F and its SRs); hence, documentation is also incomplete.	Addressed via supplemental discussion and multi compartment screening analysis added as Attachment D to Fire Scenario Report.	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O FSS-H9-1 Document justification for fire brigade response time. There NRC staff F&O text: are no uncertainties listed for manual fire brigade suppression which limits the development of the hot gas layer as given in the document Oconee FPRA 031408 Tasks 8 and 11 Scenario Development Attachment A, Scenario Summary Report.	Support for fire brigade response time of 20 minutes based on review of actual fire brigade drill performance has been added to Fire Scenario Report. Specifically, the applicant reviewed 100 plant drills, in which the time exceeded 20 minutes only once, and only slightly in that one instance. In lieu of a sensitivity analysis, the applicant bounded the effect of delayed brigade response by assuming the more conservative phenomenology for damage due to ZOI effects vs. HGL formation, showing that the compartments screened for the former and, therefore, bounded the latter. The applicant further cited an inherent conservatism that, while the brigade might not initiate suppression activities within 20 minutes, credit has not been taken for the brigade taking the simple action to open a door to prevent HGL formation in a shorter time.	See NRC staff finding on F&O FSS-D6-1.
F&O HR-G7-1 (Internal Events PRA) Dependencies should be reviewed with respect to timing.	Corrective Action 2 of PIP O-08-2915 dispositions issue as minimally conservative (compared to other sources of conservatism in Fire PRA) based on the degree of dependency between actions with different available response times.	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase. The internal events HRA methodology is addressed in the NRC staff finding HR-G3.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O HRA-A1-1 SR considered met but documentation that SSD actions carried over from FPIE model remain valid for Fire PRA was not provided. NRC staff F&O text: No documentation that for each fire scenario, for each SSD action carried over from the Internal Events PRA, each action remains valid in the context of the Fire PRA.	Discussion relative to reliance on EOP[emergency operating procedure], abnormal operating procedures (AOPs), and alarm response procedures given a fire has been added to Fire PRA Model Development report (see HRA-E1-1).	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O HRA-B2-1 HRA documentation for CASWHPIDHE and CEF0ASWDHE insufficient. NRC staff F&O text: For events CASWHPIDHE and CEF0ASWDHE the definition of the HFEs is not as detailed as that for the other HFEs.	No impact on quantification; Corrective Action 3 of PIP O-08-2915 indicates that documentation deficiency will be addressed with issuance of future Revision 4 of the ONS PRA Model.	See NRC staff's finding on F&O HRA-A1-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O HRA-C1-1 Need to consider relative timing of HFE in fire scenario; time from cue versus time from fire. NRC staff F&O text: The approach to the quantification of the HEPs is to revise the internal events HEPs using a set of rules revising the HEPs based on, among other things, allowable action time. The basis, or the set of assumptions upon which this set of rules is based is not provided. This does not seem to have been applied correctly. F&O HRA-G7-1. NRC staff F&O text: In reviewing the documentation for ZHFC-2-058 there is evidence that there is a lack of appreciation of the relative timing of events. The comment in the documentation on relative timing focuses on cognitive response time (2 minutes and 15 minutes for the two events (NSFORCMDHE and CASWHPIDHE) respectively. However, these two events are separated in time by a significant time, the first event being required at 30 minutes, the second at four hours respectively. The dependency evaluations should be reviewed carefully for the internal events model and the fire model.	Risk significant HFE's revisited with respect to timing of cues; documentation provided in Fire PRA Model Development Report.	See NRC staff's finding on F&O HRA-A1-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O HRA-C1-2 Post-initiator HEP quantifications need to be checked for consistency. NRC staff F&O text: (internal events SR referenced by HRA-C1) There is no evidence that the fire related post-initiator HEP quantifications have been checked for consistency.	Top 6 risk significant operator actions in the Fire PRA were checked for consistent application of criteria (Corrective Action 4 of PIP O-08-2915). The following operator actions were reviewed: TTRHPITDHE was compared to CHPHPMUDHE, HHPHPR0DHE was compared to LLPLPR0DHE. And finally, FEFEFW2DHE was compared to FEFEFW1DHE. The basic events in each pair were similar in characteristics and they were accordingly mapped to the same HEP adjustment case (i.e. Case 3 or 4 etc.).	See NRC staff's finding on F&O HRA-A1-1.
F&O HRA-E1-1 Address how alarm response and EOP/AOPs are followed given a fire. NRC staff F&O text: There is no documentation to describe the procedures and their use during a fire scenario. There is no documentation of the assumptions underlying the screening approach. There is no justification that the timing associated with the analyzed HFEs is appropriate for the accident scenarios.	Discussion relative to reliance on EOPs, AOPs, and alarm response procedures given a fire has been added to Fire PRA Model Development report to support the conclusion that credited Fire PRA actions are consistent with the expected plant response to a fire event including the decision to man the SSF.	See NRC staff's finding on F&O HRA-A1-1.
F&O IEPRA-1 Resolve issues from gap assessment of the ONS PRA Revision 3a. NRC staff F&O text: Document resolution of the issues identified in the Maracor review; justify any exceptions.	Appendix D to PRA Quality Self Assessment addresses ONS PRA technical adequacy for NFPA 805. Open SRs with potential impact on quantification of delta risk for change evaluations have been addressed in sensitivity analysis within NFPA 805 Fire PRA Application Calculation.	Based on the NRC staff's conclusions about the technical adequacy of the ONS PRA Revision 3a discussed in this SE, the NRC concludes that resolving the remaining issues from the gap assessment on the internal event PRA will not cause the estimated transition risk decrease to become a risk increase.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O IEPRA-2 Demonstrate convergence for selected truncation limit (FPIE issue). NRC staff F&O text: Need to perform updates credited in "Oconee Responses" to the 2006 Maracor review. See also F&O PRM-B1-1.	OSC-8863 demonstrates that the ONS PRA CDF results converge at 1E-09.	The NRC staff F&O addressed many updates credited in the "Oconee Responses," not simply a convergence issue as identified in the ONS Topic description. For disposition of the part of this F&O related to general incorporation of F&Os related to the Maracor review of the internal events PRA see the NRC staff's conclusions in Attachment C, Table 3.4-1 in this SE. The NRC staff finds that the issue of convergence has been appropriately resolved for this application.
F&O IGN-A5-1 Use of reactor year/critical year. NRC staff F&O text: Need to update ignition frequency data once NUREG/CR-6850 is updated with the correct numbers based on reactor-year basis.	To be addressed when NUREG/CR-6850 is updated with the correct numbers based on reactor-year basis; impact is expected to be insignificant. Note that Interim EPRI Report 1019189, which was not used at ONS, would lower ignition frequencies for most bins. Also Bayesian update to 'lower' ignition frequencies was not applied at ONS (only 1 bin was increased). Both measures, if applied, would offset expected increase.	Based on the use of acceptable ignition frequency data from NUREG/CR-6850, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O MUD-B4-1 Procedure lacks reference to PRA combined standard (draft). NRC staff F&O text: Fire standard needs to be referenced in XSAA-106 and the Fire PRA model should be explicitly in the scope of the procedure.	Per Corrective Action 5 of PIP O-08-2915, PRA Workplace Procedure XSAA-106 was revised to reference the Combined PRA standard	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O MUD-E1-1 Qualify the FRANC computer code for use on Fire PRA. NRC staff F&O text: The FRANC computer code and corresponding Microsoft Access databases have not been evaluated and documented at any software and data quality assurance (SDQA) classification per NSD-800.	Per Corrective Action 6 of PIP O-08-2915, the FRANC computer code was qualified as documented in SDQA-30271-NGO.	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.
F&O PP-B2-1 Justification for credit of nonrated partition boundaries insufficient. NRC staff F&O text: The FPRA credited partitioning elements that lacked fire resistance rating.	Partially addressed via multi-compartment analysis; failure to meet SR poses no adverse impact on the analysis quality or completeness. Deviation from "enclosed boundary" definition applied to compartment frequency calculation which has no impact on overall CDF/LERF results. Use of zone of influence for defining extent of fire scenario regardless of location of zone boundary ensures that scenario impacts are accurate.	See NRC staff's finding on F&O FSS-G1-1.
F&O PP-B3-1 Use of open fire zone boundaries implies credit for spatial separation. NRC staff F&O text: Some TB Fire Compartments have boundaries that do not correspond to a physical wall, and thus have no fire rating. The use of these boundaries implies crediting spatial separation; therefore the standard is not met at Category 1.	Partially addressed via multi-compartment analysis; failure to meet SR poses no adverse impact on the analysis quality or completeness (see disposition for PP-B2-1).	See NRC staff's finding on F&O FSS-G1-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
PP-B5 The NRC audit found that this supporting requirement was met at capability category I based on not crediting active fire barrier elements as a partitioning feature for the roll-up fire door between Block House 1&2 and CT-4 or fire door closure devices in the SSF.	The licensee dispositions capability category I acceptable for the application given that the results are scenario driven (the zone of influence was not arbitrarily limited to the zone boundary). Other than the rollup door between BH12 and CT4 which was credited for deterministic fire area partitioning, active fire barrier elements were not credited.	See NRC staff's finding on F&O FSS-G1-1.
F&O PP-C3-1 Improve general description and identification of unique fire protection features. NRC staff F&O text: The lack of rated barriers was documented in OSC-8979:	Partially addressed via multi-compartment analysis (see PP-B2-1); failure to meet SR poses no adverse impact on the analysis quality or completeness.	See NRC staff's finding on F&O FSS-G1-1.
F&O PRM-B1-1 Impact of the internal event PRA peer review open items on Fire PRA not addressed. NRC staff F&O text: Oconee used a version of the internal events model with a substantial number of outstanding issues (see F&O IEPRA-1) as the base model. The finding is based on the fact that the issues identified in the peer review have not been resolved.	See IEPRA-1.	See NRC staff's finding on F&O IEPRA-1.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O PRM-D1-1 Circa 2005 fire structure in the PRAE model not peer reviewed (SR PRM-D1 deleted and is now PRM-C1). NRC staff F&O text: Final documentation should identify and clearly describe all fire-related changes made to the internal event PRA in one document to meet the standard. The current documentation provides an incomplete and sometimes contradictory description of proposed changes versus real changes.	Eliminated reliance on initiators %TB0FIRE and %CSFIRE from pre-existing fire structure in Fire PRA.	The licensee's response did not completely address the NRC staff's observation that the final documentation needs to clearly describe all fire related changes made to the internal events PRA. However, based on the NRC staff's review of the Fire PRA and the licensee's response to the NRC staff's review, the NRC staff finds the licensee has appropriately resolved this issue for this application, because any additional changes are not likely to cause the estimated transition risk decrease to become a risk increase.
F&O QLS-A3-1 Discussion in Partitioning & Ignition Frequency calc implied that actions pertaining to 4 structures that were excluded from ignition source counting had not been completed. NRC staff F&O text: 22 buildings structures screened, 4 left unresolved.	No impact on quantification; updated calculation to provide justification for exclusion for the 4 structures to reflect that no further action was necessary for these structures based on the multi-compartment analysis.	Based on the identification of corrective actions being completed, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.
QNS-C1 The NRC audit found that this supporting requirement was met at capability category I based on no evaluation having been performed to demonstrate that capability category II or III was met.	The licensee dispositions this supporting requirement as not applicable since screening criteria was not applied; if a building or structure (or an area in the case of the switchyard) contained PRA credited equipment/cables and/or could result in loss of offsite power, it was not screened.	The NRC staff finds that the justification that QNS-C1 is not applicable to ONS is acceptable.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O SF-A2-1 Conduct assessment of the potential for diversion of suppression flow. NRC staff F&O text: A seismic induced assessment of the potential for diversion of suppressants from areas where it is needed for fire suppression systems associated with a common suppressant supply was not conducted.	No impact on quantification of Fire PRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). See qualitative discussion of seismic fires in the Unit 3 Fire PRA Summary report. PIP G-09-00698 will track the resolution of this open item.	In response to RAI 5-28, the licensee further clarified that ONS complies with the requirements of NFPA 805 Section 3.6.4, "Standpipe and Hose Stations Earthquake Provisions," thus satisfying this supporting requirement. Since this SR requires only a qualitative assessment and will not impact quantification of the Fire PRA, and based on the licensee's evaluation that this supporting requirement is already met, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.
F&O SF-A4-1 Plant seismic response procedures do not cover seismically induced fire. NRC staff F&O text: The plant seismic response procedures cover seismically induced flooding, but not seismically induced fires.	No impact on quantification of Fire PRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). PIP G-09-00698 will track the resolution of this open item.	In response to RAI 5-28, the licensee further clarified that the fire response procedure is entered either via a fire alarm annunciator or the report of a fire, either of which applies at all times and under any plant operating conditions and that, therefore, a reference to the fire responses procedure in the seismic response procedure is unnecessary. Since this SR requires only a qualitative assessment and will not impact quantification of the Fire PRA, and based on the licensee's evaluation that existing ONS procedures adequately address response to seismically induced fires, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.

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Table 3.4-2, Fire Events PRA F&O Resolution

Facts and Observations	Licensee's Disposition	NRC Staff's Findings
F&O SF-A5-1 Assessment of earthquake impact on fire brigade not documented NRC staff F&O text: No assessment has been conducted on the potential that an earthquake might compromise one or more of the fire brigade members.	No impact on quantification of Fire PRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). PIP G-09-00698 will track the resolution of this open item.	In response to RAI 5-28, the licensee further noted that fire brigade response during a seismic event has been considered in ONS Standard Operating Guide (SOG) #1 in that ONS has staged fire brigade equipment so that one single event will not render the fire brigade ineffective. Since this SR requires only a qualitative assessment and will not impact quantification of the Fire PRA, and based on the licensee's evaluation that existing ONS procedures adequately address fire brigade response to seismically induced fires, the NRC staff concludes that this deficiency has been appropriately resolved by the licensee for this application.
F&O UNC-A1-1 Uncertainty and sensitivity analysis incomplete (not reviewed). NRC staff F&O text: Not reviewed. When the analysis is complete and stable, the sources of model uncertainty should be identified.	Uncertainty & Sensitivity Matrix added as Appendix D to Fire PRA Summary Report; sensitivity quantitatively addressed in NFPA 805 Fire PRA Application Calculation.	Based on the NRC staff's review of the Fire PRA and the licensee's response to the NRC staff's review, the NRC staff finds that it is unlikely that a completed uncertainty and sensitivity analysis will cause the estimated risk decrease to become a risk increase.

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Table 3.4-3, V&V Basis for Fire Modeling Correlations

Correlation	Application at ONS	V&V Basis	NRC Staff Evaluation of Acceptability
Flame Height	Provide a limit on the use of the ZOI.	NUREG-1805 NUREG-1824	• The licensee stated that the flame height correlation is used in both the Consolidated Fire and Smoke Transport Model (CFAST) and NUREG-1805 fire models, for which V&V was documented in NUREG-1824. • The licensee stated that use of the correlation was limited to its range of applicability. Since the (1) V&V basis is NUREG-1824 and (2) the licensee stated that the correlation was applied within the limits of its applicability, the NRC staff finds use of this correlation in the ONS application acceptable.
Radiant Heat Flux Method of Shokri and Beyler (detailed)	Calculates target heat flux to determine the lateral extent of the ZOI.	NUREG-1805 NUREG-1824 SFPE Engineering Guide,1999	• The licensee stated that this correlation produces the most conservative results of the four correlations considered in the calculation, while maintaining a credible radiated energy fraction from the flame shapes postulated. • The licensee stated that the correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824, and the V&V basis for the correlation is documented in an authoritative publication of the SFPE Engineering Guide. • The licensee stated that the correlation was used within the limits of its range of applicability. Since (1) the correlation produces conservative results, (2) the V&V basis is NUREG-1824 and an authoritative publication of the SFPE Engineering Guide, and (3) the licensee stated that the correlation was applied within the limits of its applicability, the NRC staff finds use of this correlation in the ONS application acceptable.

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Table 3.4-3, V&V Basis for Fire Modeling Correlations

Correlation	Application at ONS	V&V Basis	NRC Staff Evaluation of Acceptability
Plume Heat Fluxes	Calculates the vertical separation distance to the target in order to determine the vertical extent of the ZOI.	Wakamatsu et al., 2003 SFPE Handbook 4th Edition, Chap. 2-14, Lattimer, B., 2008	• The licensee stated that the ZOI vertical separation distance used in the ONS application is based on the more severe result from this calculation and the calculation for plume centerline temperature (see below). • The licensee stated that the correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering and in a published conference report. • The licensee stated that the correlation was used within the limits of its range of applicability. Since (1) the application uses the more conservative of the two correlations for determining the vertical extent of the ZOI, (2) the V&V basis is an authoritative publication of the SFPE Handbook of Fire Protection Engineering and a published article in a conference report, and (3) the licensee stated that the correlation was applied within the limits of its applicability, the NRC staff finds use of this correlation in the ONS application acceptable.
Plume Centerline Temperature	Calculates the vertical separation distance to the target to determine the vertical extent of the ZOI.	NUREG-1805 NUREG-1824 SFPE Handbook 4th Edition, Chap. 2-1, Heskestad, G., 2008	• The licensee stated that the ZOI vertical separation distance used in the ONS application is based on the more severe result from this calculation and the calculation for Plume Heat Fluxes (see above). • The licensee stated that the correlation is used in the NUREG-1805 fire model, for which V&V was documented in NUREG-1824 and the V&V basis for the correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The licensee stated that the correlation was used within the

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Table 3.4-3, V&V Basis for Fire Modeling Correlations

Correlation	Application at ONS	V&V Basis	NRC Staff Evaluation of Acceptability
			limits of its range of applicability. Since (1) the application uses the more conservative of the two correlations for determining the vertical extent of the ZOI, (2) the V&V basis is NUREG-1824 and an authoritative publication of the SFPE Handbook of Fire Protection Engineering, and (3) the licensee stated that the correlation was applied within the limits of its applicability, the NRC staff finds use of this correlation in the ONS application acceptable.
Hydrocarbon Spill Fire Size	Determines the heat release rate for unconfined hydrocarbon spill fires.	SFPE Handbook 3rd Edition, Chap. 3-11, Beyler, C., 2002	• The licensee stated that the correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The licensee stated that there were no limits in treatment of range for this correlation because the spill transition from unconfined to deep pool burning would be abrupt. Since the V&V basis is an authoritative publication of the SFPE Handbook of Fire Protection Engineering, the NRC staff finds use of this correlation in the ONS application acceptable.
Flame Extension	Determines the fire offset for open panel fires; only used when the possibility of flame extensions are present.	SFPE Handbook 3rd Edition, Chap. 3-11, Beyler, C., 2002	• The licensee stated that the correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The licensee stated that the correlation applies to fires ranging in size from about 10 kW to 1.0 MW, which bounds the fire size bins for open electronic equipment cabinets defined in NUREG/CR-6850. Since the V&V basis is an authoritative publication of the SFPE Handbook of Fire Protection Engineering, and since the applicable fire

Attachment C, Fire Risk Evaluation Tables

Table 3.4-3, V&V Basis for Fire Modeling Correlations

Correlation	Application at ONS	V&V Basis	NRC Staff Evaluation of Acceptability
			size range bounds the fire size bins for open electronic equipment cabinets defined in NUREG/CR-6850 (Reference 37), the NRC staff finds use of this correlation in the ONS application acceptable.
Corner Flame Height	Determines the heat release rate for fires that are proximate to a corner to determine the vertical extent of the ZOI.	SFPE Handbook 3rd Edition, Chap. 2-14, Lattimer, B., 2002	• The licensee stated that this correlation is used to ensure that a conservative separation distance is calculated when the fire dynamics have limited entrainment conditions. • The licensee stated that the correlation is documented in an authoritative publication of the SFPE Handbook of Fire Protection Engineering. • The licensee stated that the correlation applies to corner fires ranging in size from about 10 kW to 1.0 MW, which bounds the fire size bins for open electronic equipment cabinets defined in NUREG/CR-6850. Since (1) the correlation produces conservative results, (2) the V&V basis is an authoritative publication of the SFPE Handbook of Fire Protection Engineering, and (3) since the applicable fire size range bounds the fire size bins for open electronic equipment cabinets defined in NUREG/CR-6850 (Reference 37), the NRC staff finds use of this correlation in the ONS application acceptable..
Line Fire Plume Centerline Temperature	Calculates simplified separation distances to the target to provide a limit on the use of the ZOI and the extent of the ZOI	Yuan et al.,1996 (Peer-reviewed journal experimental data)	• The licensee stated that the correlation is documented in a peer reviewed journal article. • The licensee stated that the approach bounds the methods for predicting the plume centerline temperature above the line type source fire.

Attachment C, Fire Risk Evaluation Tables

Table 3.4-3, V&V Basis for Fire Modeling Correlations

Correlation	Application at ONS	V&V Basis	NRC Staff Evaluation of Acceptability
	for cable tray fires.		Since the V&V basis is a peer-reviewed journal article, and the approach bounds the methods for predicting the plume centerline temperature above the line type source fire, the NRC staff finds use of this correlation in the ONS application acceptable.
Ventilation Limited Fire Size	Assesses the significance of vent position on the hot gas layer temperature to determine the fire size due to ventilation limitations.	SFPE Engineering Guide, 2004	• The licensee stated that this correlation was used to develop generic rules for various ventilation opening sizes while ensuring bounding cases are applied. • The licensee stated that the correlation is documented in an authoritative publication of the SFPE Engineering Guide. • The licensee stated that there were no limits in treatment of range for this correlation because the most severe environment predicted was the most conservative for the given room volumes and ventilation opening areas. Since (1) the methodology ensures use of bounding results, (2) the V&V basis is an authoritative publication of the SFPE Engineering Guide, and (3) the correlation was only used for fire scenarios where it was applicable. The NRC staff finds use of this correlation in the ONS application acceptable.

References for Table 3.4-3

1. NUREG-1805, "Fire Dynamics Tools (FDT^s) Quantitative Fire Hazard Analysis Methods for the U.S. Nuclear Regulatory Commission Fire Protection Inspection Program," U.S. NRC, Washington, DC, December 2004.

2. NUREG-1824, "Verification & Validation of Selected Fire Models for Nuclear Power Plant Applications", U.S. NRC, Washington, DC, May, 2007.

3. The SFPE Engineering Guide, "Assessing Flame Radiation to External Targets from Pool Fires," Society of Fire Protection Engineers, Bethesda, Maryland, June, 1999.

4. 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.

5. Lattimer, B. Y., "Heat Fluxes from Fires to Surfaces," Chapter 2–14, The SFPE Handbook of Fire Protection Engineering, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.

6. Beyler, C.L., "Fire Hazard Calculations for Large, Open Hydrocarbon Fire," Chapter 3-11, The SFPE Handbook of Fire Protection Engineering, 3rd Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2002.

7. Heskestad, G., "Fire Plumes, Flame Height, and Air Entrainment," Chapter 2–1, The SFPE Handbook of Fire Protection Engineering, 4th Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2008.

8. Lattimer, B. Y., "Heat Fluxes from Fires to Surfaces," Chapter 2–14, The SFPE Handbook of Fire Protection Engineering, 3rd Edition, P. J. DiNenno, Editor-in-Chief, National Fire Protection Association, Quincy, MA, 2002.

9. Yuan, L. and Cox, F., "An Experimental Study of Some Line Fires," *Fire Safety Journal*," Volume 27, 1996.

10. The SFPE Engineering Guide, "Fire Exposures to Structural Elements," Society of Fire Protection Engineers, Bethesda, Maryland, May 2004.

11. Kleinsorg Group Risk Services, LLC, "Generic Fire Modeling Treatments," Revision 0, January 23, 2008, prepared for ERIN Engineering and Research, Inc.

Attachment D, Nuclear Safety Capability Assessment Results by Fire Area

Attachment D is broken down into those ONS fire areas that were analyzed using the deterministic approach in accordance with NFPA 805, Section 4.2.3, and those using the PB approach in accordance with NFPA 805, Section 4.2.4.

Each fire area includes a discussion of how the licensee met the NFPA 805 requirement to evaluate the potential fire suppression effects on the ability to meet the nuclear safety performance criteria.

Each fire area also contains a section that addresses those NRC approved exemptions and other licensing actions that exempt the licensee from the existing deterministic fire protection licensing basis that the licensee desires to incorporate into the RI/PB FPP, as allowed by NFPA 805, Section 2.2.7. This discussion for each applicable fire area includes a description of the previously approved exemption or other licensing action exempting the licensee from the deterministic requirements, the basis for and continuing validity of the exemption or other licensing action, and the NRC staff's evaluation of that exemption or other licensing action.

Where required, each section of the attachment includes an evaluation of the DID recovery actions necessary for the applicable fire area. As discussed in SE Section 3.2.4, the licensee credited recovery actions to satisfy the DID requirements of NFPA 805, Section 1.2, but are not needed to maintain the availability of a success path and do not adversely impact risk. Because the licensee has identified these recovery actions as being necessary to provide adequate DID, the NRC staff has evaluated them as a part of the RI/PB FPP. As such, future removal of these DID recovery actions would require a plant change evaluation in accordance with NFPA 805, Section 2.4.4.

For all fire areas where the licensee utilized the PB approach to meet the nuclear safety performance criteria, each VFDR and the associated disposition has been listed.

As a part of the NSCA, the licensee evaluated fire detection and suppression systems on a fire area basis. Accordingly, the evaluation of each fire area includes a table that documents the licensee's review of these fire detection and suppression systems, as well as the NRC staff's evaluation of the review and its results.

Finally, each fire area includes a summary assessment documenting the NRC staff's conclusion regarding the ability to meet the NFPA 805 requirements and the associated nuclear safety performance criteria.

Attachment D1, Deterministic Compliance with NFPA 805 Section 4.2.3

For each fire area where the licensee selected the deterministic approach to demonstrate compliance with the NFPA 805 requirements, the NRC staff verified that the deterministic requirements of NFPA 805, Section 4.2.3, are met without the use of recovery actions. Fire areas that meet the deterministic requirements of NFPA 805 are deemed to adequately satisfy the nuclear safety performance criteria, as stated in NFPA 805, Section 1.5.1.

The licensee evaluated suppression and detection systems using a process that looked at several key aspects of the FPP to determine if a given system is required (i.e., deterministically in support of compliance with NFPA 805 Chapter 4, in support of a previous NRC approved alternative or in support of a licensee-developed EEEE).

Accordingly, each of the fire areas listed below include a section discussing those fire suppression and fire detection systems the licensee has determined to be required to meet the nuclear safety performance criteria.

Fire Area CT-4, CT-4 Block House

The licensee stated that deterministic compliance has been met in accordance with NFPA 805 Section 4.2.3.2, without the use of recovery actions, which requires that one success path of required cables and equipment be located in a separate area having boundaries containing fire barriers with a minimum fire resistance rating of three hours. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on the Nuclear Safety Performance Criteria

The licensee stated in LAR Attachment C, "NEI 04-02 Table B-3 Fire Area Transition," those safe and stable conditions can be achieved and maintained using equipment and cables outside the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Variation from Deterministic Requirements (VFDRs)

Based on the information provided in the LAR, the licensee did not identify any VFDRs or any previous exemptions or other licensing actions credited in transition.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features."

The applicable portion of Table 4-4 has been included below identifying the ionization smoke detection and CT-4 transformer deluge as required detection and suppression systems to support the engineering evaluation.

Fire Area	Fire Zone	Zone Description	Suppression Required System?				Detection Required System?			
			E	R	D	S	E	R	D	S
CT4	46	CT-4 Block House	Yes	No	No	No	Yes	No	No	No

Legend:	
E - EEEE/LA:	Systems required for acceptability of EEEE / NRC-approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of DID for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for NFPA 805, Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in Table 4-4 and Attachment S of LAR

Fire Area CT-4 Conclusion

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area CT-4 meets the deterministic requirements of NFPA 805 Section 4.2.3. This conclusion is based on the following:

1. The licensee's documented compliance to NFPA 805 Section 4.2.3.2 and the licensee's assertion that the success path will be free of fire damage without reliance on recovery actions.
2. The licensee's assessment of the impact of suppression systems on the ability to meet the nuclear safety performance criteria.
3. The licensee's determination of the suppression and detection systems required to meet the nuclear safety performance criteria.

Fire Area KEO, Keowee Hydro Station

The licensee stated that deterministic compliance has been met in accordance with NFPA 805 Section 4.2.3.2, without the use of recovery actions, which requires that one success path of required cables and equipment be located in a separate area having boundaries containing fire barriers with a minimum fire resistance rating of three hours. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in LAR Attachment C, "NEI 04-02 Table B-3 Fire Area Transition," those safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions either. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Variation from Deterministic Requirements (VFDRs)

Based on the information provided in the LAR, the licensee did not identify any VFDRs or any previous exemptions or other licensing actions credited in transition.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features." The applicable portions of Table 4-4 have been included below.

Fire Area	Fire Zone	Zone Description	Suppression Required System?				Detection Required System?			
			E	R	D	S	E	R	D	S
KEO	KEO	Keowee Hydro Station	No	No	No	No	No	No	No	No

Legend:

E - EEEE/LA:	Systems required for acceptability of EEEE / NRC-approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of DID for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for NFPA 805, Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in Table 4-4 and Attachment S of LAR

Fire Area KEO Conclusion

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area KEO meets the deterministic requirements of NFPA 805, Section 4.2.3. This conclusion is based on the following:

1. The licensee's documented compliance to NFPA 805 Section 4.2.3.2, and assertion that the success path will be free of fire damage without reliance on recovery actions.
2. The licensee's assessment of the impact of suppression systems on the ability to meet the nuclear safety performance criteria.
3. The licensee's determination that suppression and detection systems were not required to meet the nuclear safety performance criteria.

Fire Area PSW, Protected Service Water (PSW) Building (Planned)

The PSW building is a plant modification that had not been constructed as of the issuance of this SE, but is credited by the licensee in its RI/PB FPP. The licensee stated that deterministic compliance has been met in accordance with NFPA 805, Section 4.2.3.2, without the use of recovery actions, which requires that one success path of required cables and equipment be located in a separate area having boundaries containing fire barriers with a minimum fire resistance rating of three hours. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

No evaluation performed since the PSW had not been constructed at the time of the LAR submittal.

Variation from Deterministic Requirements (VFDRs)

VFDR #	VFDR Description	Component
PSW-01	Ensure PSW modification is incorporated into ale the required documents.	PSW Modification (SE Section 2.8.1, Item 1).

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems needed for this area when constructed. The results of the evaluation were documented in LAR Table 4-4, "Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features." The applicable portions of Table 4-4 have been included below. Detection will be installed during Implementation of the PSW modification (SE Section 2.8.12, Item 1). The identified fire detection system is relied upon to meet the DID criteria.

Fire Area	Fire Zone	Zone Description	Suppression Required System?				Detection Required System?			
			E	R	D	S	E	R	D	S
PSW	PSW	Protected Service Water Building	No	No	No	No	No	No	No	No

Legend:

E - EEEE/LA:	Systems required for acceptability of EEEE / NRC-approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of DID for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for NFPA 805, Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in Table 4-4 and Attachment S of LAR

Fire Area PSW Conclusion

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area PSW meets the deterministic requirements of NFPA 805, Section 4.2.3. This conclusion is based on the following:

1. The licensee's documented compliance to NFPA 805, Section 4.2.3.2, and assertion that the success path will be free of fire damage without reliance on recovery actions.
2. The licensee's assessment of the impact of suppression systems on the ability to meet the nuclear safety performance criteria.
3. The licensee's determination that suppression and detection systems were not required to meet the nuclear safety performance criteria.

Attachment D2, Performance-Based Compliance with

NFPA 805, Section 4.2.4

For each fire area where the licensee selected FRE as the PB approach, the NRC staff verified that the change in risk is appropriately defined, the magnitude is acceptable and DID and sufficient SMs are maintained. The NRC staff has also verified that the additional risk of recovery actions is acceptable.

The licensee included an assessment of DID and SMs in the FRE for each of the areas addressed using the PB approach. Each FRE assessed the aspects of DID, including passive fire protection features (fire barriers, through penetration fire stops, penetration seals, radiant energy shields, etc.), active fire protection features (doors and dampers), and programmatic controls (combustible controls, hot work, design-flame spread of surfaces, electrical design, etc.), as well as manual suppression using fire extinguishers and hoses.

The licensee evaluated suppression and detection systems using a process that looked at several key aspects of the FPP to determine if a given system is required (i.e., deterministically in support of compliance with NFPA 805 Chapter 4, in support of a previous NRC approved exemption or other licensing action, in support of a licensee-developed EEEE, or as a result of the PB evaluations).

Accordingly, in addition to a discussion regarding risk, recovery actions (as applicable), DID, and SMs, each of the fire areas listed below also include a discussion of those fire suppression and fire detection systems the licensee has determined to be required to meet the nuclear safety performance criteria.

The licensee included in the VFDR motor-operated valves that were susceptible to failure as described in NRC Information Notice 92-18 "Potential for Loss of Remote Shutdown Capability During a Control Room Fire" (Reference 55). This issue addressed hot shorts, combined with the absence of thermal overload protection, which could cause valve damage before the operator shifted control of the valves to the remote/ alternate shutdown panel. The Licensee's description of the variance includes "This valve may suffer IN 92-18 damage. This condition represents a variance from the deterministic requirements of NFPA 805, Section 4.2.3. This is a separation issue." Individual risk evaluations for these failures were dispositioned as "ε" for both the change in CDF and change in LERF and fire detection coverage was identified as required for DID in each case. An evaluation for compliance using the PB approach of NFPA 805, Section 4.2.4 was performed for each potential failure.

Fire Area AB, Auxiliary Building

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also applied deterministic simplifying assumptions in order to credit those portions of the facility design that meet the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria for Fire Area AB

The licensee stated in LAR Attachment C, "NEI 04-02 Table B-3 Fire Area Transition," that a safe and stable condition can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited four previously approved licensing actions and exemptions from the existing fire protection requirements. The licensee used the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward these exemptions and other licensing actions, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of each exemption and other licensing actions is provided in the table below.

Exemption/Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, SSF, Lack of instrumentation per III.L.2	The lack of source range monitoring is acceptable because: • Unit held at hot standby. • Control rods are inserted. • RCS makeup and boration is with SFP water as this is the only source available with the existing piping design. The lack of SG pressure instrumentation is acceptable because: • Steam pressure is not a control parameter for operators. • SG level will be used to control auxiliary feedwater flow. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.
Approval of SSD System (SSS) Design	The main design features: • Capable of maintaining a hot standby condition in all units without any damage control measures and the ability to withstand SSD earthquake seismic loadings. • Utilizes natural circulation to remove decay heat from the primary coolant, use of secondary side steam valves to the atmosphere as a heat sink, and providing an independent power system. • The SSF is designed to provide an alternate and independent means to achieve and maintain hot standby conditions for one or more of the three ONS units. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this licensing action and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains

Exemption/Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
	• Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	valid, the NRC staff finds this acceptable.
Appendix R Exemption, AB Lack of three hour fire rated barrier	Presented justification for the lack of three hour fire barriers because: • Low combustible loading in pipe tunnel access area. • Fire propagation path is circuitous, consisting of several unrated barriers and open areas. • If a fire were to occur, it would develop slowly. • Fire brigade may use portable extinguishers, manual hose stations, or a fire hose supplied from a nearby fire hydrant. Although the exact number and configuration of combustibles may have changed over time, the bases for previous acceptance remain valid as substantiated by field walkdown	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)

Fire Area AB has a total of 52 VFDRs, which are provided in the table below. 36 of these VFDRs that are NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection for the fire area AB is required to meet the risk acceptance criteria. The Fire PRA makes assumptions regarding the time of fire discovery, fire brigade notification, and brigade manual suppression. These assumptions determine the impact of the fire, including the likelihood of a HGL being formed in the compartment. Specifically, the Fire PRA assumes a fire brigade response time of 20 minutes or less. The existing fire zone detection system coverage of the general area and/or hazard necessary for this assumption to be valid was not considered sufficient to conservatively meet the risk criteria. Therefore, modifications to the fire detection system for fire area AB are required to support the fire risk analysis assumption of 20 minute brigade response time.

Based on the reliance on fire detectors in fire area AB to meet the risk criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation. Improvements to the following fire zones in AB for general area and/or fire hazard detection are required: 61, 68, 72, 77, 94, 99, 103, 108, 110, 112, 115, 118, and 121 (SE Section 2.8.1).

Five of the 51 VFDRs, AB-34, AB-35, AB-40, AB-41, and AB-42 are a VFDR of NFPA 805 Section 4.2.3 (separation issue) that will be corrected with a plant modification. According to the LAR, the walls separating the following areas will be modified:

- TB / AB
- AB/ West Penetration Rooms
- Unit 1 Purge Inlet Room / SFP Area
- Unit 2 Purge Inlet Room / SFP Area
- Unit 3 Purge Inlet Room / SFP Area

These barriers are not currently three hour fire rated walls as required by NFPA 805 Section 3.11.1, and all of the penetrations in the wall do not have a fire resistance three hour rating as required by NFPA 805 Section 3.11.3. These barriers are credited for fire area separation using the deterministic approach of NFPA 805 Section 4.2.3. The licensee has committed to make modifications to the walls to bring them into compliance with the requirements of NFPA 805 (SE Section 2.8.1).

One VFDR, AB-39, regarding the SSF DG requires that the monitoring and/or adjustment of the following parameters is required during operation of the SSF DG: generator current, voltage, power and frequency. The controls and indications required to monitor and adjust these parameters are currently not included in the SSD analysis for those fire areas where the SSF is credited for accomplishing SSD (AB Fire Area only). Incorporation of these activities into the SSD procedure is an implementation item and resolve this VFDR (SE Section 2.9, Table 2.9-1,Item 13).

By letter dated November 19, 2010 (Reference 52), the licensee identified nine additional VFDRs (AB-43 through AB-51) as a result of eliminating the "*10-minute free of fire damage*" assumption. This issue is discussed in greater detailing SE Section 3.2.1. Upon completion of these activities, all applicable FREs will be updated and compliance will be demonstrated. These activities are an implementation item (SE Section 2.9, Table 2.9-1, Item 46).

In addition, by letter dated November 19, 2010 (Reference 52), the licensee identified an additional VFDR (AB-52) as a result of reclassifying the deployment and operation of the SSF submersible pump as a (recovery action) RA. This issue is discussed in greater detail in SE Section 3.2.4. The licensee conducted a bounding assessment from the additional risk of this RA. The evaluation determined the risk was sufficiently small and met the risk acceptance guidelines associated with pre-approved RAs. For a discussion of the NRC's staff review of this issue, see SE Section 3.4.4.

VFDR #	VFDR Description	Component
AB-01	The Protected Service Water (PSW) Pump is required to be off to isolate PSW flow to the SGs. Fire damage to cables may result in a spurious start of the PSW Pump. Spurious operation could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	0PSWPU0002 - PSW Pump
AB-02	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in spurious pump start. Spurious operation could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	1FDWPU0001, 1FDWPU0002 - Main Feedwater Pumps
AB-03	The Emergency Feedwater (EFW) Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in spurious pump	1FDWPU0003, 1FDWPU0004,

VFDR #	VFDR Description	Component
	start. Spurious operation could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	1FDWPU0005 - EFW Pumps
AB-04	These normally closed and required closed valves isolate the flow path from the LDST [letdown storage tank] to the containment sump. Fire induced cable damage may result in spurious opening of valve 1HP VA0939 and/or 1HP VA0940 resulting in a diversion of BWST inventory to the containment sump, flooding of the credited SSF reactor coolant makeup (RCMU) Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	1HP VA0939, 1HP VA0940 - LDST to Emergency Sump MOVs
AB-05	The HPI Pumps are required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start. Spurious operation could increase RC pressure to the pressurizer safety relief valve set point. Subsequent failure of the HPI pump(s) and failure of the relief valve to reseat could result in loss of RC inventory in excess of the makeup capability of the SSF RCMU Pump. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1HPIPU0001, 1HPIPU0002, 1HPIPU0003 - HPI Pumps
AB-06	These normally open, required closed valves isolate the flow path from the BWST to the Low Pressure Injection (LPI) Pumps, RB Spray (RBS) Pumps, and containment sump. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of BWST inventory to the containment sump, flooding of the credited SSF RCMU Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	1LP VA0021, 1LP VA0022 - BWST Suction MOVs
AB-07	These normally open, required closed valves isolate flow paths from the Main Steam Headers (MSHs). Fire damage to cables may result in spurious opening of the valves which could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation Valves
AB-08	The pressurizer heaters are required to be off to prevent an uncontrolled increase in RC pressure. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC pressure to the setpoint of the RCMU pump discharge relief valve. Diversion of RC makeup flow from the relief valve could degrade performance of the SSF RCMU Pump and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC HE0001, 1RC HE0002 (Groups D & K), 1RC HE0003, 1RC HE0004 - Pressurizer Heaters
AB-09	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the SSF. Fire damage to cables may result in spurious operation of the RCPs and place the SSF in an unanalyzed condition, e.g., add RCP heat to RCS, disrupt natural circulation flow, cause seal leakage in excess of the makeup capability of the SSF RCMU Pump. These conditions challenge the Inventory and Pressure Control, and Decay Heat Removal Nuclear Safety Performance Criteria.	1RC PU0001, 1RC PU0002, 1RC PU0003, 1RC PU0004 - RCPs
AB-10	These normally closed, required closed valves isolate flow paths from the RCS to containment. Fire damage to cables behind the main control boards in the control room may result in spurious opening of the above valves. The spurious opening of any of the above valves may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC VA0155, 1RC VA0157, 1RC VA0159 - RC Hot Leg and Head Vent Valves
AB-11	This normally closed, required closed valve isolates the flow path from the RCS to the post accident liquid sampling system. Fire damage to cables may result in spurious opening of the above valve. Spurious opening of this valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC VA0179 - Post Accident Sample Air-Operated Valve (AOV)

VFDR #	VFDR Description	Component
AB-12	These normally closed, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump and challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	1SD VA0027, 1SD VA0290, 1SD VA0418, 1SD VA0419, 1SD VA0420, 1SD VA0421 - SG Isolation MOVs
AB-13	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in spurious pump start. Spurious operation of the MFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	2FDWPU0001, 2FDWPU0002 - Main Feedwater Pumps
AB-14	The EFW Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in spurious pump start. Spurious operation of the EFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	2FDWPU0003, 2FDWPU0004, 2FDWPU0005 - EFW Pumps
AB-15	These normally closed and required closed valves isolate the flow path from the LDST to the containment sump. Fire induced cable damage may result in spurious opening of valve 2HP VA0939 and/or 2HP VA0940 resulting in a diversion of BWST inventory to the containment sump, flooding of the credited SSF RCMU Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	2HP VA0939, 2HP VA0940 - LDST to Emergency Sump MOVs
AB-16	The HPI Pumps are required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start. Spurious operation of HPI pump(s) could increase RC pressure to the pressurizer safety relief valve set point. Subsequent failure of the HPI pump(s) and failure of the relief valve to reseat could result in loss of RC inventory in excess of the makeup capability of the SSF RCMU Pump. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2HPIPU0001, 2HPIPU0002, 2HPIPU0003 - HPI Pumps
AB-17	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of BWST inventory to the containment sump, flooding of the credited SSF RCMU Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	2LP VA0021, 2LP VA0022 - BWST Suction MOVs
AB-18	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables may result in spurious opening of the valves which could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation Valves
AB-19	The pressurizer heaters are required to be off to prevent an uncontrolled increase in RC pressure. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC pressure to the setpoint of the RCMU pump discharge relief valve. Diversion of RC makeup flow from the relief valve could degrade performance of the SSF RCMU Pump and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC HE0001, 2RC HE0002 (Groups D & K), 2RC HE0003, 2RC HE0004 - Pressurizer Heaters
AB-20	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the SSF. Fire damage to cables may result in spurious operation of the RCPs and place the SSF in an unanalyzed condition, e.g., add RCP heat to RCS, disrupt natural circulation flow, cause seal leakage in excess of the makeup capability of the SSF RCMU Pump. These conditions challenge the Inventory and Pressure Control, and Decay Heat	2RC PU0001, 2RC PU0002, 2RC PU0003, 2RC PU0004 - RCPs

VFDR #	VFDR Description	Component
	Removal Nuclear Safety Performance Criteria.	
AB-21	These normally closed, required closed valves isolate flow paths from the RCS to containment. Fire damage to cables may result in spurious opening of the above valves. The spurious opening of any of the above valves may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0155, 2RC VA0157, 2RC VA0159 - RC Hot Leg and Head Vent Valves
AB-22	This normally closed, required closed valve isolates the flow path from the RCS to the post accident liquid sampling system. Fire damage to cables may result in spurious opening of the above valve. Spurious opening of this valve would result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria	2RC VA0179 - Post Accident Sample AOV
AB-23	These normally closed, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump and challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	2SD VA0027, 2SD VA0290, 2SD VA0418, 2SD VA0419, 2SD VA0420, 2SD VA0421 - SG Isolation MOVs
AB-24	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in spurious pump start. Spurious operation of the MFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	3FDWPU0001, 3FDWPU0002 - Main Feedwater Pumps
AB-25	The EFW Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in spurious pump start. Spurious operation of the EFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the Decay Heat Removal Nuclear Safety Performance Criterion.	3FDWPU0003, 3FDWPU0004, 3FDWPU0005 - EFW Pumps
AB-26	These normally closed and required closed valves isolate the flow path from the LDST to the containment sump. Fire induced cable damage may result in spurious opening of valve 3HPVA0939 and/or 3HP VA0940 resulting in a diversion of BWST inventory to the containment sump, flooding of the credited SSF RCMU Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	3HP VA0939, 3HP VA0940 - LDST to Emergency Sump MOVs
AB-27	The HPI Pumps are required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start. Spurious operation of HPI pump(s) could increase RC pressure to the pressurizer safety relief valve set point. Subsequent failure of the HPI pump(s) and failure of the relief valve to reseat could result in loss of RC inventory in excess of the makeup capability of the SSF RCMU Pump. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3HPIPU0001, 3HPIPU0002, 3HPIPU0003 - HPI Pumps
AB-28	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of BWST inventory to the containment sump, flooding of the credited SSF RCMU Pump and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	3LP VA0021, 3LP VA0022 - BWST Suction MOVs
AB-29	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables may result in spurious opening of the valves which could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation Valves

VFDR #	VFDR Description	Component
AB-30	The pressurizer heaters are required to be off to prevent an uncontrolled increase in RC pressure. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC pressure to the setpoint of the RCMU pump discharge relief valve. Diversion of RC makeup flow from the relief valve could degrade performance of the SSF RCMU Pump and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC HE0001, 3RC HE0002 (Groups D & K), 3RC HE0003, 3RC HE0004 - Pressurizer Heaters
AB-31	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the SSF. Fire damage to cables may result in spurious operation of the RCPs and place the SSF in an unanalyzed condition, e.g., add RCP heat to RCS, disrupt natural circulation flow, cause seal leakage in excess of the makeup capability of the SSF RCMU Pump. These conditions challenge the Inventory and Pressure Control, and Decay Heat Removal Nuclear Safety Performance Criteria	3RC PU0001, 3RC PU0002, 3RC PU0003, 3RC PU0004 - RCPs
AB-32	These normally closed, required closed valves isolate flow paths from the RCS to containment. Fire damage to cables may result in spurious opening of the above valves. The spurious opening of any of the above valves may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0155, 3RC VA0157, 3RC VA0159 - RC Hot Leg and Head Vent Valves
AB-33	These normally closed, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory in excess of the makeup capability of the RCMU Pump and challenge the Decay Heat Removal Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	3SD VA0027, 3SD VA0418, 3SD VA0419, 3SD VA0420, 3SD VA0421 - SG Isolation MOVs
AB-34	The wall separating the TB and AB is not three hour rated as required by NFPA 805, Section 3.11.1 and all the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	TB / AB Wall
AB-35	The wall separating the AB and the West penetration room does not have a fire-resistance rating required by NFPA 805, Section 3.11.2 and all the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	AB / West Penetration Room Separation
AB-36	This normally closed, required closed valve isolates a flow path from the RCS to containment. Fire damage to cables in the penetration box in the East Penetration Room may result in spurious opening of the above valve. The spurious opening of the valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC VA0155 - RC Hot Leg Vent Valve
AB-37	These normally closed, required closed valves isolate flow paths from the RCS to containment. Fire damage to cables in the penetration box in the East Penetration Room may result in spurious opening of these valves. The spurious opening of the valves may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0155, 2RC VA0157 - RC Hot Leg Vent Valves
AB-38	This normally closed, required closed valve isolates a flow path from the RCS to containment. Fire damage to cables in the penetration box in the East Penetration Room may result in spurious opening of the above valve. The spurious opening of the valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0155 - RC Hot Leg Vent Valve
AB-39	The monitoring and/or adjustment of the following parameters is required during operation of the SSF DG; generator current, voltage, power and frequency. The controls and indications required to monitor and adjust these parameters are currently not included in the SSD analysis for those	SSF DG

VFDR #	VFDR Description	Component
	fire areas where the SSF is credited for accomplishing SSD.	
AB-40	The areas separating the Unit 1 Purge Inlet Room and SFP area is not three hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
AB-41	The areas separating the Unit 2 Purge Inlet Room and SFP area is not three hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
AB-42	The areas separating the Unit 3 Purge Inlet Room and SFP area is not three hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
AB-43	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates flow path from the Pressurizer upon spurious opening of the PORV. Fire damage to cables may result in spurious opening of the valve. The spurious opening of the valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1RC VA0004, RC PORV Motor Operated Block Valve
AB-44	In supplementary information (Reference 52), the licensee identified that these normally open, required closed valves isolate the flow path from the RCS. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	1HP VA0003 and 1HP VA0004 1A and 1B Letdown Cooler Outlet Valves
AB-45	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates the flow path from the RCS. Fire damage to cables may prevent this valve from being closed or may result in spurious opening of the valve. The failure to close this valve or the spurious opening of the valve may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1HP VA0020 RCP Seal Return Valve
AB-46	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates flow path from the Pressurizer upon spurious opening of the PORV. Fire damage to cables may result in spurious opening of the valve. The spurious opening of the valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2RC VA0004, RC PORV Motor Operated Block Valve
AB-47	In supplementary information (Reference 52), the licensee identified that these normally open, required closed valves isolate the flow path from the RCS. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	2HP VA0003 and 2HP VA0004 2A and 2B Letdown Cooler Outlet Valves
AB-48	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates the flow path from the RCS. Fire damage to cables may prevent this valve from being closed or may result in spurious opening of the valve. The failure to close this valve	2HP VA0020 RCP Seal Return Valve

VFDR #	VFDR Description	Component
	or the spurious opening of the valve may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	
AB-49	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates flow path from the Pressurizer upon spurious opening of the PORV. Fire damage to cables may result in spurious opening of the valve. The spurious opening of the valve may result in a loss of RC inventory in excess of that provided by the SSF RCMU Pump and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3RC VA0004, RC PORV Motor Operated Block Valve
AB-50	In supplementary information (Reference 52), the licensee identified that these normally open, required closed valves isolate the flow path from the RCS. Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	3HP VA0003 and 3HP VA0004 1A and 1B Letdown Cooler Outlet Valves
AB-51	In supplementary information (Reference 52), the licensee identified that this normally open, required closed valve isolates the flow path from the RCS. Fire damage to cables may prevent this valve from being closed or may result in spurious opening of the valve. The failure to close this valve or the spurious opening of the valve may result in a diversion of RCS inventory and challenge to the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3HP VA0020 RCP Seal Return Valve
AB-52	In supplementary information (Reference 52), the licensee identified that the retrieval, assembly, and water body deployment of the portable submersible pump including necessary hose(s) and electrical power are not predominantly conducted in the SSF or deployed during the initial transfer of control from the control room. The deployment and operation of the SSF submersible pump is only credited for floods and is not currently credited for any fire scenarios. In a fire scenario, flow is maintained to the Condenser Circulating Water (CCW) piping providing that either the CCW pumps or the Essential Siphon Vacuum (ESV) provides flow. Thus, if modeled, the delta risk associated with the failure to deploy the submersible pump is expected to be epsilon (ε) for fire events. This conclusion is further substantiated by insights gained from the internal events PRA and the associated expert panel reviews.	SSF Submersible Pump

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

By letter dated November 19, 2010 (Reference 52), the licensee identified one recovery action that is credited in this fire area for meeting the nuclear safety performance criteria, and is provided in the following table:

Component ID	Component Name	Description of Action
SSF Submersible Pump	SSF Submersible Pump	The deployment and operation of the SSF submersible pump is only credited for floods and is not currently credited for any fire scenarios. In a fire scenario, flow is maintained to the Condenser Circulating Water (CCW) piping providing that either the CCW pumps or the Essential Siphon Vacuum (ESV) provides flow. The submersible pump would only be required

		in the fire scenario under the following set of conditions: CCW flow is insufficient only if the CCW pumps are not running, the lake level is too low to support backflow through the condensate coolers, and the ESV systems are unavailable to maintain adequate siphon.

Note: A bounding assessment of the additional risk being added because of this RA was determined by the NRC staff to be sufficiently small that the risk acceptance guidelines associated with pre-approved recovery actions have all been met. See SE Section 3.4.4 for a detailed discussion of the NRC staff's review of the estimated risk for this RA.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4, applicable portions included below. The identified fire detection system modifications are to improve plant fire detection and fire brigade response time.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
AB		Auxiliary Building										
	48	Unit 3 LPI & RBSP	No	No	No	No	No	Yes	No	Yes	No	No
	49	Unit 3 LPI & RBSP	No	No	No	No	No	Yes	No	Yes	No	No
	50	Unit 3 HPI Pump Area	No	No	No	No	No	Yes	No	Yes	No	No
	50A	Unit 3 HPI Pump, Spt Resin Xfr Pump Waste Tank, Waste & CT Drain Pumps	No	No	No	No	No	Yes	No	Yes	No	No
	51	Unit 3 Purification & Deboration Demineralizers	No	No	No	No	No	No	No	No	No	No
	52	Unit 2 LPI Pumps & Valve Room (Inside Room 63)	No	No	No	No	No	Yes	No	Yes	No	No
	53	Units 1 & 2 LPI Pumps & RBSP	No	No	No	No	No	Yes	No	Yes	No	No
	54	Unit 1 LPI Pumps & Valve Room (Inside Room 61)	No	No	No	No	No	Yes	No	Yes	No	No
	55	Unit 1 RB Sump & Cmp Drain Pmp, HPI Pmp, Spt Res Transfer	No	No	No	No	No	Yes	No	Yes	No	No
	55A	Units 1 & 2 HPI Pump Area	No	No	No	No	No	Yes	No	Yes	No	No
	56	Unit 2 Spt Res Transfer Pmp, HPI Pmp, RB Sump & Cmp Drain	No	No	No	No	No	Yes	No	Yes	No	No
	57	Units 1 & 2 Purification & Deboration Demineralizers None	No	No	No	No	No	No	No	No	No	No
	58	Unit 3 Boric Acid Mix, Spt Res Storage, RC BHUT CBAST, Misc	No	No	No	No	No	Yes	No	Yes	No	No
	59	Unit 3 Decay Heat Removal Coolers, Seal Supply Filter/Pipe	No	No	No	No	No	No	No	No	No	No
	60	Unit 3 LPI Room Hatch Area	No	No	No	No	No	Yes	No	Yes	No	No
	61	Unit 3 HPI Room Hatch Area	No	No	No	No	No	No	No	Yes (MR)	No	No
	62	Unit 3 Operators Panel/Chemical Sample Hood	No	No	No	No	No	Yes	No	Yes	No	No
	63	Unit 3 LDST, LD Filters, LD Filter Hatch	No	No	No	No	No	No	No	No	No	No
	64	Unit 2 Emergency Aux SW Pump	No	No	No	No	No	Yes	No	Yes	No	No
	65	Unit 2 MWHT, Misc Waste Evaporator, CBAST, RC Bleed Xfer Pmp,	No	No	No	No	No	Yes	No	Yes	No	No
	66	Unit 2 Decay Heat Removal Coolers, Seal Supply Filter/Pipe	No	No	No	No	No	No	No	No	No	No
	67	Unit 2 LPI Room Hatch Area	No	No	No	No	No	Yes	No	Yes	No	No
	68	Unit 2 HPI Room Hatch Area	No	No	No	No	No	No	No	Yes	No	No

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
										(MR)		
	69	Unit 2 Operators Panel/Chemical Sample Hood	No	No	No	No	No	Yes	No	Yes	No	No
	70	Unit 1 LPI Room Hatch Area	No	No	No	No	No	Yes	No	Yes	No	No
	71	Unit 2 LDST, LD Filters, LD Filter Hatch	No	No	No	No	No	No	No	No	No	No
	72	Unit 1 HPI Room Hatch Area	No	No	No	No	No	No	No	Yes (MR)	No	No
	73	Unit 1 LDST, LD Filters, LD Filter Hatch	No	No	No	No	No	No	No	No	No	No
	74	Units 1 & 2 Dress out Area for Units 1 & 2 HPI Hatch Areas	No	No	No	No	No	No	No	No	No	No
	75	Unit 1 Pipe Rooms, Seal Supply Filter/Pipe Room	No	No	No	No	No	No	No	No	No	No
	76	Unit 1 RC HU Tanks, CBAST, RC Bleed Xfr Pmp, Wst Dmg, Fltr Room, SRST	No	No	No	No	No	Yes	No	Yes	No	No
	77	Unit 3 Storage, Chemistry Storage	No	No	No	No	No	No	No	Yes (MR)	No	No
	78	Unit 3 Spent Fuel Cooler Filters/Demin, Spent Fuel Coolers	No	No	No	No	No	No	No	No	No	No
	79	Unit 3 RB Component Coolers	No	No	No	No	No	Yes	No	Yes	No	No
	80	Unit 3 Waste Gas Decay Tanks, Waste Gas Comp Room	No	No	No	No	No	No	No	No	No	No
	81	Unit 2 I&E Hot Shop, Misc Evaporator Feedwater Tank, Chemical Storage,	No	No	No	No	No	Yes	No	Yes	No	No
	82	Units 1 & 2 Spent Fuel Coolers, Spent Fuel Cooler Filter/Demin	No	No	No	No	No	No	No	No	No	No
	83	Units 1 & 2 RB Component Coolers	No	No	No	No	No	Yes	No	Yes	No	No
	84	Units 1 & 2 Waste Gas Decay Tanks, Waste Gas Comp	No	No	No	No	No	No	No	No	No	No
	85	Unit 1 Chemistry Storage, High Level Storage	No	No	No	No	No	Yes	No	Yes	No	No
	86	Unit 3 Hatch Area Chemistry Labs & Change Room	No	No	No	No	No	Yes	No	Yes	No	No
	88	Unit 3 Fuel Receiving Area, SFP & Loading Area	No	No	No	No	No	No	No	No	No	No
	89	Unit 3 Equipment Room	No	No	No	No	No	Yes	No	Yes	No	No
	90	Unit 2 Hallway, Change Room, Laundry Room, RP Lab, Chemistry Laboratory,	No	No	No	No	No	Yes	No	Yes	No	No

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
		Medical Room, and Decontamination (DECON) Room										
	92	Unit 2 Equipment Room	No	No	No	No	No	Yes	No	Yes	No	No
	93	Units 1 & 2 Fuel Receiving Area, SFP & Loading Area	No	No	No	No	No	No	No	No	No	No
	94	Unit 1 Hallway, Hatch Area, Change Room, and Tool	No	No	No	No	No	No	No	Yes (MR)	No	No
	95	Unit 1 Equipment Room	No	No	No	No	No	Yes	No	Yes	No	No
	96	Unit 2 Hot Machine Shop Tunnel, Hot Machine Shop	No	No	No	No	No	No	No	No	No	No
	99	Unit 3 East Penetration Room	No	No	No	No	No	No	No	Yes (MR)	No	No
	100	Unit 3 Control Battery Room	No	No	No	No	No	Yes	No	Yes	No	No
	101	Unit 3 Cable Room	Manual	Yes	No	No	No	Yes	Yes	Yes	No	No
	103	Unit 2 East Penetration Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	104	Unit 2 Control Battery Room	No	No	No	No	No	Yes	No	Yes	No	No
	105	Unit 2 Cable Room	Yes	Yes	No	No	No	Yes	Yes	Yes	No	No
	106	Unit 1 Cable Room	Manual	Yes	No	No	No	Yes	Yes	Yes	No	No
	108	Unit 1 East Penetration Room	No	No	No	No	No	No	No	Yes (MR)	No	No
	109	Unit 1 Control Battery Room	No	No	No	No	No	Yes	No	Yes	No	No
	109A	Unit 1 Control Room Lobby/AHU Room	No	No	No	No	No	No	No	No	No	No
	110	Units 1 & 2 Control Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	111	Unit 2 Control Room Lobby/AHU Room	No	No	No	No	No	No	No	No	No	No
	112	Unit 3 Control Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	113	Unit 3 Control Room Lobby/AHU Room	No	No	No	No	No	No	No	No	No	No
	115	Unit 3 Main Purge Exhaust Room	No	No	No	No	No	No	No	Yes (MR)	No	No
	116	Unit 3 AHU Room R	No	No	No	No	No	Yes	No	Yes	No	No

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
	118	Unit 2 Main Purge Exhaust Room	No	No	No	No	No	No	No	Yes (MR)	No	No
	119	Units 1 & 2 AHU Room	No	No	No	No	No	Yes	No	Yes	No	No
	121	Unit 1 Main Purge Exhaust Room	No	No	No	No	No	No	No	Yes (MR)	No	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC-approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria: Systems required for NFPA 805, Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required Systems are committed to be modified as indicated in Table 4-4 and Attachment S of LAR

Fire Area AB Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area AB meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4, to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:
  - On a fire zone basis, the fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  - Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations and through penetration fire stops and spatial separation.

- Three exemptions and one other licensing action from the pre-transition fire protection requirements were evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Forty-two VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:
  - Fire detection upgrades and/or new installation in thirteen (13) fire zones to improve plant fire detection and fire brigade response time.
  - Fire barriers upgrades to provide three hour fire rated separation, as follows:
    - Purge Inlet Rooms and Spent Fuel Poll Area for Units 1, 2, and 3 [AB-40, AB-41, and AB-42]
    - AB / TB [AB-34]
    - AB / West Penetration Room [AB-35]

Fire Area BH12, Units 1 and 2 Block House

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805, Section 4.2.4.2, but also applied deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805, Section

4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in LAR Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

The licensee did not credit any previously approved licensing actions or exemptions from the existing fire protection requirements.

Variation from Deterministic Requirements (VFDRs)

Fire Area BH12 has a total of 15 VFDRs, which are provided in the table below. All but one of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or fire hazard detection for the fire area BH12 is required to meet the DID criteria.

Based on the reliance on fire detectors in fire area BH12 to meet the DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation (SE Section 2.8.1).

One of the 15 VFDRs, BH12-02, is a variance from NFPA 805, Section 4.2.3 (degraded fire protection feature) that will be corrected with a plant modification. According to the LAR, the wall separating fire area BH12 from the fire area YARD does not currently have a three hour rated wall as required by NFPA 805, Section 3.11.1, and all of the penetrations in the wall do not have a three hour fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make a modification to install hinged steel covers/shields to the exterior side of the tornado vents that will qualify the wall 'adequate for the hazard' (SE Section 2.8.1).

VFDR #	VFDR Description	Component (Cables)
BH12-01	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
BH12-02	The penetrations in the wall interfacing the east wall of Blockhouse 1/2 and the east yard do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Tornado Vents in Blockhouse 1/2 Building Wall
BH12-03	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
BH12-04	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from the PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches
BH12-05	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	2HP VA0023 – HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
BH12-07	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
BH12-08	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from the PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer

VFDR #	VFDR Description	Component (Cables)
		Switches
BH12-09	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
BH12-11	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
BH12-12	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
BH12-14	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
BH12-15	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
BH12-16	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling
BH12-17	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 2 Containment Cooling
BH12-18	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results were documented in LAR Table 4-4 in the LAR, and the applicable portions have been included below. Partial detection is installed, so the existing detection requires an engineering evaluation. The identified fire detection system modifications are required to improve plant fire detection and fire brigade response time.

Suppression and detection systems were identified in LAR Table B-3 as elements required for the fire resistance qualification of the three hour fire rated wall. Suppression is limited to fire brigade capability and provides adequate suppression capability given the hazards in the BH12 fire area.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
BH12	45	Units 1 & 2 Block House	No	No	No	No	No	Yes	Yes (MR)	No	Yes (MR)	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria: Systems required for NFPA 805, Chapter 4, Separation Criteria in (Section 4.2.3)
MR-Modification Required Systems are committed to be modified as indicated in Table 4-4 and Attachment S

Fire Area BH12 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area BH12 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

— The fire protection detection systems required to meet the nuclear safety performance criteria were documented.

— Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations and through penetration fire stops and spatial separation.

- No exemptions or licensing actions from the pre-transition fire protection requirements were required for transition to the NFPA 805 RI/PB FPP.

- Fifteen VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

- Modification to improve general area and/or hazard detection for fire area BH12 were identified as required. These detection modifications are to improve plant fire detection and fire brigade response time.

- Modifications to install hinged steel covers/shields to the exterior side of the tornado vents that will qualify the wall as adequate for the hazard.

Fire Area BH3, Unit 3 Block House

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805, Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805, Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions.

Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

The licensee did not credit any previously approved licensing actions or exemptions from the existing fire protection requirements.

Variation from Deterministic Requirements (VFDRs)

Fire Area BH3 has a total of 14 VFDRs, which are provided in the table below. All of these VFDRs are variances from NFPA 805, Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or fire hazard detection for the fire area BH3 is required to meet the DID criteria.

Based on the reliance on fire detectors in fire area BH3 to meet the DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation (SE Section 2.8.1).

VFDR #	VFDR Description	Component (Cables)
BH3-01	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
BH3-03	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
BH3-04	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches
BH3-05	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during	2HP VA0023 - HPI Normal Suction MOV,

VFDR #	VFDR Description	Component (Cables)
	prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	2HP VA0939 - LDST to Emergency Sump MOV
BH3-07	These normally open, required closed valves isolate flow paths from the Main Steam Headers. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
BH3-08	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from the PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
BH3-09	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being operated.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
BH3-11	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
BH3-12	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from the PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
BH3-14	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Units 1 & 2 control complex	Units 1 & 2 Control Complex

VFDR #	VFDR Description	Component (Cables)
	exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Cooling
BH3-15	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
BH3-16	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling
BH3-17	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 2 Containment Cooling
BH3-18	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results were documented in LAR Table 4-4 and the applicable portions have been included below. Partial detection is installed in BH3 and modification is required to improve general area and/or hazard detection for DID. The identified fire detection systems modifications are required to improve plant fire detection and fire brigade response time.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
BH3	47	Unit 3 Block House	No	No	No	No	No	Yes	No	No	Yes (MR)	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4)
S - Separation Criteria: Systems required for NFPA 805, Chapter 4, Separation Criteria in (Section 4.2.3)
MR - Modification Required Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area BH3 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

    a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
    b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations and through penetration fire stops and spatial separation.

- No exemptions or licensing actions from the pre-transition fire protection requirements were required for transition to the NFPA 805 RI/PB FPP.

- Fourteen VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned to address the issues. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

    - Modification to improve general area and/or hazard detection for fire area BH3 were identified as required. These detection modifications are to improve plant fire detection and fire brigade response time.

Fire Area RB1, Unit 1 Reactor Building

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805, Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805, Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited two previously approved exemptions from the existing fire protection requirements. The licensee used the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward these exemptions, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of each exemption is provided in the table below.

Exemption / Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB 20 feet separation w/o intervening combustibles	Presented justification for the lack of 20 feet horizontal distance separation between SSD circuits with no intervening combustibles. For 20 feet separation with intervening combustibles: • More than 20 feet separation. • Low concentration of cables in cable trays. • Cable insulation is comparable to IEEE-383 qualified cables which burn slowly with an initial low rate of heat release. • Fire brigade response would be adequate. For pressurizer level instrumentation 15 feet separation (RB1 Only): • No intervening combustibles. • Low combustible loading in general area. • Administrative controls to limit transient combustibles in area. • Inspections prior to starting the unit after an outage. • RB is a huge structure to dissipate heat from a fire. • Fire brigade response would be adequate. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of the portion of this exemption having to do with lack of 20 feet horizontal distance separation between SSD circuits and the statement by the licensee that the basis remains valid, the NRC staff finds this portion of this acceptable. However, for the pressurizer level instrumentation 15 feet separation exemption, the NRC staff disagrees that the bases for previous acceptance remains valid because the separation distance was reported by the licensee to be less than 15 feet described in the exemption documentation. The NRC staff does not therefore find this portion of this acceptable. The licensee, however, further evaluated this as a VFDR (VFDR RB1-11) and determined it to have negligible risk with a recovery action (1RC P 0233) to provide DID. Incorporating this recovery action in the SSD procedures is an implementation item (SE Section 2.8, Table 2.8.1-1, Item 14).
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)

Fire Area RB1 has a total of 17 VFDRs, which are provided in the table below. All of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- All VFDRs require reliance on the general area and/or hazard detection associated with existing fire detection in the RB1 to meet the DID criteria.
- For VFDRs RB1-02, RB1-09, RB1-11, and RB1-12, in addition to the reliance on existing fire detection, recovery actions are identified to monitor alternative instrumentation as stated in LAR Attachment G Table G-2 and are identified as relied upon to meet DID criteria (SE Section 2.9, Table 2.9-1, Item 14).
- For VFDRs RB1-10 and RB1-16, in addition to reliance on existing fire detection, operator guidance will be inserted into shutdown procedures for operation of RC high point vent valves for RC letdown in the event that head vent valve flow path becomes inoperable (SE Section 2.9; Table 2.9-1,Item 30).

VFDR #	VFDR Description	Component (Cables)
RB1-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 1A and 1B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 1A or 1B SGs, and a challenge to the Decay Heat Removal Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1CCWVA0269 - SG A FDW Control MOV
RB1-02	SG level indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of SG level indication resulting in the inability of the operator to monitor and control level in either the 1A or 1B SGs from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1FDWP 0270, 1FDWP 0271 - SG Level Indications
RB1-03	This normally open, required open valve is located in the EFW flow path to the 1B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 1B SG. The subsequent decrease in SG shell temperature may result in 1B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and Decay Heat Removal Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1FDWVA0347 - SG B Inlet MOV
RB1-04	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV

VFDR #	VFDR Description	Component (Cables)
	Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	
RB1-06	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	1LP VA0021, 1LP VA0022 - BWST Suction MOVs
RB1-07	Source range flux indication is required for process monitoring and control. Fire induced cable damage may result in loss of source range flux indications resulting in the inability of the operator to monitor this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1RPSP 1007, 1RPSP 1008 - Source Range Flux
RB1-08	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
RB1-09	RC pressure indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC pressure indications resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1RC CR0045 - RC Pressure Recorder
RB1-10	Operation of the pressurizer heaters is required to maintain control of RC pressure. Fire damage to cables may result in a loss of the pressurizer heaters and could challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC HE0001, 1RC HE0002, 1RC HE0003, 1RC HE0004 - Pressurizer Heaters
RB1-11	Pressurizer level indication is required for process monitoring and diagnosis of plant transients. Fire impingement on instrument sensing lines or fire induced cable damage may result in loss of pressurizer level indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1RC P 0365 - Pressurizer Level Indication
RB1-12	RC temperature indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC temperature indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1RC P 0376 - RC Temperature Indication
RB1-13	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from the PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches
RB1-14	This normally closed, required closed valve isolates the flow path from the RCS to the Quench Tank. Fire induced cable damage may result in spurious opening of the PORV causing a loss of inventory and RC subcooling. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC VA0066 - Pressurizer Power Operated Relief Valve
RB1-15	These normally closed, required closed valves isolate flow paths from the RCS to containment. Potential hot shorts within the electrical	1RC VA0155, 1RC VA0157, 1RC VA0159 - RC Hot Leg and Head

VFDR #	VFDR Description	Component (Cables)
	penetration box may spuriously open the reactor head vent and hot leg vent valves. The spurious opening of these valves may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	Vent Valves
RB1-16	These normally closed valves isolate the flow path from the RCS to containment. These valves are required opened to provide an RC letdown flow path. Fire induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	1RC VA0159, 1RC VA0160 - RC Head Vent Valves
RB1-18	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
RB1-19	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

Recovery actions credited in this fire area to satisfy the DID requirements of NFPA 805, Section 4.2.4.2, are provided in the following table:

Component ID	Component Name	Description of Action
1FDWP 0232	U1 SSF SG 1B LEVEL INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 1FDWP 0232.
1RC P 0233	U1 SSF PRESSURIZER LEVEL INDICATION	For a fire in the east side of containment, monitor 1RC P 0365 from the control room if available; if not dispatch an operator to the SSF to monitor instrument 1RC P 0233.
1RC P 0238	U1 SSF RC LOOP B PRESSURE INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 1RC P 0238.
1RC P 0315	REACTOR OUTLET LOOP B	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 1RC P 0315.

Note: The FRE for this fire area determined that the additional risk being added because of these RAs was negligible for both change in CDF and change in LERF. See SE Section 3.4.2 for a detailed discussion of the NRC staff's review of the FREs.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results were documented in LAR Table 4-4 and the applicable portion has been included below.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
RB1	122	Unit 1 Reactor Building	No	No	No	No	No	Yes	No	No	Yes	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria: Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area RB1 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area RB1 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations and through penetration fire stops and spatial separation.

- Two exemptions from the pre-transition fire protection requirements were evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Seventeen VFDRs were identified, evaluated through the performance of a FRE, and found to meet the risk acceptance criteria, as well as the requirements for DID and SMs. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- Four recovery actions were identified and evaluated for the additional risk (change in CDF and change in LERF) each poses. The additional risk of each recovery action was conservatively estimated to be taken as the change in CDF and change in LERF associated with the VFDR that resulted in the need for the recovery action. The change in CDF and change in LERF for each recovery action was determined to be negligible.

Fire Area RB2, Unit 2 Reactor Building

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited one previously approved exemption from the existing fire protection requirements. The licensee used the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward this exemption, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of the exemption is provided in the table below.

Exemption / Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)

Fire Area RB2 has a total of 17 VFDRs, which are provided in the table below. All of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- All VFDRs require reliance on the general area and/or hazard detection associated with existing fire detection in the RB2 to meet the DID criteria.
- For VFDRs RB2-02, RB2-09, RB2-11, and RB2-12, in addition to the reliance on existing fire detection, recovery actions are identified to monitor alternative instrumentation as stated in Attachment G Table G-2 of the LAR and are identified as relied upon to meet DID criteria (SE Section 2.9, Table 2.9-1, Item 14).
- For VFDRs RB2-10 and RB2-16, in addition to reliance on existing fire detection, operator guidance will be inserted into shutdown procedures for operation of RC high point vent valves for RC letdown in the event that head vent valve flow path becomes inoperable (SE Section 2.9; Table 2.9-1, Item 30).

VFDR #	VFDR Description	Component (Cables)
RB2-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 2A and 2B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 2A or 2B SGs, and a challenge to the Decay Heat Removal Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2CCWVA0269 - SG A FDW Control MOV
RB2-02	SG level indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of SG level indication resulting in the inability of the operator to monitor and control level in either the 2A or 2B SGs from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	2FDWP 0270, 2FDWP 0271 - SG Level Indications
RB2-03	This normally open, required open valve is located in the EFW flow path to the 2B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 2B SG. The subsequent decrease in SG shell temperature may result in 2B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and Decay Heat Removal Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2FDWVA0347 - SG B Inlet MOV
RB2-04	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	2HP VA0023 - HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
RB2-06	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	2LP VA0021, 2LP VA0022 - BWST Suction MOVs
RB2-07	Source range flux indication is required for process monitoring and control. Fire induced cable damage may result in loss of source range flux indications resulting	2RPSP 1007, 2RPSP 1008 -

VFDR #	VFDR Description	Component (Cables)
	in the inability of the operator to monitor this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	Source Range Flux
RB2-08	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the Decay Heat Removal Nuclear Safety Performance Criterion.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
RB2-09	RC pressure indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC pressure indications resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	2RC CR0046 - RC Pressure Recorder
RB2-10	Operation of the pressurizer heaters is required to maintain control of RC pressure. Fire damage to cables may result in a loss of the pressurizer heaters and could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC HE0001, 2RC HE0002, 2RC HE0003, 2RC HE0004 - Pressurizer Heaters
RB2-11	Pressurizer level indication is required for process monitoring and diagnosis of plant transients. Fire impingement on instrument sensing lines or fire induced cable damage may result in loss of pressurizer level indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	2RC P 0365 - Pressurizer Level Indication
RB2-12	RC temperature indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC temperature indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	2RC P 0376 - RC Temperature Indication
RB2-13	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
RB2-14	This normally closed, required closed valve isolates the flow path from the RCS to the Quench Tank. Fire induced cable damage may result in spurious opening of the PORV causing a loss of inventory and RC subcooling. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0066 - Pressurizer Power Operated Relief Valve
RB2-15	These normally closed, required closed valves isolate flow paths from the RCS to containment. Potential hot shorts within the electrical penetration box may spuriously open the reactor head vent and hot leg vent valves. The spurious opening of these valves may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0155, 2RC VA0157, 2RC VA0159 - RC Hot Leg and Head Vent Valves
RB2-16	These normally closed valves isolate the flow path from the RCS to containment. These valves are required opened to provide an RC letdown flow path. Fire induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	2RC VA0159, 2RC VA0160 - RC Head Vent Valves
RB2-18	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety	Units 1 & 2 Control Complex Cooling

VFDR #	VFDR Description	Component (Cables)
	Performance Criterion.	
RB2-19	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 2 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.

Recovery Actions (RAs)

Recovery actions credited in this fire area to satisfy the DID requirements of NFPA 805, Section 4.2.4.2, are provided in the following table:

Component ID	Component Name	Description of Action
2FDWP 0232	U2 SSF SG 1B LEVEL INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 2FDWP 0232.
2RC P 0233	U2 SSF PRESSURIZER LEVEL INDICATION	For a fire in the east side of containment, monitor 2RC P 0365 from the control room if available; if not dispatch an operator to the SSF to monitor instrument 2RC P 0233.
2RC P 0238	U2 SSF RC LOOP B PRESSURE INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 2RC P 0238.
2RC P 0315	REACTOR OUTLET LOOP B	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 2RC P 0315.

Note: The FRE for this fire area determined that the additional risk being added because of these RAs was negligible for both change in CDF and change in LERF. See SE Section 3.4.2 for a detailed discussion of the NRC staff's review of the FREs.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results were documented in LAR Table 4-4 and the applicable portion has been included below.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
RB2	123	Unit 2 Reactor Building	No	No	No	No	No	Yes	No	No	Yes	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC-approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria: Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area RB2 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

- Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area RB2 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following: fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4, to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  The fire protection detection systems required to meet the nuclear safety performance criteria were documented.

  a. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops.

- One exemption from the pre-transition fire protection requirements was evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Seventeen VFDRs were identified, evaluated through the performance of a FRE, and found to meet the risk acceptance criteria, as well as the requirements for DID and SMs. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- Four recovery actions were identified and evaluated for the additional risk (change in CDF and change in LERF) each poses. The additional risk of each was conservatively estimated to be taken as the change in CDF and change in LERF associated with the VFDR that resulted in the need for the recovery action. The change in CDF and change in LERF for each recovery action was determined to be negligible.

Fire Area RB3, Unit 3 Reactor Building

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited one previously approved exemption from the existing fire protection requirements. The licensee utilized the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward this exemption, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of the exemption is provided in the table below.

Exemption / Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)

Fire Area RB3 has a total of 17 VFDRs, which are provided in the table below. All of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- All VFDRs require reliance on the general area and/or hazard detection associated with existing fire detection in the RB3 to meet the DID criteria.

- For VFDRs RB3-02, RB3-09, RB3-11, and RB3-12, in addition to the reliance on existing fire detection, recovery actions are identified to monitor alternative instrumentation as stated in Attachment G Table G-2 of the LAR and are identified as relied upon to meet DID criteria (SE Section 2.9; Item 14).

- For VFDRs RB3-10 and RB3-16, in addition to reliance on existing fire detection, operator guidance will be inserted into shutdown procedures for operation of RC high point vent valves for RC letdown in the event that head vent valve flow path becomes inoperable (SE Section 2.9; Item 30).

VFDR #	VFDR Description	Component (Cables)
RB3-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 3A and 3B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 3A or 3B SGs, and a challenge to the DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3CCWVA0269 - SG A FDW Control MOV
RB3-02	SG level indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of SG level indication resulting in the inability of the operator to monitor and control level in either the 3A or 3B SGs from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3FDWP 0270, 2FDWP 0271 - SG Level Indications
RB3-03	This normally open, required open valve is located in the EFW flow path to the 3B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 3B SG. The subsequent decrease in SG shell temperature may result in 3B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3FDWVA0347 - SG B Inlet MOV
RB3-04	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
RB3-06	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	3LP VA0021, 3LP VA0022 - BWST Suction MOVs
RB3-07	Source range flux indication is required for process monitoring and control. Fire induced cable damage may result in loss of source range flux indications resulting in the inability of the operator to monitor this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3RPSP 1007, 3RPSP 1008 - Source Range Flux
RB3-08	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
RB3-09	RC pressure indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC pressure indications resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3RC CR0045 - RC Pressure Recorder
RB3-10	Operation of the pressurizer heaters is required to maintain control of RC pressure. Fire damage to cables may result in a loss of the pressurizer heaters and could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC HE0001, 3RC HE0002, 3RC HE0003, 3RC

VFDR #	VFDR Description	Component (Cables)
		HE0004 - Pressurizer Heaters
RB3-11	Pressurizer level indication is required for process monitoring and diagnosis of plant transients. Fire impingement on instrument sensing lines or fire induced cable damage may result in loss of pressurizer level indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3RC P 0365 - Pressurizer Level Indication
RB3-12	RC temperature indication is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in loss of RC temperature indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3RC P 0376 - RC Temperature Indication
RB3-13	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
RB3-14	This normally closed, required closed valve isolates the flow path from the RCS to the Quench Tank. Fire induced cable damage may result in spurious opening of the PORV causing a loss of inventory and RC subcooling. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0066 - Pressurizer Power Operated Relief Valve
RB3-15	These normally closed, required closed valves isolate flow paths from the RCS to containment. Potential hot shorts within the electrical penetration box may spuriously open the reactor head vent and hot leg vent valves. The spurious opening of these valves may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0155, 3RC VA0157, 3RC VA0159 - RC Hot Leg and Head Vent Valves
RB3-16	These normally closed valves isolate the flow path from the RCS to containment. These valves are required opened to provide an RC letdown flow path. Fire induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	3RC VA0159, 3RC VA0160 - RC Head Vent Valves
RB3-18	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
RB3-19	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

Recovery actions credited in this fire area to satisfy the DID requirements of NFPA 805, Section 4.2.4.2, are provided in the following table:

Component ID	Component Name	Description of Action
3FDWP 0232	U3 SSF SG 1B LEVEL INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 3FDWP 0232.
3RC P 0233	U3 SSF PRESSURIZER LEVEL INDICATION	For a fire in the east side of containment, monitor 3RC P 0365 from the control room if available; if not dispatch an operator to the SSF to monitor instrument 3RC P 0233.
3RC P 0238	U3 SSF RC LOOP B PRESSURE INDICATION	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 3RC P 0238.
3RC P 0315	REACTOR OUTLET LOOP B	For a fire in the east side of containment, dispatch an operator to the SSF to monitor instrument 3RC P 0315.

Note: The FRE for this fire area determined that the additional risk being added because of these RAs was negligible for both change in CDF and change in LERF. See SE Section 3.4.2 for a detailed discussion of the NRC staff's review of the FREs.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portion has been included below.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
RB3	124	Unit 3 Reactor Building	No	No	No	No	No	Yes	No	No	Yes	No

Legend:	
E - EEEE/LA:	Systems required for acceptability of EEE Evaluations / NRC-approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in Table 4-4 and Attachment S of LAR

Fire Area RB3 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area RB3 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- One exemption from the pre-transition fire protection requirements was evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Seventeen VFDRs were identified, evaluated through the performance of a FRE, and found to meet the risk acceptance criteria, as well as the requirements for DID and SMs. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- Four recovery actions were identified and evaluated for the additional risk (change in CDF and change in LERF) each poses. The additional risk of each recovery action was conservatively estimated to be taken as the change in CDF and change in LERF associated with the VFDR that resulted in the need for the recovery action. The change in CDF and change in LERF for each recovery action was determined to be negligible.

- No modifications were identified as necessary for meeting requirements of NFPA 805.

- Two VFDR dispositions, in addition to reliance on existing fire detection, require operator guidance to be inserted into shutdown procedures for operation of RC high point vent valves for RC letdown in the event that head vent valve flow path becomes inoperable.

Fire Area SSF, Standby Shutdown Facility

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

The licensee did not credit any previously approved licensing actions or exemptions from the existing fire protection requirements.

Variation from Deterministic Requirements (VFDRs)

Fire Area SSF has a total of 32 VFDRs, which are provided in the table below. All of these VFDRs are variances from NFPA 805, Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection associated with the existing fire detection panel is required to meet the DID criteria.

VFDR #	VFDR Description	Component (Cables)
SSF-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 1A and 1B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 1A or 1B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	1CCWVA0269 - SG A FDW Control MOV
SSF-02	This normally closed, required closed valve isolates the flow path between the SSF ASW Pump and the 1B EFW header. Fire induced cable damage may result in spurious opening of this valve resulting in a diversion of PSW flow from the 1B EFW header and challenge the DHR Nuclear Safety Performance criterion. This valve may suffer IN 92-18 damage.	1CCWVA0287 - SSF ASW Pump to SG Supply MOV
SSF-03	This normally open, required open valve is located in the EFW flow path to the 1B SG. Fire-induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 1B SG. The subsequent decrease in SG shell temperature may result in 1B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1FDWVA0347 - SG B Inlet MOV
SSF-04	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
SSF-05	This normally closed, required closed valve isolates the RC letdown flow path to the SFP. Fire induced cable damage may result in spurious opening of this valve resulting in a loss of RC inventory from the RCS to the SFP. This could challenge the Reactivity and Inventory Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1HP VA0426 - RC Letdown to SFP MOV
SSF-07	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and challenge the Reactivity,	1LP VA0021, 1LP VA0022 - BWST Suction MOVs

VFDR #	VFDR Description	Component (Cables)
	Inventory and Pressure Control Nuclear Safety Performance Criteria.	
SSF-08	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area, and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
SSF-09	Pressurizer heaters are required for RC pressure control. Groups C & D of pressurizer heater bank 2 can be controlled from the SSF. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC HE0002 - Pressurizer Heater Bank 2 (Groups C & D)
SSF-11	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches
SSF-12	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 2A and 2B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 2A or 2B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	2CCWVA0269 - SG A FDW Control MOV
SSF-13	This normally closed, required closed valve isolates the flow path between the SSF auxiliary service water (ASW) Pump and the 2B EFW header. Fire induced cable damage may result in spurious opening of this valve resulting in a diversion of PSW flow from the 2B EFW header and challenge the DHR Nuclear Safety Performance criterion. This valve may suffer IN 92-18 damage.	2CCWVA0287 - SSF ASW Pump to SG Supply MOV
SSF-14	This normally open, required open valve is located in the EFW flow path to the 2B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 2B SG. The subsequent decrease in SG shell temperature may result in 2B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2FDWVA0347 - SG B Inlet MOV
SSF-15	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	2HP VA0023 - HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
SSF-16	This normally closed, required closed valve isolates the RC letdown flow path to the SFP. Fire induced cable damage may result in spurious opening of this valve resulting in a loss of RC inventory from the RCS to the SFP. This could challenge the Reactivity and Inventory Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2HP VA0426 - RC Letdown to SFP MOV
SSF-18	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	2LP VA0021, 2LP VA0022 - BSWT Suction MOVs
SSF-19	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area, and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079,

VFDR #	VFDR Description	Component (Cables)
	could challenge the DHR Nuclear Safety Performance Criterion.	2MS VA0082, 2MS VA0084 - SG Isolation MOVs
SSF-20	Pressurizer heaters are required for RC pressure control. Groups C & D of pressurizer heater bank 2 can be controlled from the SSF. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC HE0002 - Pressurizer Heater Bank 2 (Groups C & D)
SSF-22	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
SSF-23	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 3A and 3B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 3A or 3B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	3CCWVA0269 - SG A FDW Control MOV
SSF-24	This normally closed, required closed valve isolates the flow path between the SSF ASW Pump and the 3B EFW header. Fire induced cable damage may result in spurious opening of this valve resulting in a diversion of PSW flow from the 3B EFW header and challenge the DHR Nuclear Safety Performance criterion. This valve may suffer IN 92-18 damage.	3CCWVA0287 - SSF ASW Pump to SG Supply MOV
SSF-25	This normally open, required open valve is located in the EFW flow path to the 3B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 3B SG. The subsequent decrease in SG shell temperature may result in 3B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage	3FDWVA0347 - SG B Inlet MOV
SSF-26	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
SSF-27	This normally closed, required closed valve isolates the RC letdown flow path to the SFP. Fire induced cable damage may result in spurious opening of this valve resulting in a loss of RC inventory from the RCS to the SFP. This could challenge the Reactivity and Inventory Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3HP VA0426 - RC Letdown to SFP MOV
SSF-29	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	3LP VA0021, 3LP VA0022 - BSWT Suction MOVs
SSF-30	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area, and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
SSF-31	Pressurizer heaters are required for RC pressure control. Groups C & D of pressurizer heater bank 2 can be controlled from the SSF. Fire damage to cables may result in the spurious operation of these heaters resulting in an increase in RC	3RC HE0002 - Pressurizer Heater Bank 2 (Groups C & D)

VFDR #	VFDR Description	Component (Cables)
	pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	
SSF-33	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
SSF-35	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
SSF-36	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
SSF-37	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion	Unit 1 Containment Cooling
SSF-38	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion	Unit 2 Containment Cooling
SSF-39	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portion has been included below.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
SSF	SSF	Standby Shutdown Facility	Yes	No	No	No	No	Yes	No	No	Yes	No

Legend:

E - EEEE/LA:	Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area SSF Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area SSF meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- No exemptions or licensing actions from the pre-transition fire protection requirements were required.

- Thirty-two VFDRs were identified, evaluated through the performance of a FRE, and found to meet the risk acceptance criteria, as well as the requirements for DID and SMs. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- No modifications were identified as necessary for meeting requirements of NFPA 805.

Fire Area TB, Turbine Building

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805, Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805, Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

The licensee did not credit any previously approved licensing actions or exemptions from the existing fire protection requirements.

Variation from Deterministic Requirements (VFDRs)

Fire Area TB has a total of 45 VFDRs, which are provided in the table below. All but one of these VFDRs are variances from NFPA 805, Section 4.2.3, (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection for the TB Fire Area is required to meet the risk acceptance criteria. The Fire PRA makes assumptions regarding the time of fire discovery, fire brigade notification, and brigade manual suppression. These assumptions determine the impact of the fire, including the likelihood of a HGL being formed in the compartment. Specifically, the Fire PRA is based on a fire brigade response time of 20 minutes or less. The existing fire zone detection system coverage of the general area and/or hazard necessary for this assumption to be valid was not considered sufficient to conservatively meet the risk criteria. Therefore, modifications to the fire detection system in the TB Fire Area are required to support the fire risk analysis assumption of 20 minute brigade response time.

- General area and/or hazard detection associated with the Auxiliary Shutdown Panels FZ 39 and 41 is required to meet the DID criteria.

Based on the reliance on fire detectors in the TB Fire Area to meet the risk and DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation. Improvements to the following TB Fire Zones for general area and/or fire hazard detection are required: 3, 6, 12, 15, 19, 24, 25, 28, 29, 32, 33, 33A, 34, 34A, 35, 37, 39, 39A, and 41 (SE Section 2.8.1).

One of the 45 VFDRs, TB-06, is a variance from the deterministic requirements of NFPA 805, Section 4.2.3 (separation issue) that will be corrected with a plant modification. According to the LAR, the wall separating the TB and the AB is not currently a three hour rated wall as required by NFPA 805, Section 3.11.1, and all of the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make modifications to the wall to bring it into compliance with the requirements of NFPA 805 (SE Section 2.8.1).

VFDR #	VFDR Description	Component (Cables)
TB-01	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the MFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	1FDWPU0001, 1FDWPU0002 - Main Feedwater Pumps
TB-02	The EFW Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the EFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	1FDWPU0003, 1FDWPU0004, 1FDWPU0005 - EFW Pumps
TB-03	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being repositioned. Fire damage to cables for 1HP VA0939 may cause this valve to spuriously open prematurely, fail to open or spuriously close once opened. Valve 1HP VA0939 may suffer IN 92-18 damage.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
TB-04	This normally closed valve isolates the flow path from the BWST to the suction of the HPI Pumps. This valve is required to be open to supply borated water from the BWST to the HPI pump for RC boration and inventory control, and seal injection to the RCPs. Prior to the transfer of the power supply to the PSW system, fire damage to cables may prevent this valve from opening or cause the valve to spuriously close and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1HP VA0024 - HPI BWST Suction MOV
TB-05	This normally throttled, required closed valve provides normal RC makeup flow for pressurizer level control. This valve is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor	1HP VA0120 - RC Volume Control AOV

VFDR #	VFDR Description	Component (Cables)
	of the TB fire area. Fire induced cable damage may result in spurious opening of this valve, resulting in an uncontrolled increase in RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	
TB-06	The wall separating the TB and AB is not three hour rated as required by NFPA 805, Section 3.11.1 and all the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	TB / AB Wall
TB-07	This normally closed, required closed valve isolates the flow path from the LDST to the containment sump. This valve is required to remain closed to prevent diverting BWST inventory to the containment sump via HPI pump recirculation to the LDST. Fire damage to cables for electrical equipment supplying power to this valve may cause this valve to spuriously open and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1HP VA0940 - LDST to Emergency Sump MOV
TB-08	The 1C High Pressure Injection Pump is required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1HPIPU0003 - HPI Pump 1C
TB-09	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables may prevent these valves from being closed or cause them to spuriously open resulting in excess BWST inventory loss to the containment sump and a challenge to the Reactivity and Inventory Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	1LP VA0021, 1LP VA0022 - BWST Suction MOVs
TB-10	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves could result in overcooling and shrinkage of RC inventory and challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage	1MS VA0017 1MS VA0024 1MS VA0026 1MS VA0033 1MS VA0035 1MS VA0036 1MS VA0076 1MS VA0079 1MS VA0082 1MS VA0084 – SG Isolation MOVs
TB-11	Pressurizer heaters are required for RC pressure control. Pressurizer heater bank 2 is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor of the TB fire area. Fire damage to cables may result in the spurious operation of this heater resulting in an uncontrolled increase in RC pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC HE0002 - Pressurizer Heater Bank 2 (Groups B & D)
TB-13	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the PSW system. Fire damage to cables may result in an inability to secure the RCPs or result in a spurious pump start. This will place the unit in an unanalyzed condition and challenge the DHR Nuclear Safety Performance Criterion.	1RC PU0001 1RC PU0002 1RC PU0003 1RC PU0004 - RCPs
TB-14	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power

VFDR #	VFDR Description	Component (Cables)
	heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	Transfer Switches
TB-15	These normally closed, required closed valves isolate flow paths from the MSHs . Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	1SD VA0027, 1SD VA0290, 1SD VA0418, 1SD VA0419, 1SD VA0420, 1SD VA0421 - SG Isolation MOVs
TB-16	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the MFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	2FDWPU0001, 2FDWPU0002 - Main Feedwater Pumps
TB-17	The EFW Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the EFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	2FDWPU0003, 2FDWPU0004, 2FDWPU0005 - EFW Pumps
TB-18	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being repositioned. Fire damage to cables for 2HP VA0939 may cause this valve to spuriously open prematurely, fail to open or spuriously close once opened. Valve 2HP VA0939 may suffer IN 92-18 damage.	2HP VA0023 - HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
TB-19	This normally closed valve isolates the flow path from the BWST to the suction of the HPI Pumps. This valve is required to be open to supply borated water from the BWST to the HPI pump for RC boration and inventory control, and seal injection to the RCPs. Prior to the transfer of the power supply to the PSW system, fire damage to cables may prevent this valve from opening or cause the valve to spuriously close and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2HP VA0024 - HPI BWST Suction MOV
TB-20	This normally throttled, required closed valve provides normal RC makeup flow for pressurizer level control. This valve is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor of the TB fire area. Fire induced cable damage may result in spurious opening of this valve, resulting in an uncontrolled increase in RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2HP VA0120 - RC Volume Control AOV
TB-22	This normally closed, required closed valve isolates the flow path from the LDST to the containment sump. This valve is required to remain closed to prevent diverting BWST inventory to the containment sump via	2HP VA0940 - LDST to Emergency Sump MOV

VFDR #	VFDR Description	Component (Cables)
	HPI pump recirculation to the LDST. Fire damage to cables for electrical equipment supplying power to this valve may cause this valve to spuriously open and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	
TB-23	The 2C High Pressure Injection Pump is required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2HPIPU0003 - HPI Pump 2C
TB-24	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables may prevent these valves from being closed or cause them to spuriously open resulting in excess BWST inventory loss to the containment sump and a challenge to the Reactivity and Inventory Control Nuclear Safety Performance Criteria. These valves may suffer IN 92-18 damage.	2LP VA0021, 2LP VA0022 - BWST Suction MOVs
TB-25	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves could result in overcooling and shrinkage of RC inventory and challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
TB-26	Pressurizer heaters are required for RC pressure control. Pressurizer heater bank 2 is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor of the TB fire area. Fire damage to cables may result in the spurious operation of this heater resulting in an uncontrolled increase in RC pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC HE0002 - Pressurizer Heater Bank 2 (Groups B & D)
TB-28	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the PSW system. Fire damage to cables may result in an inability to secure the RCPs or result in a spurious pump start. This will place the unit in an unanalyzed condition and challenge the DHR Nuclear Safety Performance Criterion.	2RC PU0001, 2RC PU0002, 2RC PU0003, 2RC PU0004 - RCPs
TB-29	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
TB-30	These normally closed, required closed valves isolate flow paths from the MSHs . Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	2SD VA0027, 2SD VA0290, 2SD VA0418, 2SD VA0419, 2SD VA0420, 2SD VA0421 - SG Isolation MOVs
TB-31	The Main Feedwater (MFW) Pumps are required to be off to isolate MFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the MFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	3FDWPU0001, 3FDWPU0002 - Main Feedwater Pumps

VFDR #	VFDR Description	Component (Cables)
TB-32	The EFW Pumps are required to be off to isolate EFW to the SGs. Fire damage to cables may result in an inability to secure the pumps or may result in a spurious pump start. Spurious operation of the EFW Pumps could result in overfill of the SGs, overcooling of the RCS and a challenge to the DHR Nuclear Safety Performance Criterion.	3FDWPU0003, 3FDWPU0004, 3FDWPU0005 - EFW Pumps
TB-33	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Fire damage to cables for electrical equipment supplying power to these valves may prevent the valves from being repositioned.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
TB-35	This normally throttled, required closed valve provides normal RC makeup flow for pressurizer level control. This valve is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor of the TB fire area. Fire induced cable damage may result in spurious opening of this valve, resulting in an uncontrolled increase in RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3HP VA0120 - RC Volume Control AOV
TB-37	The 3C High Pressure Injection Pump is required to be off to prevent an uncontrolled increase in RC inventory. Fire damage to cables may result in spurious pump start and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3HPIPU0003 - HPI Pump 3C
TB-38	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and a challenge to the Inventory and Reactivity Control Nuclear Safety Performance Criteria.	3LP VA0021, 3LP VA0022 - BWST Suction MOVs
TB-39	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables may prevent these valves from being closed or may result in spurious opening of the valves. The failure to close these valves or the spurious opening of the valves could result in overcooling and shrinkage of RC inventory and challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
TB-40	Pressurizer heaters are required for RC pressure control. Pressurizer heater bank 2 is normally controlled from the main control room, but an alternate control station is provided in the Auxiliary Shutdown Panel located on the operating floor of the TB fire area. Fire damage to cables may result in the spurious operation of this heater resulting in an uncontrolled increase in RC pressure which may challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC HE0002 - Pressurizer Heater Bank 2 (Groups B & D)
TB-42	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the PSW system. Fire damage to cables may result in an inability to secure the RCPs or result in a spurious pump start. This	3RC PU0001, 3RC PU0002, 3RC PU0003, 3RC PU0004 - RCPs

VFDR #	VFDR Description	Component (Cables)
	will place the unit in an unanalyzed condition and challenge the DHR Nuclear Safety Performance Criterion.	
TB-43	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
TB-44	These normally closed, required closed valves isolate flow paths from the MSHs . Fire damage to cables may result in spurious opening of the valves. The spurious opening of the valves could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion. These valves may suffer IN 92-18 damage.	3SD VA0027, 3SD VA0418, 3SD VA0419, 3SD VA0420, 3SD VA0421 - SG Isolation MOVs
TB-45	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components located within the control complex and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
TB-46	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components located within the control complex and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
TB-47	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components located within containment and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling
TB-48	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components located within containment and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 2 Containment Cooling
TB-49	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components located within containment and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling
TB-51	This normally closed valve isolates the flow path from the discharge of the 2A HPI pump to the reactor coolant system (RCS). This valve is required to be open to supply borated water from the BWST to the RCS for RC boration and inventory control. Prior to the transfer of the power supply to the PSW system, fire damage to cables may prevent this valve from opening or cause the valve to spuriously close and challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2HP VA0026 - 2A HP Injection MOV
TB-52	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3LPIP 0345 - BWST Level Indication

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portions have been included below. The identified fire detection system modifications are to improve plant fire detection and fire brigade response time. The existing detectors within the TB are required with improvements to the following fire zones for general area and/or fire hazard detection: 3, 6, 12, 15, 19, 24, 25, 28, 29, 32, 33, 33A, 34, 34A, 35, 37, 39, 39A, 41.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
TB		**Turbine Building**										
	1	Unit 3 Lube Oil Purifier Area	No	No	No	Yes	No	Yes	No	Yes	No	No
	2	Unit 3 Electro-Hydraulic Control (EHC) Area	No	No	No	No	No	Yes	No	Yes	No	No
	3	Unit 3 Heater Bay Area	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	4	Unit 3 Turbine Driven EFDW Pump Area	No	No	No	No	No	Yes	No	Yes	No	No
	5	Unit 3 Condensate Booster Pump Area	No	No	No	No	No	No	No	No	No	No
	6	Unit 3 Main Feedwater Pump Area	Yes	No	Yes	No	No	Yes	No	Yes (MR)	No	No
	7	Unit 3 Motor Driven EFDW Pump Area	No	No	No	No	No	No	No	Yes	No	No
	8	Unit 3 Hotwell Pump & TB Sump Area	No	No	No	No	No	No	No	Yes	No	No
	9	Unit 3 Powdex/LSPW Pump Area	No	No	No	No	No	No	No	Yes	No	No
	10	Unit 2 Lube Oil Purifier Area	No	No	No	No	No	No	No	Yes	No	No
	11	Unit 2 EHC Area	No	No	No	No	No	No	No	Yes	No	No
	12	Unit 2 Heater Bay Area	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	13	Unit 2 Turbine Driven EFDW Pump Area	No	No	No	No	No	Yes	No	Yes	No	No
	14	Unit 2 Condensate Booster Pump Area	No	No	No	No	No	No	No	No	No	No
	15	Unit 2 Main Feedwater Pump	Yes	No	Yes	No	No	Yes	No	Yes (MR)	No	No

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System? E	R	D	S	Detection Provided?	Detection Required System? E	R	D	S
		Area										
	16	Unit 2 Motor Driven EFDW Pump Area	No	No	No	No	No	No	No	Yes	No	No
	17	Unit 2 HW Pump, LPSW Pump - B Area	No	No	No	No	No	Yes	No	Yes	No	No
	17A	Unit 2 HPSW Pump B Area	No	No	No	No	No	No	No	No	No	No
	18	Unit 2 Powdex, Backup IA Compressors	No	No	No	No	No	Yes	No	Yes	No	No
	19	Unit 1 Main Feedwater Pump Area	Yes	No	Yes	No	No	Yes	No	Yes (MR)	No	No
	20	Unit 1 Motor Driven EFDW Pump Area R	No	No	No	No	No	Yes	No	Yes	No	No
	21	Unit 1 HW Pump, LPSW Pump - A Area	No	No	No	No	No	Yes	No	Yes	No	No
	21A	Unit 1 HPSW Pump A Area	No	No	No	No	No	No	No	No	No	No
	22	Unit 1 Powdex Area	No	No	No	No	No	Yes	No	Yes	No	No
	22A	Unit 1 Lube Oil Storage House	No	No	No	No	No	No	No	No	No	No
	23	Unit 1 Condensate Booster Pump Area	No	No	No	No	No	No	No	No	No	No
	24	Unit 1 TDEFDW Pump, EHC, Oil Purifier	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	25	Unit 1 Feedwater Heaters & Drain Pumps	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	26	Unit 3 Moisture Separators (MS) B1 & B2	No	No	No	No	No	No	No	No	No	No
	27	Unit 3 Main Turbine (MT) Oil Tank and MS Stop & Control Valves	No	No	No	No	No	Yes	No	Yes	No	No
	28	Unit 3 Heater Bay Area, Moisture Separator Reheaters (MSRH) A1 & A2	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	29	Unit 3 4160 Volt Switchgear	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	30	Unit 2 MSs B1 & B2	No	No	No	No	No	No	No	No	No	No
	31	Unit 2 MT Oil Tank and MS Stop & Control Valves	No	No	No	No	No	Yes	No	Yes	No	No
	32	Unit 2 Heater Bay Area, MSRH A1 & A2	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	33	Unit 2 6900/4160	No	No	No	No	No	Yes	No	Yes	No	No

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
		Volt Switchgear								(MR)		
	33A	Unit 2 Power Batteries	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	34	Unit 1 6900/4160 Volt Switchgear	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	34A	Unit 1 Power Batteries	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	35	Unit 1 Heater Bay Area, MSRH A1 & A2	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	36	Unit 1 MT Oil Tank and MS Stop & Control Valves	No	No	No	No	No	Yes	No	Yes	No	No
	37	Unit 2 2X11, 2X11A, 3X5, 3X6 Area	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	38	Unit 3 Main Turbine, Turbine Flr, Offices	No	No	No	No	No	Yes	No	Yes	No	No
	39	Unit 3 Heater Bay & Upper Surge Tanks	No	No	No	No	No	Yes	No	No	Yes (MR)	No
	39A	Unit 3 Power Batteries	No	No	No	No	No	Yes	No	Yes (MR)	No	No
	40	Unit 2 Main Turbine, Turbine Flr, Offices	No	No	No	No	No	Yes	No	Yes	No	No
	41	Unit 2 Heater Bay & Upper Surge Tanks	No	No	No	No	No	Yes	No	No	Yes (MR)	No
	42	Unit 1 Main Turbine, Turbine Flr, Offices	No	No	No	No	No	Yes	No	Yes	No	No
	43	Unit 1 Heater Bay & Upper Surge Tanks	No	No	No	No	No	No	No	No	No	No
	44	Unit 1 TB Truck Receiving Bay	No	No	No	No	No	No	No	No	No	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S – Separation Criteria: Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR-Modification Required Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area TB Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area TB meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- No exemptions or licensing actions from the pre-transition fire protection requirements were required.

- Forty-five VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned/ implemented to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

  a. Modifications for TB are required to the following zones for general area and/or fire hazard detection: 3, 6, 12, 15, 19, 24, 25, 28, 29, 32, 33, 33A, 34, 34A, 35, 37, 39, 39A, 41. These detection modifications are to improve plant fire detection and fire brigade response time.

  b. A modification (VFDR TB-06) to the wall separating the AB and the TB requires modification to upgrade to a three hour fire barrier. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.

Fire Area WP1, Unit 1 West Penetration Room

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited two previously approved exemptions from the existing fire protection requirements. The licensee utilized the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward these exemptions, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of each exemption is provided in the table below.

Exemption / Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple. • Containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.
Appendix R Exemption, AB Lack of three hour fire rated barrier	Presented justification for the lack of three hour fire barriers because: • Low combustible loading in pipe tunnel access area. • Fire propagation path is circuitous, consisting of several unrated barriers and open areas. • If a fire were to occur, it would develop slowly. • Fire brigade may use portable extinguishers, manual hose stations, or a fire hose supplied from a nearby fire hydrant. In conclusion, although the exact number and configuration of combustibles may have changed over time, the bases for previous acceptance remain valid as substantiated by field walkdown.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)
Fire Area WP1 has a total of 14 VFDRs, which are provided in the table below. All but two of these VFDRs are variances NFPA 805, Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6), and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection for the Unit 1 Purge Inlet Room (Fire Zone 120) is required to meet the risk acceptance criteria. The Fire PRA makes assumptions regarding the time of fire discovery, fire brigade notification, and brigade manual suppression. These assumptions determine the impact of the fire, including the likelihood of a HGL being formed in the compartment. Specifically, the Fire PRA is based on a fire brigade response time of 20 minutes or less. The existing room detection system coverage of the general area and/or hazard necessary for this assumption to be valid was not considered sufficient to conservatively meet the risk criteria. Therefore, modification to the fire detection system in the Purge Inlet Room is required to support the fire risk analysis assumption of 20 minute brigade response time.

- General area and/or hazard detection associated with the Unit 1 West Penetration Pen Room (Fire Zone 107) and the Unit 1 Cask Decon Tank Room (Fire Zone 97) are required to meet the DID criteria.

Based on the reliance on fire detectors in the Unit 1 West Penetration Room (Fire Area WP1) to meet the risk and DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation. Improvements of general area and/or fire hazard detection are required for the Unit 1 West Penetration Pen Room (Fire Zone 107), the Unit 1 Cask Decon Tank Room (Fire Zone 97), and the Unit 1 Purge Inlet Room (Fire Zone 120) (SE Section 2.8.1).

Two of the 14 VFDRs, WP1-04 and WP1-10, are variances from NFPA 805 Section 4.2.3 (separation issue) that will be corrected with a plant modification. According to the LAR, the wall separating the Unit 1 Purge Inlet Room from the SFP Area and the wall separating the AB (Fire Area AB) from the Unit 1 West Penetration Room (Fire Area WP1) are not currently three hour fire rated as required by NFPA 805, Section 3.11.1, and all of the penetrations in the walls do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These walls are credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make modifications to the wall to bring it into compliance with the requirements of NFPA 805, Section 2.8.1.

VFDR #	VFDR Description	Component (Cables)

VFDR #	VFDR Description	Component (Cables)
WP1-01	RB pressure instrumentation is required for process monitoring and diagnosis of plant transients. Fire induced cable damage may result in a loss of RB pressure indication resulting in the inability of the operator to monitor and control this parameter from the MCR and challenge the Process Monitoring Nuclear Safety Performance Criterion.	1BS P 0011 - RB Pressure Indication
WP1-02	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 1A and 1B trains of Emergency Feedwater. Fire-induced cable damage may result in the spurious opening of this valve, a diversion of flow to either the 1A or 1B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	1CCWVA0269 - SG A FDW Control MOV
WP1-03	This normally open, required open valve is located in the EFW flow path to the 1B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 1B SG. The subsequent decrease in SG shell temperature may result in 1B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1FDWVA0347 - SG B Inlet MOV
WP1-04	The areas separating the Unit 1 Purge Inlet Room and SFP area is not three hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
WP1-05	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
WP1-07	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	1LP VA0021, 1LP VA0022 - BWST Suction MOVs
WP1-08	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire-induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance criterion.	1LPIP 0345 - BWST Level Indication
WP1-09	These normally open, required closed valves isolate flow paths from the MSHs. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
WP1-10	The wall separating the AB and the West penetration room does not have a	AB / West

VFDR #	VFDR Description	Component (Cables)
	fire-resistance rating required by NFPA 805, Section 3.11.2 and all the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Penetration Room Separation
WP1-11	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches
WP1-12	These normally closed, required closed valves isolate flow paths from the RCS to containment. A series of potential hot shorts within the terminal box of the electrical penetration may spuriously open the reactor head vent and hot leg vent valves. The spurious opening of these valves may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	1RC VA0157, 1RC VA0159 - RC Hot Leg and Head Vent Valves
WP1-13	These normally closed valves isolate the flow path from the RCS to containment. These valves are required to open to provide an RC letdown flow path. Fire induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	1RC VA0159, 1RC VA0160 - RC Head Vent Valves
WP1-15	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
WP1-16	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portions have been included below. These identified fire detection system modifications are to improve plant fire detection and fire brigade response time.

Fire Area	Fire Zone	Zone Description	Auto Suppression Required?	Suppression Required System?				Detection Required?	Detection Required System?			
				E	R	D	S		E	R	D	S
WP1		Unit 1 West Penetration Room										
WP1	97	Unit 1 Cask Decon Tank Room	No	No	No	No	No	Yes	Yes	No	No	No
WP1	107	Unit 1 West Penetration Pen Room	No	No	No	No	No	Yes	No	No	Yes (MR)	No

WP1	120	Unit 1 Purge Inlet Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No	

Legend:	
E - EEEE/LA:	Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

<u>Fire Area WP1 Conclusion</u>

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area WP1 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- Two exemptions from the pre-transition fire protection requirements were evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Fourteen VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned/ implemented to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

  a. Improve general area and/or hazard detection for Unit 1 Purge Inlet Room, Unit 1 West Penetration Pen Room and Unit 1 Cask Decon Tank Room.
     - o In order to take credit for evaluations in the fire area, the following barrier modifications are required:
       - o AB / Unit 1 West Penetration Room separation
       - o Unit 1 Purge Inlet Room / SFP Area separation

Fire Area WP2, Unit 2 West Penetration Room

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions.

Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited two previously approved exemptions from the existing fire protection requirements. The licensee utilized the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward these exemptions, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of each exemption is provided in the table below.

Exemption / Licensing Action	Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three hour fire rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.
Appendix R Exemption, AB Lack of three hour fire rated barrier	Presented justification for the lack of three hour fire barriers because: • Low combustible loading in pipe tunnel access area. • Fire propagation path is circuitous, consisting of several unrated barriers and open areas. • If a fire were to occur, it would develop slowly. • Fire brigade may use portable extinguishers,	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.

	manual hose stations, or a fire hose supplied from a nearby fire hydrant. In conclusion, although the exact number and configuration of combustibles may have changed over time, the bases for previous acceptance remain valid as substantiated by field walkdown.	

Variation from Deterministic Requirements (VFDRs)

Fire Area WP2 has a total of 14 VFDRs, which are provided in the table below. All but two of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with an FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection for the Unit 2 Purge Inlet Room (Fire Zone 117) is required to meet the risk acceptance criteria. The Fire PRA makes assumptions regarding the time of fire discovery, fire brigade notification, and brigade manual suppression. These assumptions determine the impact of the fire, including the likelihood of an HGL being formed in the compartment. Specifically, the Fire PRA is based on a fire brigade response time of 20 minutes or less. The existing room detection system coverage of the general area and/or hazard necessary for this assumption to be valid was not considered sufficient to conservatively meet the risk criteria. Therefore, modification to the fire detection system in the Purge Inlet Room is required to support the fire risk analysis assumption of a 20-minute brigade response time.

- General area and/or hazard detection associated with the Unit 2 West Penetration Pen Room (Fire Zone 102) and the Unit 2 Cask Decon Tank Room (Fire Zone 91) are required to meet the DID criteria.

Based on the reliance on fire detectors in the West Penetration Room WP2 Fire Area to meet the risk and DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation. Improvements of general area and/or fire hazard detection are required for the Unit 2 West Penetration Pen Room (Fire Zone 102), the Unit 2 Purge Inlet Room (Fire Zone 117), and the Unit 2 Cask Decon Tank Room (, 2.8.1).

Two of the 14 VFDRs, WP2-04 and WP2-10, are variances from NFPA 805 Section 4.2.3, (separation issue) that will be corrected with a plant modification. According to the LAR, the wall separating the Unit 2 Purge Inlet Room from the SFP Area and the wall separating the AB (Fire Area AB) from the Unit 2 West Penetration Room (Fire Area WP2) are not currently a three-hour rated wall as required by NFPA 805, Section 3.11.1, and all of the penetrations in the walls do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These walls are credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make modifications to the wall to bring it into compliance with the requirements of NFPA 805, Section 2.8.1.

VFDR #	VFDR Description	Component (Cables)
WP2-02	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 2A and 2B trains of Emergency Feedwater. Fire-induced cable damage may result in the spurious opening of this valve, a diversion of flow to either the 2A or 2B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	2CCWVA0269 - SG A FDW Control MOV
WP2-03	This normally open, required open valve is located in the EFW flow path to the 2B SG. Fire-induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 2B SG. The subsequent decrease in SG shell temperature may result in 2B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2FDWVA0347 - SG B Inlet MOV
WP2-04	The areas separating the Unit 2 Purge Inlet Room and SFP area is not three-hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
WP2-05	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	2HP VA0023 - HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
WP2-07	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	2LP VA0021, 2LP VA0022 - BWST Suction MOVs
WP2-08	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire-induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance criterion.	2LPIP 0345 - BWST Level Indication
WP2-09	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
WP2-10	The wall separating the AB and the West penetration room does not have a fire-resistance rating required by NFPA 805, Section 3.11.2 and all the penetrations in the wall do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	AB / West Penetration Room Separation

VFDR #	VFDR Description	Component (Cables)
WP2-11	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
WP2-13	This normally closed, required closed valve isolates the flow path from the RCS to the Quench Tank. Fire-induced cable damage may result in the spurious opening of the PORV causing a loss of RC inventory and RC subcooling. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0066 - Pressurizer Power Operated Relief Valve
WP2-14	These normally closed valves isolate the flow path from the RCS to containment. These valves are required to open to provide an RC letdown flow path. Fire-induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	2RC VA0159, 2RC VA0160 - RC Head Vent Valves
WP2-15	This normally closed, required closed valve isolates the flow path from the RCS to containment. A series of potential hot shorts within the terminal box of the electrical penetration may spuriously open the reactor head vent valve. The spurious opening of this valve may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	2RC VA0159 - RC Head Vent Valve
WP2-17	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
WP2-18	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 2 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portions have been included below. These identified fire detection system modifications are to improve plant fire detection and fire brigade response time.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Required?	Detection Required System?			
				E	R	D	S		E	R	D	S
WP2		Unit 2 West Penetration Room										
WP2	91	Unit 2 Cask Decon Tank Room	No	No	No	No	No	Yes	Yes	No	No	No

WP2	102	Unit 2 West Penetration Pen Room	No	No	No	No	No	Yes	No	No	Yes (MR)	No
WP2	117	Unit 2 Purge Inlet Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria: Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR- Modification Required Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area WP2 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. An FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area WP2 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three-hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- Two exemptions from the pre-transition fire protection requirements were evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Fourteen VFDRs were identified, evaluated through the performance of an FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned/ implemented to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

Improve general area and/or hazard fire detection for Unit 2 Purge Inlet Room, Unit 2 West Penetration Pen Room, and Unit 2 Cask Tank Decon Room.

  a. In order to take credit for evaluations in the fire area, the following barrier modifications are required:

AB / Unit 2 West Penetration Room separation

- Unit 2 Purge Inlet Room / SFP Area separation

Fire Area WP3, Unit 3 West Penetration Room

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805, Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805, Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions. Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

Based on the information provided in the LAR, the licensee credited two previously approved exemptions from the existing fire protection requirements. The licensee utilized the process described in LAR Section 4.2.3, "Licensing Action Transition," and Attachment K, "Licensing Action Review," to carry forward these exemptions, which requires a determination of the basis of acceptability and a determination that the basis of the acceptability is still valid. The NRC staff's evaluation of each exemption is provided in the table below.

Exemption / Licensing Action	Licensee's Statement on Basis and Continuing Validity	NRC Staff Evaluation
Appendix R Exemption, RB Unrated Containment Mechanical Penetrations	Provides the following justification for the lack of three-hour fire-rated pipe penetrations: • RB walls serve as a substantial heat sink. • Combustible loading near penetrations is low. • Mechanical pipe penetrations are designed to meet multiple containment integrity criteria and are substantial. • Large room volumes on both sides dissipate heat from a fire away from penetration area. The bases for previous acceptance remain valid.	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains valid, the NRC staff finds this acceptable.
Appendix R Exemption, AB Lack of three hour fire rated barrier	Presented justification for the lack of three-hour fire barriers because: • Low combustible loading in pipe tunnel access area. • Fire propagation path is circuitous, consisting of	Based on the previous NRC staff approval of this exemption and the statement by the licensee that the basis remains

	several unrated barriers and open areas. • If a fire were to occur, it would develop slowly. • Fire brigade may use portable extinguishers, manual hose stations, or a fire hose supplied from a nearby fire hydrant. In conclusion, although the exact number and configuration of combustibles may have changed over time, the bases for previous acceptance remain valid as substantiated by field walkdown.	valid, the NRC staff finds this acceptable.

Variation from Deterministic Requirements (VFDRs)

Fire Area WP3 has a total of 12 VFDRs, which are provided in the table below. All but one of these VFDRs are variances from NFPA 805 Section 4.2.3 (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee's FRE determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- General area and/or hazard detection for the Unit 3 Purge Inlet Room (Fire Zone 114) is required to meet the risk acceptance criteria. The Fire PRA makes assumptions regarding the time of fire discovery, fire brigade notification, and brigade manual suppression. These assumptions determine the impact of the fire, including the likelihood of a HGL being formed in the compartment. Specifically, the Fire PRA is based on a fire brigade response time of 20 minutes or less. The existing room detection system coverage of the general area and/or hazard necessary for this assumption to be valid was not considered sufficient to conservatively meet the risk criteria. Therefore, modification to the fire detection system in the Purge Inlet Room is required to support the fire risk analysis assumption of 20 minute brigade response time.

- General area and/or hazard detection associated with the Unit 3 West Penetration Pen Room (Fire Zone 98) and the Unit 3 Cask Decon Tank Room (Fire Zone 87) are required to meet the DID criteria.

Based on the reliance on fire detectors in the West Penetration Room WP3 Fire Area to meet the risk and DID criteria, the licensee has committed to make modifications to the fire detection system, which may include fire detector upgrades and/or new installation. Improvements of general area and/or fire hazard detection are required for the Unit 3 West Penetration Pen Room (Fire Zone 98), the Unit 3 Purge Inlet Room (Fire Zone 114), and the Unit 3 Cask Decon Tank Room (Fire Zone 87) (SE Section 2.8.1).

One of the 12 VFDRs, WP3-03 is a variance from NFPA 805, Section 4.2.3, (separation issue) that will be corrected with a plant modification. According to the LAR, the wall separating the Unit 3 Purge Inlet Room from the SFP Area is not currently a three hour rated wall as required by NFPA 805, Section 3.11.1, and all of the penetrations in the walls do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These walls are credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make modifications to the wall to bring it into compliance with the requirements of NFPA 805, Section 2.8.1.

VFDR #	VFDR Description	Component (Cables)
WP3-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 3A and 3B trains of EFW . Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 3A or 3B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	3CCWVA0269 - SG A FDW Control MOV
WP3-02	This normally open, required open valve is located in the EFW flow path to the 3B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 3B SG. The subsequent decrease in SG shell temperature may result in 3B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	3FDWVA0347 - SG B Inlet MOV
WP3-03	The areas separating the Unit 3 Purge Inlet Room and SFP area is not three hour rated as required by NFPA 805, Section 3.11.1 and the penetrations (seals and doors) do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. These barriers are credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Purge Inlet Room / SFP Area
WP3-04	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
WP3-06	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and result in a diversion of BWST inventory to the containment sump via the LPI system. In addition, an inadvertent ES actuation could result in a diversion of BWST inventory to the containment sump via the RBS system. A loss of BWST inventory could challenge the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria.	3LP VA0021, 3LP VA0022 - BWST Suction MOVs
WP3-07	These normally open, required closed valves isolate flow paths from the MSHs . Although unaffected by fire, the power supplies to these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
WP3-09	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
WP3-11	This normally closed, required closed valve isolates the flow path from the RCS to the Quench Tank. Fire induced cable damage may result in spurious opening of the PORV causing a loss of RC inventory and RC subcooling. This could challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0066 - Pressurizer Power Operated Relief Valve

VFDR #	VFDR Description	Component (Cables)
WP3-12	These normally closed, required closed valves isolate flow paths from the RCS to containment. A series of potential hot shorts within the terminal box of the electrical penetration may spuriously open the reactor head vent and hot leg vent valves. The spurious opening of these valves may result in a loss of RC inventory and challenge the Inventory and Pressure Control Nuclear Safety Performance Criteria.	3RC VA0157, 3RC VA0159 - RC Hot Leg and Head Vent Valves
WP3-13	These normally closed valves isolate the flow path from the RCS to containment. These valves are required to open to provide an RC letdown flow path. Fire induced cable damage may prevent these valves from being opened resulting in the lifting of the pressurizer safety relief valves and a challenge to the Inventory Control Nuclear Safety Performance Criterion.	3RC VA0159, 3RC VA0160 - RC Head Vent Valves
WP3-15	Although unaffected by fire, the power supplies for the station HVAC system are not credited following a fire in this fire area and a loss of power may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
WP3-16	Fire damage to cables may result in a loss of power to the containment cooling system and may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components. This could challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

Detection & Suppression Systems Required to Meet the Nuclear Safety Performance Criteria

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portions have been included below.

Partial detection is installed over electrical penetrations. Purge Inlet Room general area and/or hazard detection is required for risk criteria. West Penetration Room general area and/or hazard detection is required for DID. Modification is required to improve general area and/or hazard detection for Purge Inlet Room and West Penetration Room. These detection modifications are to improve plant fire detection and fire brigade response time.

Fire Area	Fire Zone	Zone Description	Auto Suppression Required?	Suppression Required System?				Detection Required?	Detection Required System?			
				E	R	D	S		E	R	D	S
WP3		Unit 3 West Penetration Room										
WP3	87	Unit 3 Cask Decon Tank Room	No	No	No	No	No	Yes	Yes	No	No	No
WP3	98	Unit 3 West Penetration Pen Room	No	No	No	No	No	Yes	No	No	Yes (MR)	No
WP3	114	Unit 3 Purge Inlet Room	No	No	No	No	No	Yes	No	Yes (MR)	No	No
Legend:												

E - EEEE/LA:	Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk:	Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth:	Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S - Separation Criteria:	Systems required for Chapter 4 Separation Criteria in (Section 4.2.3)
MR - Modification Required	Systems are committed to be modified as indicated in LAR Table 4-4 and Attachment S

Fire Area WP3 Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area WP3 meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805, Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- Two exemptions from the pre-transition fire protection requirements were evaluated and found to be valid and applicable under the NFPA 805 RI/PB FPP.

- Twelve VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned/ implemented to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

  a. To improve general area and/or hazard detection for the Unit 3 Purge Inlet Room, Unit 3 West Penetration Pen Room and Unit 3 Cask Decon Tank Room.
  b. Fire barrier separating the Unit 3 Purge Inlet Room / SFP Area is required.

Fire Area YARD, Yard

The licensee analyzed this fire area using the FRE approach in accordance with NFPA 805 Section 4.2.4.2, but also used deterministic simplifying assumptions in order to credit those portions of the facility design that met the deterministic requirements of NFPA 805 Section 4.2.3. The licensee identified the SSCs necessary to meet the nuclear safety performance criteria in this fire area.

Evaluation of Fire Suppression Effects on Nuclear Safety Performance Criteria

The licensee stated in Attachment C, "NEI 04-02, Table B-3, Fire Area Transition," that safe and stable conditions can be achieved and maintained using equipment and cables outside of the area of fire suppression activity. Flooding of the suppression areas and discharge of suppression water to adjacent compartments is controlled and will not jeopardize achievement of safe and stable conditions.

Based on the information provided by the licensee in the NFPA 805 LAR, the NRC staff finds the licensee's evaluation of fire suppression effects on nuclear safety performance criteria acceptable because the results of the licensee's analysis indicate that fire suppression activities will not adversely affect achievement of the nuclear safety performance criteria.

Exemptions and Other Licensing Actions

The licensee did not credit any previously approved licensing actions or exemptions from the existing fire protection requirements.

Variation from Deterministic Requirements (VFDRs)

Fire Area YARD has a total of 28 VFDRs, which are provided in the table below. All but one of these VFDRs are variances from NFPA 805, Section 4.2.3, (separation issues) that were dispositioned with a FRE (SE Section 3.4.3). The licensee determined that these variances are acceptable based on 1) the change in CDF and LERF for the fire area and the total CDF and LERF for each unit meet the acceptance criteria of RG 1.174 (SE Section 3.4.6) and 2) adequate DID and SMs are maintained for each fire area (SE Section 3.4.2). This determination relies on the following fire protection systems and features to meet the acceptance criteria:

- Fire suppression for Transformers CT-1, CT-2, and CT-3 is required to meet the DID criteria.

- Pre-fire plans will be updated to include fire brigade guidance for protection of the TB wall, combustible controls will be established, and vehicle traffic controls will be established in the vicinity of the Fire Area TB wall, transformers, and trenches to meet the DID criteria (SE Section 2.9, Table 2.9-1, Item 9).

The licensee does not require any system or barrier modifications because reliance on existing transformer suppression systems is sufficient to meet the criteria of DID. Fire brigade guidance update, combustible controls, and traffic control in the vicinity of the TB wall will be required to be in place to meet the criteria of DID.

One of the 28 VFDRs, YARD-04, is a variance from NFPA 805 Section 4.2.3 (separation issue) that will be corrected with a plant modification. According to the LAR, the wall separating the east YARD and the tornado vents of the Blockhouse 1 & 2 exterior wall currently is not "adequate for the hazard" as required by NFPA 805, Section 3.11.1, and all of the penetrations in the wall do not have a fire resistance rating as required by NFPA 805 ,Section 3.11.3. This wall is credited for area separation using the deterministic approach of NFPA 805, Section 4.2.3. The licensee has committed to make a modification to install hinged steel covers/shields to the exterior side of the tornado vents to qualify the wall 'adequate for the hazard' thereby bringing it into compliance with the requirements of NFPA 805 (SE Section 2.8.1).

VFDR #	VFDR Description	Component (Cables)
YARD-01	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 1A and 1B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 1A or 1B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	1CCWVA0269 - SG A FDW Control MOV
YARD-02	This normally open, required open valve is located in the EFW flow path to the 1B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 1B SG. The subsequent decrease in SG shell temperature may result in 1B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	1FDWVA0347 - SG B Inlet MOV
YARD-03	Normally open valve 1HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 1HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 1HP VA0939 and closing 1HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	1HP VA0023 - HPI Normal Suction MOV, 1HP VA0939 - LDST to Emergency Sump MOV
YARD-04	The penetrations in the wall interfacing the east wall of Blockhouse 1 & 2 and the east yard do not have a fire resistance rating as required by NFPA 805, Section 3.11.3. This wall is credited for area separation in the deterministic approach of NFPA 805, Section 4.2.3.	Tornado Vents in Blockhouse 1 & 2 Building Wall
YARD-05	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and a challenge to the Inventory and Reactivity Control Nuclear Safety Performance Criteria.	1LP VA0021, 1LP VA0022 - BWST Suction MOVs
YARD-06	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance Criterion	1LPIP 0345 - BWST Level Indication
YARD-07	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	1MS VA0017, 1MS VA0024, 1MS VA0026, 1MS VA0033, 1MS VA0035, 1MS VA0036, 1MS VA0076, 1MS VA0079, 1MS VA0082, 1MS VA0084 - SG Isolation MOVs
YARD-08	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 1 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	1RC SXTRN001, 1RC SXTRN002 - Pressurizer Heaters PSW Power Transfer Switches

VFDR #	VFDR Description	Component (Cables)
YARD-10	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 2A and 2B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 2A or 2B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage	2CCWVA0269 - SG A FDW Control MOV
YARD-11	This normally open, required open valve is located in the EFW flow path to the 2B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 2B SG. The subsequent decrease in SG shell temperature may result in 2B SG exceeding its tube to shell differential temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	2FDWVA0347 - SG B Inlet MOV
YARD-12	Normally open valve 2HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 2HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 2HP VA0939 and closing 2HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	2HP VA0023 - HPI Normal Suction MOV, 2HP VA0939 - LDST to Emergency Sump MOV
YARD-14	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and a challenge to the Inventory and Reactivity Control Nuclear Safety Performance Criteria.	2LP VA0021, 2LP VA0022 - BWST Suction MOVs
YARD-15	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance Criterion.	2LPIP 0345 - BWST Level Indication
YARD-16	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	2MS VA0017, 2MS VA0024, 2MS VA0026, 2MS VA0033, 2MS VA0035, 2MS VA0036, 2MS VA0076, 2MS VA0079, 2MS VA0082, 2MS VA0084 - SG Isolation MOVs
YARD-17	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 2 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	2RC SXTRN001, 2RC SXTRN002, 2RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
YARD-19	This normally closed, required closed valve provides train separation by isolating the cross connect header between the 3A and 3B trains of Emergency Feedwater. Fire induced cable damage may result in spurious opening of this valve, a diversion of flow to either the 3A or 3B SGs, and a challenge to the DHR Nuclear Safety Performance Criterion. This valve may suffer IN 92-18 damage.	3CCWVA0269 - SG A FDW Control MOV
YARD-20	This normally open, required open valve is located in the EFW flow path to the 3B SG. Fire induced cable damage may result in spurious closing of this valve, isolating Protected Service Water flow to the 3B SG. The subsequent decrease in SG shell temperature may result in 3B SG exceeding its tube to shell differential	3FDWVA0347 - SG B Inlet MOV

VFDR #	VFDR Description	Component (Cables)
	temperature limit. This could challenge the Inventory Control and DHR Nuclear Safety Performance Criteria. This valve may suffer IN 92-18 damage.	
YARD-21	Normally open valve 3HP VA0023 is in the flow path from the LDST to the suction of the credited HPI pump. Normally closed valve 3HP VA0939 isolates the flow path from the LDST to the containment sump. Recirculation flow to the LDST during prolonged operation of the HPI pump at low flow conditions may result in an increase in temperature of LDST contents to the operability limit of the HPI pump. The contents of the LDST must be diverted to the containment sump by opening 3HP VA0939 and closing 3HP VA0023 prior to the operability limit of the HPI pump being exceeded to prevent challenging the Reactivity, Inventory and Pressure Control Nuclear Safety Performance Criteria. Although unaffected by fire, the power supplies for these valves are not credited following a fire in this fire area and a loss of power may prevent these valves from being repositioned.	3HP VA0023 - HPI Normal Suction MOV, 3HP VA0939 - LDST to Emergency Sump MOV
YARD-23	These normally open, required closed valves isolate the flow path from the BWST to the LPI Pumps, RBS Pumps, and containment sump. Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed resulting in excess BWST inventory loss to the containment sump and a challenge to the Inventory and Reactivity Control Nuclear Safety Performance Criteria.	3LP VA0021, 3LP VA0022 - BWST Suction MOVs
YARD-24	BWST level indication is required to monitor the performance of the reactivity and inventory control systems. Fire induced cable damage may result in loss of BWST level indication and challenge the Process Monitoring Nuclear Safety Performance Criterion.	3LPIP 0345 - BWST Level Indication
YARD-25	These normally open, required closed valves isolate flow paths from the MSHs . Fire damage to cables for electrical equipment supplying power to these valves may prevent these valves from being closed and could result in overcooling and shrinkage of RC inventory. This could challenge the DHR Nuclear Safety Performance Criterion.	3MS VA0017, 3MS VA0024, 3MS VA0026, 3MS VA0033, 3MS VA0035, 3MS VA0036, 3MS VA0076, 3MS VA0079, 3MS VA0082, 3MS VA0084 - SG Isolation MOVs
YARD-26	The Reactor Coolant Pumps (RCPs) are required off when SSD is being accomplished by the PSW system. Unit 3 6900V RCP SWGR is located in Fire Area YARD. Fire damage to cables may result in an inability to secure the RCPs or result in a spurious pump start. This will place the unit in an unanalyzed condition and challenge the DHR Nuclear Safety Performance Criterion.	3RC PU0001, 3RC PU0002, 3RC PU0003, 3RC PU0004 - RCPs
YARD-27	Pressurizer heaters are required for RC pressure control. The heaters receive non-credited power from Unit 3 and credited power from a PSW system power supply. The transfer of credited power to the pressurizer heaters requires a recovery action. Failure to transfer credited power to the heaters could challenge the Pressure Control Nuclear Safety Performance Criterion.	3RC SXTRN001, 3RC SXTRN002, 3RC SXTRN003 - Pressurizer Heaters PSW Power Transfer Switches
YARD-29	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Units 1 & 2 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Units 1 & 2 Control Complex Cooling
YARD-30	Fire damage to cables for electrical equipment supplying power to the station HVAC system may result in the temperature inside the Unit 3 control complex exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Control Complex Cooling
YARD-31	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 1 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 1 Containment Cooling
YARD-32	Fire damage to cables for electrical equipment supplying power to the containment	Unit 2

VFDR #	VFDR Description	Component (Cables)
	cooling system may result in the temperature inside the Unit 2 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Containment Cooling
YARD-33	Fire damage to cables for electrical equipment supplying power to the containment cooling system may result in the temperature inside the Unit 3 RB exceeding the operability limit of SSD components and challenge the Vital Auxiliaries Nuclear Safety Performance Criterion.	Unit 3 Containment Cooling

Note: The additional risk added because of these VFDRs, as determined from the FRE for this fire area, is provided in SE Table 3.5.

Recovery Actions (RAs)

The licensee did not identify any recovery actions required for this fire area.

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

The licensee performed an evaluation of the fire detection and suppression systems in this area. The results of the evaluation were documented in LAR Table 4-4 and the applicable portions have been included below. Partial fire suppression is installed in the YARD fire area. Suppression for transformers CT-1, CT-2, and CT-3 is required for DID. Suppression on transformers 1T, 2T, and 3T is required by Engineering Evaluation. Partial detection is installed in YARD fire area. Detection for Unit 3 RCP SWGR is required for DID.

Fire Area	Fire Zone	Zone Description	Auto Suppression Provided?	Suppression Required System?				Detection Provided?	Detection Required System?			
				E	R	D	S		E	R	D	S
YARD		Yard										
YARD	SYDYARD	230 KV Switchyard and Relay House	No	No	No	No	No	No	No	No	No	No
YARD	TRENCH	Cable Trench T-100	No	No	No	No	No	No	No	No	No	No
YARD	Yard- East	Yard Area – East	Yes	Yes	No	Yes	No	Yes	No	No	Yes	No
YARD	Yard - West	Yard Area- West	No	No	No	No	No	No	No	No	No	No

Legend:
E - EEEE/LA: Systems required for acceptability of EEE Evaluations / NRC approved Licensing Action (Section 2.2.7)
R - Risk: Systems required to meet the Risk Criteria for the PB Approach (Section 4.2.4)
D - Defense-in-Depth: Systems required to maintain adequate balance of Defense-in-Depth for a PB Approach (Section 4.2.4.2)
S – Separation Criteria: Systems required for NFPA 805 Chapter 4 Separation Criteria in Section 4.2.3
MR- Modification Required Systems are committed to be modified as indicated in Table 4-4 and Attachment S of TR

Fire Area YARD Conclusion

The licensee has utilized the FRE PB approach to demonstrate the ability to meet the NFPA 805 nuclear safety performance criteria for this fire area. A FRE in accordance with NFPA 805, Section 4.2.4.2, in conjunction with deterministic methods for simplifying assumptions, was used in applying this approach.

Based on the information provided in the LAR, as supplemented, the NRC staff finds Fire Area YARD meets the nuclear safety goals, objectives, and performance criteria of NFPA 805. This conclusion is based on the following:

- Fire protection SSCs were evaluated in accordance with NFPA 805 Chapter 4 to determine which, if any, were required to meet the nuclear safety performance criteria. This evaluation included:

  a. The fire protection detection systems required to meet the nuclear safety performance criteria were documented.
  b. Fire Area boundaries were defined using three hour rated walls, ceilings and floors, including fire barriers, fire barrier penetrations, and through penetration fire stops and spatial separation.

- No exemptions or licensing actions from the pre-transition fire protection requirements were required.

- Twenty-eight VFDRs were identified, evaluated through the performance of a FRE, and either found to meet the risk acceptance criteria, as well as the requirements for DID and SMs, or modifications were planned / implemented to address the issue. The acceptability of the risk for this fire area is contingent on the risk reduction from the planned PSW modification (see SE Section 3.4 for a detailed discussion of the NRC staff's review of the adequacy of the FRE method used at ONS).

- This fire area did not require the use of recovery actions to meet the nuclear safety performance criteria.

- The following modifications were identified to address VFDRs:

  a. Modification to install hinged steel covers/shields to exterior side of tornado vents to support FRE of separation between Units 1 & 2 Block House and the East Yard.

Attachment E, Radioactive Release Tables

In order to assess whether the ONS FPP to be implemented under NFPA 805 meets the radioactive release performance criteria, the licensee reviewed the existing ONS pre-fire plans and fire brigade training materials. Pre-fire plans that address fire areas/zones where there is no possibility of radioactive materials being present were screened from further review. All other pre-fire plans were reviewed to ascertain whether existing engineering controls are adequate to ensure that radioactive materials (contamination) generated as a direct result of fire suppression activities are contained and monitored before release to unrestricted areas, such that the release would meet the NFPA 805 radioactive release performance criteria.

The licensee's review determined that existing engineering controls, such as drains and forced air ventilation, supplemented by pre-fire plans and fire brigade training, were adequate to meet the NFPA 805 radioactive release requirements. In addition, the licensee identified the need for monitoring and control of potentially contaminated run-off into non-contaminated areas and developed a new fire brigade instruction (SOG-16) to address this need, which has been incorporated into fire brigade training. Chemistry/radiation protection (RP) personnel are part of the responding fire brigade team. The licensee stated that current RP procedures and practices adequately describe how to monitor and control liquid and gaseous effluents from the site.

This attachment contains Table 3.6-1, "ONS Fire Areas and Their Compliance with the NFPA 805 Radioactive Release Performance Criteria," which summarizes, for each pre-fire plan, (1) the fire areas included in the pre-fire plan, (2) the engineered controls used to minimize radioactive releases generated from the combustion of radioactive materials or from fire suppression activities, and (3) the NRC staff evaluation of the adequacy of these engineered controls and fire brigade training in meeting the NFPA 805 Radioactive Release Performance Criteria.

Attachment E, Radioactive Release Tables
Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
TB	Turbine Building					
1	Lube Oil Purification Pad	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
2	EHC Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
3	Heater Bay Area	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
4	TDEFDW Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
5	Condensate Booster Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
6	Main Feedwater Pump Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
7	Motor-Driven Emergency Feedwater (MDEFDW) Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
8	Hotwell Pump and TB Sump Oil Skimmer	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
9	Powdex/LPSW Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
10	Unit 2 Lube Oil Purification	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
11	Unit 2 EHC Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
12	Unit 2 Heater Bay	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
13	Unit 2 TDEFDWP	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
14	Unit 2 Condensate Booster Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
15	Unit 2 Main Feedwater Pump Area	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
16	Unit 2 MDEFDWP	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
17	Unit 2 HWP,LPSW-B Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
17A	HPSW·B Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
18	Unit 2 Powdex, Backup 1A Compressor and Control Room Chillers	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
19	Unit 1 Main Feed Water Pump Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
20	Unit 1 MDEFDWP and Seal Oil Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
21	Unit 1 HWP,LPSW-A Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
21A	HPSW Pump A	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
22	Unit 1 Powdex Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
22A	Lube Oil Storage House	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
23	Unit 1 Condensate Booster Pump	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
24	Unit 1 TDEFDW Pump, EHC, Turbine and Lube Oil Purification	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
25	Unit 1 Feed Water Heater Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
26	Moisture Separators B1 & B2	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
27	Turbine Oil Tank and MS Stop and Control Valves	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
28	Heater Bay Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
29	4160 Switchgear	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
30	Unit 2 Moisture Separator B1 & B2	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
31	Unit 2 MS Stop and Control Valves and Main Turbine Oil Tank (MTOT)	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
32	Unit 2 Heater Bay Area	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
33	Unit 2 4160 Switchgear	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
33A	Unit 2 Power Batteries	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
34	Unit 1 6900/4160V Switchgear	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
34A	Unit 1 Power Batteries PA	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
35	Unit 1 Heater Bay Area	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
36	Unit 1 MS Stop Valves, MSRH's and MTOT	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
37	MCC 2X11, 2X11A, 3X5 and 3X6 Areas	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
38	Main Turbine and Offices	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
39	Heater Bay and Upper Surge Tanks	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
39A	Turbine Deck and Offices	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
40	Unit 2 Main TB Deck	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
41	Unit 2 Heater Bay and Upper Surge Tank	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
42	Unit 1 Main Turbine	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
43	Unit 1 Heater Bay and Upper Surge Tanks	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
44	TB Truck Receiving Bay	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
BH12 - 45	Units 1/2 Block House	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
CT4 - 46	CT-4 Block House	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
BH3 - 47	Unit 3 Block House	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
48	3A LPI and RBS Pumps	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, radiation indicating alarms (RIAs) monitor contamination levels of smoke	Based on the availability of engineered controls and fire brigade training for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
49	Unit 3 C LPI and B RBS Pumps	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
50	3C HPI Pump	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

Attachment E, Radioactive Release Tables
Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
50A	HPI, Spent resin, L/H AWT and Comp Drain Pumps	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
51	Purifications Demineralizer Room and Hatch Area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
52	Low Pressure Injection Pump's (LPIP's) 2A, 2C and Reactor Building Spray Pump (RBSP) 2A	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
53	LPIP's 1B, 2B and RBSP 1B & 2B	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
54	LPIP's 1A, 1C & RBSP 1A	0	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
55	HPIP's 1A & 1B	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
55A	HPIP's 1C & 2C	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
56	HPIP's 2A & 2B	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths, RIAs for smoke control monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
57	Purification & Deborating Demineralizer	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

Attachment E, Radioactive Release Tables
Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
58	1st Floor Hallway/Tank Room and 2nd Floor Tank Rooms	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
59	LPI Cooler, HPI Seal Return Coolers	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
60	LPI Hatch Area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
61	HPI Hatch area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
62	Waste Disposal Control Room	0	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
63	Letdown Storage Tank	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
64	Auxiliary Service Water and Switchgear	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
65	Unit 2 Decay Heat Cooler, Seal Supply Filter and CRD	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
66	HPI Hatch area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
67 & 70	Units 1&2 Hatch Area and Hot Machine Shop Tunnel	0		Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
68 & 72	Units 1 & 2 HPI Hatch area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
69	Units 1 & 2 Waste Control Panel	1		Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
71	Unit 2 Letdown Storage Tank and Letdown Filter	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
73	Unit 1 Letdown Storage Tank and Filter Room	0		Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

Attachment E, Radioactive Release Tables
Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
74	Unit 1 Post-Accident Liquid Sample (PALS) Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
75	Unit 1 LPI Cooler, Pipe Chase, CRD Filter Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
76	IA & 1B BHUT, CBAST and RC Bleed Transfer Pump	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
77	Second Floor Hallway	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
78	Spent Fuel Cooler Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
79	Component Cooler Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
80	Waste Gas Compressor and Tank Area	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
81	Units 1 & 2 Second Floor Hallway	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
82	Units 1 & 3 Spent Fuel Coolant Pump and Cooler	0	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

Attachment E, Radioactive Release Tables

Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
83	Units 1 &2 Component Cooling Pump	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
84	Units 1 & 2 Waste Gas Compressor and Tank	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
85	Unit 1 Second Floor Hallway	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
86	3rd Floor Hallway, Change Rooms, Hatch and Lab Areas	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
87	Cask Decon Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
88	Spent Fuel Receiving Bay	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
89	Equipment Room	2	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
90	Unit 2 Hallway, Change Room, Laundry RM, and RP Lab	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
91	Unit 2 Cask Decon Room	1	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
92	Unit 2 Equipment Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
93	Units 1 & 2 Fuel Loading Area and Spent Fuel	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
94	Unit 1 Hatch, Change Room and Tool Storage	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
95	Unit 1 Equipment Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
96	Hot Machine Shop	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
97	Unit 1 Cask Decon Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
98	West Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
99	East Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
100	Control Battery Room	0	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
101	Cable Room and Elevator lobby	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
102	Unit 2 West Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
103	Unit 2 East Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
104	Unit 2 Control Battery Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
105	Unit 2 Cable Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
106	Unit 1 Cable Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
107	Unit 1 West Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
108	Unit 1 East Penetration Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
109	Unit 1 Control Battery Room	0	Yes	Floor Drains routed to Radwaste Processing	Pre-fire Plans specify ventilation paths for smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				system for monitoring and processing prior to release	control, RIAs monitor contamination levels of smoke	meet the NFPA 805 Radioactive Release Performance Criteria.
109A	Unit 1 AHU, Storage, and Control Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
110	Units 1 & 2 Control Room	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable
111	Unit 2 AHU Room, Storage, and Control Room Lobby	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
112	Unit 3 Control Room	0	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable
113	AHU, Control Room Entrance Lobby	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
114	Purge Inlet Room	0	Yes	Floor Drains routed to Radwaste	Pre-fire Plans specify ventilation paths	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				Processing system for monitoring and processing prior to release	for smoke control, RIAs monitor contamination levels of smoke	that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
115	Purge Exhaust Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
116	AHU and SFP Change Rooms	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
117	Unit 2 Purge Inlet Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
118	Unit 2 Purge Exhaust Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				prior to release	levels of smoke	
119	Units 1 & 2 Air Handling Room	1	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
120	Unit 1 Purge Inlet Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
121	Unit 1 Purge Inlet Room	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
Reactor Buildings						
RB1 122	Unit 1 Reactor Bldg	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
RB2 123	Unit 2 Reactor Building	0	Yes	Floor Drains	Pre-fire Plans	Based on the availability of engineered controls

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
				routed to Radwaste Processing system for monitoring and processing prior to release	specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
RB3 124	Unit 3 Reactor Building	0	Yes	Floor Drains routed to Radwaste Processing system for monitoring and processing prior to release	Pre-fire Plans specify ventilation paths for smoke control, RIAs monitor contamination levels of smoke	Based on the availability of engineered controls for both fire suppression agent run-off and smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
ESV	Essential Vacuum Siphon (ESV) Building	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable
KEO	Keowee Hydro Station	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable
SSF	Standby Shutdown Facility	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
SYD	230 KV Switchyard and Relay House	1	No	N/A	N/A	The NRC staff finds the licensee's statement that the area has no radiological hazards acceptable.
Building 8027	Independent Spent Fuel Storage Installation (ISFSI) Facility	1	Yes	N/A	N/A	Area is regulated under a different NRC license and is therefore not subject to the requirements of 10 CFR 50.48(c).
Building 8055	Warehouse 5Z – Old Warehouse 7	1	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination .	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
Building 8087	RCP Motor Refurbishment	1	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination.	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
Building 8089	Radwaste Facility	1	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination.	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
Building 8091	Scaffold Storage	1	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination.	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.
Building 8093	Warehouse 3 Zone 2-Old Warehouse	2	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination.	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

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Table 3.6-1, Compliance with NFPA 805 Radioactive Release Performance Criteria

Fire Area Fire Zone	Pre-Fire Plan	Rev	RCA or RCZ	Effluent Engineered Controls		NRC Staff Evaluation
				Suppression Water	Combustion Smoke	
Building 8096	Warehouse 3C – Old Warehouse 6	1	Yes	Drainage paths go to Chemical Treatment Pond #3 for monitoring and release	Radiation Protection personnel responding with Fire Brigade Monitor smoke for contamination.	Based on the availability of engineered controls for fire suppression agent run-off and fire brigade monitoring of smoke as described in the LAR, the NRC staff concludes that, the licensee's approach is acceptable to meet the NFPA 805 Radioactive Release Performance Criteria.

N/A – Not Applicable

Principal Contributors: Paul Lain, NRR/DRA
Stephen Dinsmore, NRR/DRA

Date: December 29, 2010

Pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR), by letter dated December 6, 2010, the NRC sent the licensee the draft Safety Evaluation approving the proposed amendments for an opportunity for the licensee to comment on any proprietary or security-related aspects of the draft Safety Evaluation. By letter dated December 22, 2010, the licensee provided comments. The NRC reviewed and accepted all comments made by the licensee. Pursuant to 10 CFR 2.390 the NRC has redacted information as identified by blank space enclosed within double brackets as shown here [[ ]].

In addition, the December 6, 2010, letter also requested the licensee to provide comments on factual errors or clarity concerns contained in the draft Safety Evaluation. By letter dated December 22, 2010, the licensee provided comments. The NRC has considered each comment and changed the Safety Evaluation as appropriate.

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.

If you have any questions, please call me at 301-415-1345.

Sincerely,

/RA/

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

Docket Nos. 50-269, 50-270, and 50-287

Enclosures:
1. Amendment No. 371 to DPR-38
2. Amendment No. 373 to DPR-47
3. Amendment No. 372 to DPR-55
4. Safety Evaluation ~~contains official use only security-related information~~

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~~Enclosure 4 transmitted herewith contains security-related information. When separated from Enclosure 4, this document is decontrolled.~~