T. PRESTON GILLESPIE, Jr.
Vice President
Oconee Nuclear Station

Duke Energy
ON01VP / 7800 Rochester Hwy.
Seneca, SC 29672

864-873-4478
864-873-4208 fax
T.Gillespie@duke-energy.com

November 19, 2010

U. S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

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) 2008-01

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke Energy) proposes to amend Renewed Facility Operating Licenses (FOLs) Nos. DPR-38, DPR-47, and DPR-55. This License Amendment Request (LAR) requests Nuclear Regulatory Commission (NRC) review and approval for adoption of a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a), 10 CFR 50.48(c), and the guidance in Regulatory Guide (RG) 1.205. The LAR was submitted to the NRC on April 14, 2010 and supplemented on September 13, 2010, September 27, 2010, and October 14, 2010.

Enclosure 1 addresses RAIs issued on October 14, 2010 and RAI 2-25 which was received electronically on November 10, 2010.

There are no new commitments being made as a result of this supplement. Other activities requiring further actions will be tracked in the corrective action program and are not considered commitments.

In a conference call on November 10, 2010, the NRC and Duke Energy decided that a common understanding has been reached for all RAIs except RAI 5-75. Initially, the NRC requested that all RAI responses be held until a common understanding can be achieved for RAI 5-75. However, after a conference call on November 18, 2010, Duke Energy decided to submit the RAI responses since the requested information provided in RAI 5-75 is correct. The NRC stated this is an acceptable course of action. Discussions with the NRC will continue until agreement is reached for RAI 5-75. If required, the LAR will be supplemented at that time. Additionally, the revised version of RAI 5-75, if required, will be posted to the NFPA 805 sharepoint when completed.

A006
NRR

If there are any questions regarding this submittal, please contact Reene' Gambrell at (864) 873-3364 or David J. Goforth at (704) 382-2659.

I declare under penalty of perjury that the foregoing is true and correct. Executed on November 19, 2010.

Sincerely,

T Gillespie

T. Preston Gillespie, Jr.
Vice President
Oconee Nuclear Station

Enclosure:

1. 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)

cc: w/o enclosures

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Marquis One Tower
245 Peachtree Center Ave., NE, Suite 1200
Atlanta, Georgia 30303-1257

Mr. John Stang, Project Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-8 G9A
Washington, D. C. 20555

Mr. Paul Lain, NFPA-805 Program Manager
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Mail Stop O-10 C15
Washington, D. C. 20555

Mr. Andy Sabisch
Senior Resident Inspector
Oconee Nuclear Station

Susan E. Jenkins, Manager
Radioactive & Infectious Waste Management
SC Dept. of Health and Env. Control
2600 Bull St.
Columbia, SC 29201

ENCLOSURE 1

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)

ENCLOSURE 1

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)

REQUEST FOR ADDITIONAL INFORMATION (RAI) 2-5c:

The Nuclear Regulatory Commission (NRC) staff requires follow-up information on the response to RAI 2-5. During the review of the B-1 Table in the Oconee Nuclear Station (ONS) LAR, the NRC staff identified the following issues that are linked to specific B-I Table elements. The licensee should review the license amendment request (LAR) submittal and ensure that these and any similar conditions are resolved appropriately.

B-1 Table: NFPA 805, Chapter 3 3.2.2 – Management Policy Direction and Responsibility

NFPA 805, Section 3.2.2 states: "A policy document shall be prepared that defines management authority and responsibilities and establishes the general policy for the site fire protection program." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3 – Prevention

NFPA 805, Section 3.3 states, in part: A fire prevention program with the goal of preventing a fire from starting shall be established, documented, and implemented as part of the fire protection program. The two basic components of the fire prevention program shall consist of both of the following:

(1) Prevention of fires and fire spread by controls on operational activities

(2) Design controls that restrict the use of combustible materials

Please provide a compliance statement that addresses how these two requirements are met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1 – Fire Prevention for Operational Activities

NFPA 805, Section 3.3.1 states: "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." Please provide a compliance

statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1.1 - General Fire Prevention Activities

NFPA 805, Section 3.3.1.1, states: "The fire prevention activities shall include but not be limited to the following program elements..." Please provide a compliance statement that addresses how the "but not be limited to" aspect of the NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.1.2 - Control of Combustible Materials

NFPA 805, Section 3.3.1.2, states: "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..." Please provide a compliance statement that addresses how both the requirements to develop and implement procedures for the control of general housekeeping and transient combustibles and the "but not be limited to" aspect of this NFPA 805 requirement are met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.7 – Bulk Flammable Gas Storage

NFPA 805, Section 3.3.7, states: "Bulk compressed or cryogenic flammable gas storage shall not be permitted inside structures housing systems, equipment, or components important to nuclear safety." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.3.12 – Reactor Coolant Pumps

The NRC staff has determined that this parent element contains requirements related to seismic and other accident/off-normal conditions that are not addressed in the detailed sub-parts to this element. Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the elements required for a B-1 Table entry (for example, the reference that documents the compliance).

B-1 Table: NFPA 805, Chapter 3, Element 3.4.2 – Pre-Fire Plans

NFPA 805, Element 3.4.2, states: "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." Please provide a compliance statement that addresses how this NFPA 805 requirement is met at ONS. Include in the response all of the

elements required for a B-1 Table entry (for example, the reference that documents the compliance).

RAI 2-5c RESPONSE:

Compliance statements and supporting documentation for support of the various B-1 table elements discussed above are provided below:

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 fire protection program.

The Design Basis Specification for Fire Protection is the primary policy document that defines management responsibilities and establishes the general policy for the site fire protection program. Section 3.2.1 of the Design Basis Specification describes the Fire Protection Program Organization and Responsibilities.

Reference(s): Design Basis Specification for Fire Protection, Section 3.2.1

Compliance Statement: Comply

Compliance Basis: The Design Basis Specification for Fire Protection is the primary policy document that defines management responsibilities and establishes the general policy for the site fire protection program.

Section 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 fire protection program. The two basic components of the fire prevention program shall consist of both of the following:

(1) Prevention of fires and fire spread by controls on operational activities
(2) Design controls that restrict the use of combustible materials

The design control requirements listed in the remainder of this section shall be provided as described.

The ONS fire prevention program objective is documented in the Design Basis Specification for Fire Protection, which includes the policy for control of combustible materials and ignition controls. These objectives are implemented by Nuclear System Directives (NSD) and other documents as described in the subsequent sections. Section 3.3.1 below describes the implementation of requirements for fire prevention for

operational activities and design controls for the use of combustible materials (objectives (1) and (2)).

Reference(s): Design Basis Specification for Fire Protection, Section 3.4

Compliance Statement: Comply

Compliance Basis: The ONS fire prevention program objective is documented in the Design Basis Specification for Fire Protection, which includes the policy for control of combustible materials and ignition controls. These objectives are implemented by Nuclear System Directives (NSD) and other documents as described in the subsequent NFPA 805 sections (3.3.1).

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.

The ONS fire prevention program objectives are implemented by NSD and other documents as described in the subsequent sections.

Reference(s): EM 4.6, "Oconee Engineering Support Program Walkdowns", Attachment 3
EDM-601, "Engineering Change Manual", Appendix K and N
General Employee Training – Plant Access Training, Page 14
NSD-104, "Materiel Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns", All
NSD-228, "Applicability Determination", Appendix E
NSD-301, "Engineering Change Program", Appendix A
NSD-313, "Control of Flammable and Combustible Materials", All
NSD-314, "Hot Work Authorization", All
NSD-316, "Fire Protection Impairment and Surveillance", All
NSD-320, "Guidance for Performing Licensing Review of Proposed Changes to the Fire Protection Program", All
NSWP 4.2, "Fire Prevention", Enclosure 3.4

Compliance Statement: Comply

Compliance Basis: The ONS fire prevention program objectives are implemented by NSD and other documents as described in the subsequent sections.

3.3.1.1 General Fire Prevention Activities

The fire prevention activities shall include but not be limited to the following program elements:

Duke Energy has developed multiple directives to address fire prevention. These directives address, at a minimum, the fire protection program elements identified in this section. Upon review of the elements listed below, the NFPA 805 code requirements are satisfied and no other additional elements were evaluated.

Reference(s): NSD-104 "Materiel Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns", All
NSD-313 "Control of Flammable and Combustible Materials", All
NSD-314, "Hot Work Authorization", All
NSWP 4.2 "Fire Prevention", All

Compliance Statement: Comply

Compliance Basis: Multiple directives and work practices have been developed to address fire prevention. These directives include but are not limited to the programmatic elements provided in NFPA 805 Section 3.3.1.1."

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:

Compliance with procedures for control of general housekeeping practices and the control of transient combustibles include but are not limited to the following elements in the subsections of Section 3.3.1.2. Upon review of the elements listed in the subsections of 3.3.1.2, the NFPA 805 code requirements are satisfied and no other additional elements were evaluated.

NSD-104, "Material Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns," provides direction on maintaining the cleanliness of the plant. NSD-313, "Control of Flammable and Combustible Materials", establishes the minimum requirements for the control of transient combustible and flammable materials.

Reference(s): NSD-104 "Materiel Condition/Housekeeping, Foreign Material Exclusion and Seismic Concerns", All
NSD-313 "Control of Flammable and Combustible Materials", All

Compliance Statement: Comply

Compliance Basis: Directives have been developed and implemented for general housekeeping practices and control of combustible materials. These directives include but are not limited to the programmatic elements provided in NFPA 805 Section 3.3.1.2 (1) through (6).

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.

ONS compressed gas storage is generally located outdoors in the East Yard. Flammable gas cylinders are located in various areas of the plant, by design, for chemistry labs and for Post Accident Monitoring instrumentation.

Bulk storage of Hydrogen Gas is stored in the northeast yard area. There is also a propane storage tank adjacent to the Unit 1 Transformer area. These storage tanks are located outdoors and away from the SSCs important to safety.

Previously ONS stated that there is no bulk gas storage in areas affecting safe shutdown equipment (The Design Basis Specification for Fire Protection, Duke Energy Comparison to BTP 9.5-1 letter and UFSAR Section 9.5.1.4.2 state "There is no bulk gas storage in areas affecting safe shutdown equipment."). Under NFPA 805, the requirement is no bulk storage inside structures housing systems, equipment or components important to nuclear safety. NFPA 55 (2005 edition) defines bulk hydrogen compressed gas system as an assembly, including storage containers, with a capacity greater than 400 ft^3. There are hydrogen storage cylinders located in the Auxiliary Building by design. The Auxiliary Building contains systems, equipment or components important to nuclear safety; however, the immediate areas in which the cylinder storage exists do not contain systems, equipment or components important to nuclear safety.

NSD-313, "Control of Combustible and Flammable Material," provided restrictions on the storage location of flammable materials with specific restrictions for ONS. Any changes, to the amount of materials or the storage location of materials, require review and approval from the Fire Protection Engineer. EDM-601, "Engineering Change Manual," provides controls on the introduction of combustible/flammable materials which may increase the expected magnitude of a fire. The introduction of flammable gases requires the approval of the Site Fire Protection Engineer.

The chemical control program, which is administered in accordance with NSD-116, ensures compliance with applicable regulations and requirements including NFPA Codes. Any changes to the bulk storage of combustible/flammable gases would require authorization in accordance with both NSD-116 and NSD-313.

Reference(s): Response to Appendix A to Branch Technical Position APCSB 9.5-1, February 1982, Section 4.B(2), Page 26
UFSAR Section 9.5.1.4.2, Page 9.5-4
NSD-116 "Nuclear Chemical Control Program", Section 116.6.7
Design Basis Specification for Fire Protection, Section 3.4.2
EWP 7.2 "Storing Chemicals", Section 3
EDM-601, "Engineering Change Manual", Attachment K
NSD-313, "Control of Combustible and Flammable Material", Section 313.6.1
Drawing O-3-10

Compliance Statement: Comply

Compliance Basis: Bulk flammable gas is not stored in areas important to nuclear safety unless used in plant operations or systems by design.

Reference(s): Design Basis Specification for Fire Protection, Section 3.4.2

Compliance Statement: Submit for NRC Approval

Compliance Basis: Bulk flammable gas is not stored in areas important to nuclear safety unless used in plant operations or systems by design. See Attachment L of the Transition Report for further details on the request for NRC approval for the use of flammable gas cylinders in certain areas of the plant.

3.3.12 Reactor Coolant Pumps

For facilities with non-inerted containments, reactor coolant pumps 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.

The reactor coolant system components are designed to maintain their functional integrity during a design basis earthquake. Each reactor coolant pump oil collection system and sub-components are designed to meet QA-3 seismic criteria.

Reference(s): Design Basis Specification for Reactor Coolant System, Section 2.3.4
Engineering Support Document for Reactor Coolant Pumps, Section 1.1.4

Compliance Statement: Comply

Compliance Basis: Each reactor coolant pump oil collection system and sub-components are designed to maintain their functional integrity during a design basis earthquake.

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.

Pre-fire plans are provided for all areas of ONS, both those within the power block (all areas within the protected area) and select important structures within the owners controlled area. The pre-fire plans are detailed as outlined in the response to Section 3.4.2.1. The pre-fire plans are maintained current as outlined in the response to Section 3.4.2.2. The pre-fire plans are available to the fire brigade as outlined in the response to Section 3.4.2.3.

Reference(s): ONS Fire Plan, All

Compliance Statement: Comply

Compliance Basis: Pre-fire plans are provided for all areas of ONS, both those within the power block (all areas within the protected area) and select important structures within the owners controlled area. The pre-fire plans are detailed as outlined in the response to Section 3.4.2.1. The pre-fire plans are maintained current as outlined in the response to Section 3.4.2.2. The pre-fire plans are available to the fire brigade as outlined in the response to Section 3.4.2.3.

RAI 2-11c:

The NRC staff requires follow-up information on the response to RAI 2-11. The following documents are used in Table B-1 of the LAR to document previous NRC approval of alternatives to certain NFPA 805, Chapter 3 requirements. [Bracketed statement indicates the Chapter 3 element where the document is used].

- NRC Safety Evaluation (SE) dated August 11, 1978 (ADAMS Accession No. ML7911280619) [3.5.3; 3.5.4; 3.5.15; 3.5.16; 3.6.1]

- NRC SE dated June 7, 1988 (ADAMS Accession No. ML8806170310) [3.6.1]

- NRC Letter dated April 14, 1981 (ADAMS Accession No. ML810422024) [3.11.2]
- NRC Exemption dated August 21, 1989 (ADAMS Accession No. ML8908290074) [3.11.2; 3.11.4]

Please justify the continued validity of each of these approvals, with regard to their use in the B-1 Table.

Used in this context, the term "valid" means that the technical basis for approval still applies. For example, plant modifications or other changes have not invalidated the assumptions or analysis that formed the basis for the approval, or new information has not surfaced that would invalidate the original finding.

RAI 2-11c RESPONSE:

Justification for the validity of the approvals discussed above were enhanced as outlined below:

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.

Reference(s): Design Basis Specification for Fire Protection, Section 3.3.1.1.2
NRC SE Report dated August 11, 1978, Section 4.3.1.2, Page 17

Compliance Statement: Complies via Previous Approval

Compliance Basis: The use of the High Pressure Service Water (HPSW) pumps, although not listed fire pumps to supply fire water was found to be acceptable by the NRC in the August 11, 1978 SE Report. Section 4.3.1.2 states:

> *"There are two 6,000 gpm HPSW pumps and one 500 gpm jockey pump, all rated at 117 psi net pressure. The two large pumps are considered redundant, each capable of supplying the largest design fire flows plus other simultaneous demands on the HPSW system. The pumps are electric motor driven receiving power from separate Unit No. 1 buses."*
>
> *"We find that the fire pumps meet the objectives outlined in Section 2.2 of this report and are, therefore, acceptable."*

The HPSW pumps, as approved by the SE Report, are still used as the source of fire water at the Oconee plant. There have been no plant modifications or other changes that would invalidate the basis for approval. The HPSW pumps have not been changed.

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.

Reference(s): Design Basis Specification for Fire Protection, Section 3.3.1.1.2
NRC SE Report dated August 11, 1978, Section 4.3.1.2, Page 17

Compliance Statement: Complies via Previous Approval

Compliance Basis: Diesel-engine driven fire water pumps are not utilized. The NRC has accepted the use of the HPSW pumps as Oconee fire pumps in the August 11, 1978 SE Report. Section 4.3.1.2 states:

> *"There are two 6,000 gpm HPSW pumps and one 500 gpm jockey pump, all rated at 117 psi net pressure. The two large pumps are considered redundant, each capable of supplying the largest design fire flows plus other simultaneous demands on the HPSW system. The pumps are electric motor driven receiving power from separate Unit No. 1 buses."*
>
> *"We find that the fire pumps meet the objectives outlined in Section 2.2 of this report and are, therefore, acceptable."*

The HPSW pumps, as approved by the SE Report, are still used as the source of fire water at the Oconee plant. There have been no plant modifications or other changes that would invalidate the basis for approval. The HPSW pumps have not been changed.

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.

Reference(s): Design Basis Specification for Fire Protection, Section 3.3.1.1.1
NRC SE Report dated August 11, 1978, Section 4.3.1.3, Page 18

Compliance Statement: Complies via Previous Approval

Compliance Basis: The NRC determined that the hydrants installed at approximately 300 feet intervals are acceptable in the August 11, 1978 SE Report which Section 4.3.1.3 states:

> *"Yard fire hydrants have been provided at approximately 300 ft. intervals around the exterior of the plant. Auxiliary gate valves are not provided on the hydrant laterals, with the result that a portion of the fire water heads would have to be removed from service if a fire hydrant has to be isolated for maintenance."*
>
> *"Hose houses have been provided at seven of the yard hydrants, each equipped with at least 200 ft. of 2-1/2 inch hose, 200 ft. of 1-1/2 inch hose and other manual fire fighting tools."*
>
> *"We conclude that, upon implementation of the modifications described above, the fire water piping system meets the objectives outlined in Section 2.2 of this report and are, therefore, acceptable."*

Note that modifications referred to in the conclusion pertain to cross-connections of the auxiliary building headers to provide dual feed for hose stations in the area and are not applicable to this NFPA 805 section. The fire hydrants at the Oconee plant are still situated consistent with the SE Report. There have been no plant modifications or other changes that would invalidate the basis for approval. The hydrants are placed at intervals no greater than 300 feet. Hose houses are evaluated as part of the NFPA 24 code compliance evaluation and are not part of the prior approval compliance statement.

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.

Reference(s): Design Basis Specification for Fire Protection, Section 3.3.1.1.2
NRC SE Report dated August 11, 1978

Compliance Statement: Complies via Previous Approval

Compliance Basis: The use of the HPSW system and LPSW system for fire protection was found acceptable by the NRC. The August 11, 1978 SE Report Section 4.3.1.2 states:

"There are two 6,000 gpm HPSW pumps and one 500 gpm jockey pump, all rated at 117 psi net pressure. The two large pumps are considered redundant, each capable of supplying the largest design fire flows plus other simultaneous demands on the HPSW system."

"We find that the fire pumps meet the objectives outlined in Section 2.2 of this report and are, therefore, acceptable."

The HPSW pumps, as approved by the SE Report, are still used as the source of fire water at the Oconee plant. There have been no plant modifications or other changes that would invalidate the basis for approval. The HPSW system has not been changed which would affect the capacity to provide the required fire flows.

The NRC August 11, 1978 SE Report Section 4.3.1.4 states:

"Interior hose stations equipped with 1-1/2 inch fire hose have been provided through the plant except in containment. Some areas are too far away from a hose station for effective fire fighting. The licensee has proposed to provide additional hose stations so that all areas containing or exposing safety-related equipment will be within effective fire fighting range of at least one hose

station using not more than 100 ft. of 1-1/2 inch fire hose, and to provide hose stations inside containment supplied by the LPSW system."

"Upon completion of the proposed modification, we find that the interior hose stations meet the requirements of Appendix A to BTP 9.5-1 and are, therefore, acceptable."

The modification referenced above to provide hose stations inside containments supplied by the LPSW system has been implemented. The LPSW system, as referenced by the SE Report, is still used as the source of fire hose stations inside containment There have been no plant modifications or other changes that would invalidate the basis for approval. The LPSW system has not been changed with respect to the supply of the reactor building hose stations.

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.

Reference(s): NRC SE Report dated August 11, 1978, Section 4.3.1.4, Page 18
NRC SE Report dated June 7, 1988
Duke Letter to the NRC dated February 17, 1978
Design Basis Specification for Fire Protection, Section 3.3.1.1.7

Compliance Statement: Complies via Previous Approval

Compliance Basis: The standpipe and fire hose systems were found acceptable by the NRC. The NRC August 11, 1978 SE Report Section 4.3.1.4 states:

"Interior hose stations equipped with 1-1/2 inch fire hose have been provided through the plant except in containment. Some areas are too far away from a hose station for effective fire fighting. The licensee has proposed to provide additional hose stations so that all areas containing or exposing safety-related equipment will be within effective fire fighting range of at least one hose station using not more than 100 ft. of 1-1/2 inch fire hose, and to provide hose stations inside containment supplied by the LPSW system."

> *"Upon completion of the proposed modification, we find that the interior hose stations meet the requirements of Appendix A to BTP 9.5-1 and are, therefore, acceptable."*

The modification referenced above to provide additional hose stations has been implemented. The fire hose stations as approved by the NRC and installed by Oconee are still situated in accordance with the SE Report. There have been no plant modifications or other changes that would invalidate the basis for approval.

The NRC June 7, 1988 SE Report states:

> *"The licensee has proposed a modification to move five fire hose station presently located within the Cable Spreading rooms to stairwell locations outside the rooms. The purpose of this modification is to make the fire hose station accessible for fire fighting activities in these rooms."*
>
> *"The proposed modification of fire hose stations will enhance the fire fighting capabilities in the Cable Spreading Rooms and meets the intent of the guidance provided in National Fire Protection Association (NFPA) Std. 14, Standpipe and Hose Systems, for the location of fire hose connections. After reviewing these proposed TS changes, the staff has determined they are acceptable."*

The modification referenced above to relocate fire hose stations from the Cable Spreading rooms has been implemented.

Fire hose stations are not installed in the Cable Spreading Rooms as specified by the SE Report. There have been no plant modifications or other changes that would invalidate the basis for approval.

Note for NFPA 805 Section 3.11.2 Fire Barriers

Upon further review of the Table B-1 compliance basis for Section 3.11.2, the Complies via Previous Approval alternative is not required.

NRC Exemption to various Appendix R Requirements dated August 21, 1989, Exemption 1 was included for the West Penetration Fire Area ceiling fire barrier. During the transition process, the West Penetration Fire Areas were revised to include the Purge Inlet Rooms over the West Penetration Rooms, thereby invalidating the need for

this exemption. This is documented in the LAR, Attachment K Existing Licensing Action Transition, page 5 of 115 that this exemption is not required for transition.

NRC Letter to Duke Energy dated April 14, 1981 was included as it documented that an exemption was not required for the east / west penetration room fire barrier separation. Since there was no NRC approval required, this compliance statement will be removed. Compliance is documented via engineering evaluation in this case.

The revised Section 3.11.2 compliance basis is provided below:

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.

Fire Barriers required by Chapter 4 include the following:

- Auxiliary Building to Turbine Building
- Auxiliary Building to Reactor Building
- Auxiliary Building to West Penetration Room
- West Penetration Room to Reactor Building
- Turbine Building to Blockhouse 1/2
- Turbine Building to Blockhouse 3
- Blockhouse 1/2 to CT-4

OSC-7185 and OSC-9302 provide documentation regarding the fire barriers separating the aforementioned fire areas.

Compliance Statement: Complies with Use of EEEE

Compliance Basis: Appropriately rated fire barriers have been found acceptable by licensee evaluation.

Reference(s): OSC-7185, All
OSC-9302, All

3.11.4 Through Penetration Fire Stops
Through penetration fire stops for penetrations such as pipes, conduits, bus ducts, cables, wires, pneumatic tubes and ducts, and similar building service equipment that pass through fire barriers 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.

(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.

Reference(s): NRC Exemption to various Appendix R Requirements dated August 21, 1989, Exemption 5

Compliance Statement: Complies via Previous Approval

Compliance Basis: The NRC approved the mechanical pipe penetrations in the reactor building walls.

"*The mechanical pipe penetration design was observed during the plant Appendix R inspection during the week of January 26-30, 1987, to be similar to penetration designs used at other facilities. The penetrations have been designed to meet multiple containment integrity criteria.*

The combustible loadings near the penetration are low; therefore, a fire of significant magnitude or duration should not occur near the penetrations. If a fire does occur, it is probable that the substantial construction of the piping penetrations, combined with the large room volumes on either side of the penetrations, will prevent fire propagation through the containment boundary. It is, therefore, concluded that the existing unrated containment mechanical pipe penetrations provide reasonable assurance that a fire will not propagate through the barrier and are, therefore, an acceptable deviation from the technical requirements of Section III.G.2.a of Appendix R.

Based on the above evaluation, the existing mechanical pipe penetrations in the reactor building walls provide a level of fire protection equivalent to the technical requirements of Section III.G.2.a of Appendix R and provide reasonable assurance that the fire will not propagate through the containment boundaries."

The mechanical pipe penetrations are consistent with the SE Report. There have been no plant modifications or other changes that would invalidate the basis for approval. This item is addressed in the LAR dated April 14, 2010, Attachment K, Existing Licensing Action Transition, page 15 of 115. The Existing Licensing Action Transition review concluded the bases for previous acceptance are still valid.

RAI 2-19c:

The NRC staff requires follow-up information on the response to RAI 2-19, Approval Request # 2, Fuel-Fired Heaters.

NFPA 805 is a consensus code developed by a diverse group of individuals that represented subject matter experts in the field of fire protection, nuclear insurers, manufacturers, enforcement, and lastly experts from the utilities. Their collective expertise concluded, in Section 3.3.1.3.4 of NFPA 805 that 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 fire. The NRC staff finds that the licensee's response to RAI 2-19 does not justify the use of fuel-fired heaters in areas containing equipment important to nuclear safety or where there is a potential for radiological releases resulting from fire. Therefore, the NRC staff is not prepared to approve this request.

Should the licensee choose to pursue this request, the NRC expects to receive all of the following information, for each area these heaters may be used:

1. The increase in core damage frequency due to the use of these heaters;
2. The increase in large early release frequency due to the use of these heaters;
3. A demonstration, in detail, that defense-in-depth is maintained despite the addition of combustible liquids and ignition sources due to these heaters;
4. A demonstration, in detail, that safety margins are maintained despite the use of these heaters;
5. A demonstration, in detail, that the radioactive release goal, objectives, and performance criteria are met, despite the use of these heaters
6. A demonstration, in detail, that the nuclear safety goal, objectives, and performance criteria are met, despite the use of these heaters

RAI 2-19c RESPONSE:

ONS is withdrawing Approval Request #2 from Attachment L of the LAR dated April 14, 2010 from NRC review and approval.

Table B-1 Element 3.3.1.3.4 is subsequently revised as follows:

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.

ONS will prohibit the use of portable fuel-fired heaters within plant areas containing equipment important to nuclear safety or where there is the potential for radiological release due to fire.

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. This item is being tracked in the Corrective Action Program.

Reference(s): NSD-316 "Fire Protection and Impairment Surveillance", Section 316.6
S.D. 3.2.14 "Fire Protection Program Compensatory Measures Process", Chart 6

Compliance Statement: Comply

Compliance Basis: Specific administrative directives have been developed for control of temporary heating devices. 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. The use of portable fuel-fired heaters is prohibited within plant areas containing equipment important to nuclear safety or where there is the potential for radiological release due to fire.

Confirmatory Item: 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. Reference Attachment S in the LAR dated April 14, 2010.

RAI 2-22c:

The NRC staff requires follow-up information on the response to RAI 2-22, Approval Request #8. Demonstrate the capability to extinguish expected fires inside containment during all operational modes. Specifically, in light of the limited ability of the current low-pressure service water (LPSW) system that supplies the existing standpipe system, explain how additional equipment and capabilities, are sufficient to handle the expected hazards in this area during all modes of operation.

Additionally, explain in summary detail, how the above capabilities are integrated into the Fire Protection Program (for example in procedures, fire pre-plans, fire brigade training, etc.).

RAI 2-22c RESPONSE:

Fire hazards in the reactor buildings include electrical panels (lighting panels, power panels), small transformers, motors (sump pump, fan), reactor coolant pumps, cable insulation, and lubricating oil for the reactor coolant pumps. During outages, additional fire hazards are introduced (transient combustibles) as part of the maintenance work that occurs inside containment. The majority of these hazards, were a fire to occur and the reactor building be accessible, could be controlled with the use of portable fire extinguisher(s) or the hose stations present in containment. If the unit is at power and a fire occurs, the fire would likely burn out before the reactor building could be accessed. The most significant fire hazard is the reactor coolant pump lubricating oil. The lubricating oil is contained in a seismically designed system for collection and containment. If a fire were to occur involving the lubricating oil, it would likely be of the magnitude that the fire brigade would stage outside of the reactor building and use the auxiliary building or yard water supply to attack the fire.

There are six fire extinguishers permanently staged adjacent to the reactor building personnel hatch in the Auxiliary Building for each unit. In addition to these six extinguishers, portable fire extinguishers are placed in each reactor building during refueling outages in accordance with the Oconee Fire Protection Equipment Inspection maintenance procedure. There are 19 extinguishers placed in various locations of the Unit 1 Reactor Building and 16 extinguishers each placed in various locations of the Unit 2 and Unit 3 Reactor Buildings. If entry into the reactor building is required for non-refueling outage maintenance activities, portable fire extinguishers will be provided commensurate with the maintenance activities. This placement includes addressing the hazards and locations for the particular maintenance activity. Fire extinguishers are the primary tool in fire fighting within containment for situations within their capabilities.

During power operations, the standard response to a fire in containment is not to enter the reactor building and let the fire burn out either via fuel consumption or lack of oxygen. As part of the fire brigade training and standard operating procedures, the fire brigade will not enter an area to fight a fire without a charged hose line. If access to containment in order to fight a fire that occurs during power operations is deemed prudent, it would take time to open containment, assess fire conditions, assess radiological conditions, then if conditions are permissible defeat

the personnel hatch interlocks to introduce a hose line and proceed with firefighting operations using a charged water supply from outside of containment. Defeating the personnel hatch interlocks which permit both hatch doors open simultaneously, while capable, is not regularly performed. The charged hose line would be connected to the HPSW system via hose stations in the Auxiliary Building. The HPSW system is the normal plant fire protection water system.

There are hose stations located in the immediate area adjacent to each personnel hatch in the Auxiliary Building.

During non-power operation modes a fire could be attacked using the existing reactor building hose stations (if at the fire brigades discretion the fire is within the capabilities of the hose stations), with a hose line connected to the Auxiliary Building standpipe system as described above, or with a hose line connected to a yard hydrant if the exterior equipment hatch is open. There are fire hydrants located on the yard HPSW loop near each reactor building. Hose lines from the Auxiliary Building hose station or the yard hydrant locations would utilize a 2-1/2" hose line for entry. It is standard operating procedure that when greater than 100 feet of hose is necessary that larger diameter hose will be used to overcome friction losses. Hose and nozzles are available for these firefighting applications.

The fire brigade may utilize the reactor building hose stations for a small fire such as a trash barrel, rags, or other small localized fire within the capabilities of the standpipe system. This type of fire would be controlled with a limited volume of water. The flows and pressures present in the reactor building hose stations are sufficient for this type of operation. Fires of this size may also be controlled with the use of fire extinguishers. The method of extinguishment is at the discretion of the fire brigade.

A review was performed of combustible control and hot work directives to determine if appropriate guidance for controls within the reactor building is provided. The review determined that the existing directives include appropriate controls.

The fire brigade will develop a Standard Operating Guide (SOG) for fighting a fire in the reactor building. Training is already performed on tactics for fighting fires of this nature. This training will be reinforced with a new SOG. The Fire Brigade Coordinator will review the Fire Plans to determine if enhancement is necessary. This item is being tracked by the Corrective Action Program.

RAI 2-25:

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

NFPA 805 Section 3.5.4 states:
"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."

Demonstrate how the pump arrangement supplying the fire protection water supply for the Keowee Hydro Station, an ONS designated power block plant structure, complies with Sections 3.5.3 and 3.5.4 of NFPA 805. Alternatively, and in accordance with 10 CFR 50.48(c)(2(vii), the licensee may request NRC approval for use of a performance-based method as a means of demonstrating compliance with NFPA 805 Chapter 3.

RAI 2-25 RESPONSE:

The fire pump for the Keowee Hydro Station (Keowee) complies with NFPA 805 Section 3.5.3 via engineering evaluation. The fire pump is installed in accordance with NFPA 20 as evaluated in calculation OSC-9678, "NFPA 20 Code Conformance Review," and is capable of providing the required hose station flow and pressure in accordance with NFPA 13-E, 1966 "Standpipe Systems". NFPA 13-E, 1966 is the code associated with Keowee construction (which reiterates the requirements in NFPA 14, 1963 edition for Standpipe and Hose Systems) based on a review of available internal documentation.

ONS is requesting NRC approval for use of a performance-based method as means of demonstrating compliance of the Keowee fire pump arrangement with regards to NFPA 805 Section 3.5.4 in accordance with 10 CFR 50.48(c)(2)(vii). The following is the approval request.

NFPA 805 Section 3.5.4

NFPA 805 Section 3.5.4 states:

> *"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."*

The Keowee Hydro Station (Keowee) is provided with one electric motor-driven fire pump. There is no secondary/back-up fire pump. The existing electric fire pump is installed in accordance with NFPA 20 and is capable of providing the required flow rate and pressure to the Keowee hose stations. NFPA 20 does not contain a requirement to install a second fire pump. The requirement for a secondary or back-up fire pump in NFPA 805 is to ensure the "Power Block" or power production areas are protected. These areas include the turbine/auxiliary/reactor buildings. Keowee is spatially separated from any other ONS power production areas by approximately 3000 feet. Secondary fire pumps are necessary in areas where there are a multitude of adjacent fire areas/zones that could allow fire to spread/migrate/extend between redundant fire area/zones and affect "safe shutdown" capability if the fire were to go unsuppressed due to fire pump operation adversely affected by the initial fire (e.g. the initial fire impacts the primary fire pump power cables).

Keowee is an extension of the ONS "power block" as Keowee is used as emergency power. ONS may rely on Keowee for shutdown power in the event of a fire occurring at the ONS plant.

A fire at Keowee is mitigated by the ONS fire brigade with support from mutual aid fire companies. Keowee is a credited safe shutdown system; however, a fire at Keowee does not impact the safe shutdown capability at ONS using other credited power systems. A fire at ONS is not postulated concurrent with a fire at Keowee, therefore, a fire at ONS does not impact the ability of Keowee to provide emergency power.

The main hazards inside the Keowee structure are the electrical generators which are each protected by an automatic CO_2 extinguishing system. While they are not associated with NFPA 805 credited equipment and therefore not NFPA 805 required fire protection features, other suppression systems at Keowee include the main transformer water spray system and the oil storage room water spray systems.

The basis for the approval request of this deviation is:

- A fire at Keowee does not impact the safe shutdown 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 ONS power production structures.
- The main purpose of the fire pump at Keowee is to supply the fire hose stations with flow and pressure.
- Compensatory measures are provided in the event the Keowee fire pump is out of service.

Nuclear Safety and Radiological Release Performance Criteria:

The single electric motor-driven fire pump at Keowee does not affect nuclear safety since a fire is not simultaneously postulated at ONS and Keowee. Keowee is the emergency power source for ONS. If a fire at Keowee were to render it unavailable, ONS would proceed to shutdown using normal power. Therefore, there is no impact on the nuclear safety performance criteria.

There is no impact on the overall ONS radiological release performance criteria due to Keowee not having redundant fire pumps installed. There are no radiological fire suppression concerns at the Keowee location.

Safety Margin and Defense-in-Depth:

The single electric fire motor-driven pump at Keowee does not impact "safe shutdown" fire protection for the "power block" or power production areas of the turbine/auxiliary/reactor buildings or the SSF. 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. Therefore, the safety margin inherent in the analysis for the fire event has been preserved.

There is no impact to ONS fire protection defense-in-depth due to not having redundant fire pumps at Keowee. Compensatory measures are required to ensure there is a supply of water for fire operations if the Keowee fire pump is unavailable.

Conclusion:

NRC approval is requested for the single electric fire pump at the Keowee Hydro Station.

The engineering analysis performed determined that the performance-based approach utilized to evaluate a variance from the requirements of NFPA 805 Chapter 3:

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

RAI 3-38c:

The NRC staff requires follow-up information on the standby shutdown facility (SSF) design assumption.

DISCUSSION

Response to RAI 3-1 stated the following:

The licensee requested that the NRC formally document as a "prior approval" recognition that within the first 10 minutes following the identification of a confirmed active fire requiring activation of the SSF growth will not reach a point where fire damage will:

- Result in spurious equipment operation
- Result in a loss of offsite power condition
- Preclude operation of plant equipment from the control room

Response to RAI 3-38 stated the following:

Alignment basis will be revised (per corrective action program) to reflect that the 10-minute assumption is only used for SSF risk areas. It is not used for non-SSF risk areas. Other spurious operations beyond this assumption were postulated to occur until mitigating actions are taken. This is consistent with the scope of Prior-Approval Clarification Request 1 of Attachment T: "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."

The licensee's letter dated September 20, 1982: The licensee responsed to an NRC staff RAI regarding spurious actuations of components which may affect the safe shutdown capability, which states:

> The originally accepted design concept for SSF operation was based on a 10 minute capability to transfer control to the SSF. Hot shorts or spurious actuation due to fire within the first 10 minutes of the event are not part of the design basis.
>
> This justification was based on the extreme unlikelihood of multiple spurious operations resulting in unacceptable coolant loss essentially coincident with loss of multiple mitigating systems within a 10-minute period...
>
> Upon confirmation of a fire in the plant, operating personnel will be dispatched to the SSF where they will establish communication with the control room.

Since time zero (t_0) for the event is not defined, the phrase "within the first 10 minutes of the event" could be interpreted to mean that no spurious operations are assumed to occur for 10 minutes following fire initiation. If a detector response time of 5 minutes is assumed, upon receipt of an alarm in the control room, operators would have 5 minutes to man the SSF prior to spurious actuations. This time would be reduced further if the time for operators to confirm the type and size of the fire is considered.

The stated justification (likelihood of fire damage causing unacceptable coolant loss coincident with loss of multiple mitigating systems within a 10-minute period) is limited to high/low pressure interface valves identified in the NRC staff's original RAI and does not bound the current assumption for all potential spurious operations.

The statement "Upon confirmation of a fire in the plant" infers that operators will man the SSF in response to all fires not just those that are deemed to be of a certain size, thus minimizing the time to enter the procedure. RAI 3-1 stated that "A Confirmed Active Fire is defined as a locally observed fire with smoke and either radiant heat or visible flame".

Regulatory Guide (RG) 1.205 and Industry Guidance Nuclear Energy Institute (NEI) 00-01

RG 1.205 states when the requirements in Chapter 4 of NFPA 805 are not met for the protection of required circuits, circuit analysis assumptions regarding the number of spurious actuations, the manner in which they occur (e.g., sequentially or simultaneously), and the time between spurious actuations should be supported by engineering analysis, test results, or both, that are accepted by the NRC.

Within NEI 00-01, Section 3, Deterministic Methodology, the requirements of Appendix R Sections III.G.1, III.G.2 and III.G.3 apply to equipment and cables required for achieving and maintaining safe shutdown in any fire area. Also assume that the fire may affect all unprotected cables and equipment within the fire area. This assumes that neither the fire size nor the fire

intensity is known. This is conservative and bounds the exposure fire that is postulated in the regulation.

Loss-of-Offsite Power

NEI 00-01 assumes the loss-of-offsite power has the potential to affect safe shutdown capability. In addition, the regulatory requirements for offsite power differ between the redundant and alternative/dedicated shutdown capability. Therefore, consideration must be given for the loss-of-offsite power when evaluating its effect on safe shutdown. The Appendix R requirement to consider a loss-of-offsite power is specified in Section III.L.3.

ONS Licensee Event Reports (LERs)

The NRC staff reviewed the following LERs associated with the SSF concerning the 10-minute assumption.

- LER 269/2006-004, Design Oversight Results in Appendix R Deficiencies (ADAMS Accession No. ML063470037)
- LER 269/2003-01 (8-4-2003), Design Oversight Results In Appendix R Control Cable Separation Issue (ADAMS Accession No. ML032250125)
- LER 269/2002-02 (5-28-2002), Potential for Fire to Indirectly Damage Mitigation Component (ADAMS Accession No. ML021580287)

The LERs documented concerns with the current licensing basis and the 10-minute assumption associated with the SSF.

Armored Control Cable Testing

The NRC staff witnessed ONS proprietary armored cable fire testing and observed that the armored cable material was combustible, had horizontal flame spread, and did not self-extinguish when the test burner was removed.

The NRC requests the following:

1. Provide the definition of a confirmed active fire requiring activation of the SSF.

2. Provide additional documentation that confirms the basis that spurious operations will not occur within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. acceptable engineering analysis or test results
    c. LERs
    d. armored cable fire tests

3. Provide documentation that confirms the basis that no loss-of-offsite power conditions occur within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. LERs
    c. armored cable fire tests
    d. the method used to meet NEI 00-01

4. Provide documentation that confirms the basis that fire growth will not have reached a point where fire damage will preclude operator actions from the control room within the first 10 minutes following the identification of a confirmed active fire and that the assumption remains valid in light of:
    a. "prior approval"
    b. LERs
    c. armored cable fire tests

RAI 3-38 RESPONSE:

It has been the ONS licensing position that the 10 minute allowance is a fundamental design concept for the SSF mitigation of a fire, which is supported by industry guidance (NEI 00-01), Duke/NEI cable testing, deterministic and Fire Probablistic Risk Assessment (FPRA) treatment, in addition to the context of these substantiating references:

- April 28, 1983 SER – established that the design of the SSF was to assume a period of 10 minutes would be available to mitigate potential spurious operations from the Control Room allowing operators time to activate the SSF.
- July 17, 1989 NRC Letter referencing Inspection Report 87-02 – "Based on the clarifications provided by DPC in the April 20, 1988 letter, and the results of an indepth Office of Nuclear Reactor Regulation (NRR) review of this issue, we have confirmed that the previous tacit acceptance of a ten minute delay in postulation of spurious signals in the NRC SER of April 28, 1983 is still valid."
- October 4, 1989 – NRC Inspection Report 89-27 – "...the staff concludes that the previous tacit acceptance of a 10 minute delay in postulation of spurious signals is still valid."

In conference calls with the NRC, Duke Energy was informed that they would not be allowed to carry the 10 minute allowance forward as licensing basis; thereby, eliminating the need to supply the originally requested information in this RAI. The NRC asked Duke Energy to evaluate elimination of the 10 minute allowance using risk informed processes. The information below provides the detail of that evaluation.

Duke Energy has initially identified additional variance from deterministic requirements (VFDRs) that require evaluation to account for scenarios that currently rely on a simplifying deterministic assumption (no spurious actuations within the first 10 minutes of confirmation of an active fire). For a variance to be supported by the risk informed process, its risk differential in terms of core

damage frequency must be calculated. Included as an attachment to this response is a sensitivity study, which provides the estimated delta core damage frequency (CDF) and delta large early release frequency (LERF) for these identified VFDRs. The acceptance of these new VFDR risk contributions must be less than the benefit received from the installation of PSW or other identified risk reducing modifications. This fire scenario is limited to the Auxiliary Building fire area since this is the only fire area which deterministically credited the SSF with the previous 10 minute allowance for no spurious operation or loss of power.

As part of the strategy to preserve risk margin, potential modifications are being considered. These modifications include elimination of select potential spurious actuations to reclaim the current risk contribution of the identified VFDRs. These modifications or others will be available in the event that significant margin in the FPRA is lost during transition or if invoked by the change evaluation process.

The loss of the 10 minute allowance for no spurious operation or loss of power with respect to deterministic analysis requires licensing disposition since it is a basic design concept associated with the SSF. With this change it will be ONS's position that 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.

UFSAR, section 3.1.11, criterion 11 specifies design and operation of the control room, which contains a reference to fire. It states:

> *"It shall be possible to shut the reactor down and maintain it in a safe condition if access to the control room is lost to fire or other cause."*

Current compliance with this criterion is possible due to the 10 minute allowance for no spurious operation or loss of power which makes transfer to the SSF successful. Elimination of the 10 minute allowance will prohibit ONS from deterministically complying with this requirement. FAQ 07-0032 (ML081400292) supports the basis of 10 CFR 50.48(c) meeting or exceeding the requirements of 10 CFR 50.48(a) and GDC 3. Duke Energy requests that the NRC approve that 10 CFR 50.48(c) also meets or exceeds the requirements of UFSAR criterion 11 for fire response.

Additional supporting activities for this effort will include revision of the B-2, B-3, Fire Risk Evaluation (FRE), FPRA, Nuclear Safety Capability Assessment (NSCA), and Operator Manual Action (OMA) Feasibility calculations. Also, previously docketed information concerning the 10 minute allowance (i.e., Executive Summary and Attachment T of LAR dated April 14, 2010 and RAI responses) will be superseded by the information contained in this RAI response.

In conclusion, Duke Energy 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 variances. Changes to the fire protection program, as a result of these variances, will be resolved using the change evaluation process. Upon completion of

this activity, which is being tracked through Duke Energy's corrective action program, all applicable FRE(s) will be updated and compliance will be demonstrated consistent with NFPA 805, Section 4.2.4.2.

Attachment 1: Delta Risk Resulting from Elimination of 10-minute Rule

Purpose: Perform sensitivity study to estimate the delta risk for four valves excluded from the FRE process due to deterministic application of the 10-minute rule. All of the non-compliances associated with these valves involve potential SSF vulnerabilities. The four valves are:

- RC-4 Failure to isolate from the SSF could jeopardize the ability of the SSF Reactor Coolant Makeup (RCMU) Pump to provide seal injection if the Power Operated Relief Valve (PORV) spuriously opens and does not re-close.
- HP-3 & 4 Failure to isolate letdown from the SSF could jeopardize the ability of the SSF RCMU Pump to provide seal injection if other downstream valves remain open.
- HP-20 Failure to isolate the seal return line from the SSF could jeopardize the ability of the SSF RCMU Pump to provide seal injection.

Analysis: The applicable Fire Area for each of these VFDRs is the AB, which relies on an SSF shutdown strategy. Each of the valves is analyzed for delta risk impact below.

1. The delta risk for failure to close RC-4 from the SSF due to Information Notice (IN) 92-18 damage resulting from spurious hot shorts was calculated for other fire areas by toggling off the spurious PORV basic event G00PORVDEX. The SSF is not credited for mitigation of a PORV LOCA (assuming the action to close the block valve from SSF is failed), so this potential non-compliance is adequately captured in the FPRA. RC-4 is operable from the SSF unless a spurious hot short has resulted in 92-18 damage. So the compliant case for the delta risk in the AB fire area can be estimated by toggling off the spurious PORV basic event G00PORVDEX.
2. Failure to close HP-3 and HP-4 is only modeled to preclude containment bypass under the LERF logic. Unisolated letdown leading to unmitigated loss of RCS inventory was not modeled in the PRA. The loss of inventory in excess of the RCMU pump capacity can be tolerated for some time. The eventual impact of unisolated letdown is emptying of the pressurizer and resulting loss of subcooling. There are numerous valves downstream of HP-3 and HP-4 which fail closed on loss of air or loss of power; however, no credit is taken for isolating letdown due to a consequential loss of air. HP-3 and HP-4 are operable from the SSF unless a spurious hot short has resulted in IN 92-18 damage. Also, while HP-5 is not controlled from the SSF, it receives an auto-close signal and fails closed on loss of air or power. A preliminary fault tree change was made to capture the additional SSF impact in order to estimate the delta risk. Accordingly, the delta risk in

the AB for this VFDR is estimated by toggling off new basic events ZHP3FIRDEX and ZHP4FIRDEX for the compliant case.

3. Failure to close HP-20 is modeled in the FPRA as an Inter System Loss of Coolant Accident (ISLOCA). An ISLOCA cannot be mitigated by the SSF. The inability to isolate seal return under non-ISLOCA conditions resulting in unmitigated loss of RCS inventory was not modeled in the PRA. Similar to the letdown pathway, the loss of inventory in excess of the RCM pump capacity can be tolerated for some time. The flow differential presented by the unisolated seal return pathway is much less than the letdown pathway. However, unlike the normal letdown pathway, the seal return pathway is not isolated by a fail-closed valve (HP-21 fails closed on loss of air but fails open on loss of power). HP-20 is operable from the SSF unless a spurious hot short has resulted in 92-18. Additionally, the individual seal return isolation valves (HP-226, 228, 230, and 232) are interlocked to automatically close if seal inlet flow to their associated RCP is low and either the pump is idle or component cooling flow to the thermal barrier heat exchanger is low. A preliminary fault tree change was made to capture the additional SSF impact in order to estimate the delta risk. Accordingly, the delta risk in the AB for this VFDR is estimated by toggling off HHP0020DEX, NHP0020MVT, and NHP0020MVC for the compliant case.

All preliminary changes made to the FPRA model in order to assess the delta risk of these potential VFDRs will be evaluated for permanent inclusion during the planned FPRA model update. The updated FPRA model will receive an industry peer review prior to NFPA 805 implementation.

Conclusion: The estimated delta CDF and delta LERF values for these potential VFDRs based on a modified preliminary Unit 3 FPRA model (as described above) are tabulated below:

VFDR	Delta CDF	Delta LERF
RC-4	1.3E-06	2.9E-08
HP-3 & 4	1.6E-07	8.0E-08
HP-20	2.1E-07	6.7E-08
Total	1.7E-06	1.8E-07

These VFDRs will be included in an update to the AB FRE. The acceptance of these new VFDR risk contributions must be less than the benefit received from the installation of protected service water (PSW) or other identified risk reducing modifications.

RAI 3-49:

The nuclear safety goal of NFPA 805 is to provide reasonable assurance that a fire during any operational mode and plant configuration will not prevent the plant from achieving and maintaining the fuel in a "safe and stable condition." NFPA 805 does not define a time period in which a "safe and stable condition" should be evaluated.

Section 4.2.1.2 of the LAR proposes that safe and stable conditions be deemed fulfilled if hot standby is achieved for 72 hours. This section of the LAR also indicates that long-term actions would be required to indefinitely maintain hot standby beyond the proposed 72-hour "mission time." However, it does not identify the specific actions that may need to be taken or describe why they are needed to maintain the fuel in a safe and stable condition.

Demonstrating the ability to maintain safe and stable conditions for only the first 72 hours following a fire does not, by itself, provide adequate assurance that the nuclear safety goal of NFPA 805 is *met.* The licensee should be able to demonstrate that safe and stable conditions can be maintained indefinitely, once achieved. The licensee would only need to discuss a time limit with respect to safe and stable if there was some physical limitation on the part of one or more plant systems/components that would result in the failure to be able to maintain safe and stable conditions (e.g., cooling water inventory would be depleted, battery capacity is limited, centrifugal pump may overheat as a result of too low flow, etc.). In this case, the licensee should discuss the limitations, and the basis for why that limitation will not have an adverse impact on the long-term ability to maintain the safe and stable requirements (e.g., by the time the plant/system problem occurs, additional resources/equipment will be made available, damage repairs can be made, etc.).

RAI 3-49 RESPONSE:

As documented within NFPA 805 Section 1.3, 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. As defined in NFPA 805 Section 1.6.56, 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.

As part of the transition to NFPA 805, each fire area was evaluated for a 72 hour coping period for maintaining safe and stable conditions at hot standby through either a "Deterministic Approach" (NFPA 805 Section 4.2.3) or "Performance Based Approach" (NFPA 805 Section 4.2.4). During the 72 hour coping period, the necessary systems and equipment have been evaluated to ensure their capability to achieve a safe and stable fuel condition as described within NFPA Section 1.3 and 1.6.56.

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 an RCS 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.

The scope of the needed repair activities is dependent upon the magnitude and location of a potential fire. Based on initial reviews, the scope of the most limiting set of repair activities at a high level may be summarized within the framework of the two basic mitigation strategies: Main Control Room or SSF Shutdown. For all fire areas except the Auxiliary Building, a main control room shutdown is utilized. For the Auxiliary Building fire area, an SSF shutdown is utilized. Based on preliminary reviews, the scope of repair activities for each mitigation strategy is summarized below:

Main Control Room Shutdown

- Restoration/Alignment of equipment associated with primary system depressurization (either PORV or Pressurizer Auxiliary Spray flow path)
- Restoration/Alignment of Reactor Vessel Head Vent flow path
- Restoration/Alignment of Core Flood Isolation capability
- Alignment of Atmospheric Dump Valves (ADVs)
- Restoration of control complex cooling
- Installation of the SSF Submersible Pump

SSF Shutdown

- Restoration/Alignment of Electrical Power for long term safe shutdown equipment
- Restoration/Alignment of High Pressure Injection (HPI) system (HPI motor replacement, HPI system valve repair/re-alignment)
- Restoration/Alignment of equipment associated with primary system depressurization (either PORV or Pressurizer Auxiliary Spray flow path)
- Restoration/Alignment of Reactor Vessel Head Vent flow path
- Restoration/Alignment of primary system sampling path
- Restoration/Alignment of Core Flood Isolation capability
- Alignment of Atmospheric Dump Valves (ADVs)
- Resupply of SSF Diesel Fuel Oil or Restoration/Alignment of PSW power to SSF

The mitigation strategy, damage assessment procedures, and repair equipment to maintain safe and stable conditions are to be established within a long term safe shutdown program. The long term safe shutdown program is to be a part of the scope of the NFPA 805 program. Based on the initial achievement of safe and stable conditions in conjunction with the availability of procedural actions, repair equipment, and initial coping period, the risk of recovery of the long term safe shutdown equipment is qualitatively deemed to have no significant measurable contribution to risk. The qualitative risk associated with the recovery of long term safe shutdown 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 onsite
- 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 within the allowable coping period of 72 hours provides reasonable assurance that the fuel will be maintained in a safe and stable condition.

The development and documentation of the long term safe shutdown program including analyses, equipment reviews, and procedural guidance is to be completed as part of implementation activities. Any changes that need to be made to the long term safe shutdown program during implementation will be resolved using the change evaluation process. An action item has been created in the ONS corrective action program to track completion of the long term safe shutdown program.

RAI 5-71c:

Question 5-71 requested the licensee to describe the process to confirm that the final estimates developed for the as-built, as-operated, proposed protected service water (PSW) system are consistent with, or is bounded by, the initial estimates. The description should include the quantitative criteria that will be developed and the relation of these criteria to the functional reliability estimates in the response to RAI 5-70. The process should include the actions to be taken if the final estimates cause the acceptable change-in-risk guidelines to be exceeded.

Licensee's Response

A table was provided in related RAI 5-70 that gives the change in core damage frequency (CDF) associated with each of the three PSW functions.

The installed PSW system will be treated similarly in maintenance rule space as the current ONS auxiliary service water system. Given that the proposed PSW system represents a significant risk offset for the risk of VFDRs associated with the transition to NFPA-805, the post-PSW system installation FPRA results will be compared to the current pre-PSW system installation FPRA results. The FPRA update will address additional changes not associated with installation of PSW system. If there is an increase of more than 10 percent in the current FPRA results, the individual scenarios that are driving that increase would be investigated. If the increase is attributed to a difference in PSW system credit, then the impacted fire risk evaluations will be revisited to ensure that the delta risk results remain within the range of the results used to support the previous conclusion.

NRC Staff's Comment

The proposed "monitoring" does not appear to be capable of monitoring the availability, reliability, and performance of the PSW system to ensure that the assumptions in the performance-based methods remain valid. The maintenance rule may be an acceptable monitoring process, but the statement that the PSW system will be treated similarly is inconclusive for the quantitative criteria. Conversely, a quantitative criterion of 10 percent in the current FPRA results of individual scenarios bears no clearly definable relation to the availability, reliability, and performance of the PSW system credited in the transition risk analysis. Changes to the scenario frequencies are the cumulative effect of changes to the facility, changes in the probabilistic risk assessment (PRA) models, changes in the PRA modeling assumption. Changes to the modeling assumptions, for example, could easily lead to reductions in scenario frequencies despite potentially large reductions in the PSW system availability, reliability, and performance.

Please provide the estimate for the availability of the PSW system functions credited in the current PRA.

RAI 5-71c RESPONSE:

The planned PSW modification was credited in the FPRA on a limited basis until the modification is installed and the actual cable routing and additional design/installation details are known. The FPRA assumptions regarding PSW are documented in response to RAI 5-70. It is expected that additional fault tree structure changes will be required to adequately capture the as-built configuration of the PSW system. Workplace Procedure XSAA- 106 governs the maintenance and update of the PRA models including the ONS FPRA. It is expected that in addition to modeling actual cable routes for the credited PSW functions, additional basic events will be added to address additional design details. Basic events representing equipment failure rates and unavailabilities will be updated as necessary. In that regard, credit for PSW prior to installation is no different than other FPRA model attributes. Given that PSW represents a significant risk offset for the risk of VFDRs associated with the transition to NFPA-805, the post-PSW installation FPRA results will be compared to the current pre-PSW installation FPRA results.

It is expected that credit for PSW has been underestimated in the FPRA. The basis for this assessment is as follows:

Since PSW is unaffected by Turbine Building fires, PSW was credited throughout the Turbine Building. However, since the precise routing of PSW related cables in the Auxiliary Building is not yet known, credit for PSW relative to providing power to an HPI pump or additional SSHR capability was not taken in Auxiliary Building fire scenarios. Accordingly, the fire risk is expected to improve since many fire scenarios in the Auxiliary Building may be able to credit PSW features after the modification is installed.

Additionally, the reliability inputs are considered to be either conservative or consistent with similar plant equipment modeled in the PRA. Each of the 3 PSW functions is discussed below. Failure to provide the PSW function is based primarily on 2 inputs: an operator action and a hardware failure. When not assumed to be failed by the fire, all of the PSW related failures are currently dominated by the operator action:

1) PSW will provide an additional power source for an HPI pump given failure of the normal power source (i.e., the 4kV switchgear located in the Turbine Building). Basic event PSW_HPI3B was installed under gate HPIP3BACASW to use in FRANC to turn on/off power to HPI pump 3B by location. This function is assumed to be failed in the Auxiliary Building. When the gate is not failed by the fire, failure to recover an HPI pump using PSW power is based primarily on 2 inputs:

    a. The operator fails to re-align power to the HPI pump from the PSW Switchgear (BHP0ASWDHE); the Human Error Probability (HEP) of 5.9E-02 is from the prior related action to power an HPI pump from the Auxiliary Service Water (ASW) switchgear and should improve post-installation since the action via the PSW switchgear is significantly less complicated.

    b. The PSW power source fails due to one or more hardware failures, most of which are to be located in the new PSW Blockhouse; a probability of 1.51E-03 was assigned to this hardware event (PSW_MOD) as described in the response to RAI 5-70.

2) PSW will provide an additional power source to support the SSF functions even with the SSF Diesel Generator unavailable. Basic event PSW_OTS1 was added under BSF_ALT to use in FRANC to turn on/off PSW power to the SSF (OTS1). This function was credited everywhere except at the SSF itself. It will also be failed in the PSW Blockhouse once that structure is included in the analysis. PSW power to the SSF is based on 2 inputs:

    a. The operator fails to align the SSF to the PSW power source (BSFAPWRDHE). A conservative HEP of 0.15 was assumed for this action. It's possible, perhaps even likely, that PSW power will actually be the preferred source and procedural enhancements will greatly reduce the likelihood of failure of this action.

    b. The PSW power source fails due to one or more hardware failures most of which are to be located in the new PSW Blockhouse; a probability of 1.51E-03 was assigned to this hardware event (PSW_MOD) as described in the response to RAI 5-70.

3) The PSW pump will provide additional Secondary Side Heat Removal (SSHR) capability by replacing the current station ASW pump. This function is assumed to be failed in the Aux Building by inclusion of a surrogate event within the Unknown Location (UNL) table.

When the function is not failed by the fire, failure to provide SSHR via the PSW pump is based primarily on 2 inputs:

a. The operator fails align PSW to provide secondary side heat removal (UPSWSHRDHE) from the Control Room. This basic event (HEP = 0.2) was added for aligning PSW as an alternate to Emergency Feedwater (EFW) at the following gates: TBU, TBX, TBQPU, TBQPX, TBQRU, TBQRX, TB1QPUD, TB1QPUX, TB1QPX, and for small LOCAS, BSL001. The HEP for aligning the Station ASW Pump locally was 5.5E-01; CEF0ASWDHE is a complex action that includes manual depressuration using the steam dump valves.

b. Use of the UNL table in lieu of specific cable routing information as well as the HEP for the above action obviates the need to model the PSW pump hardware as an input under each of the gates listed above. However, where modeled, PSW pump hardware reliability is dominated by the previous maintenance unavailability value of 2.2E-02 for the Station ASW Pump which will be replaced with the newer and larger PSW Pump. Similarly, PSW power is not currently modeled as an input to the PSW pump in the FPRA. Once the cable routing information is modeled and the UNL entries are removed, the simplified PSW structure will be replaced with a more detailed model to allow credit for the PSW pump in Aux Building fire scenarios with no PSW cable damage.

Finally, the FPRA modeling assumptions relative to PSW were provided to the PSW modification engineers to ensure that these important considerations are factored into the final design.

PSW and Fire PRA related assumptions were provided to the PSW group when the concern of alignment of NFPA 805 and the PSW project arose. The assumptions were grouped by the three main PSW functions credited by the FPRA: Alternate SSHR Capability, Additional SSF Power Source, and Alternate HPI Pump B Power Source. The assumptions are listed below:

Alternate SSHR Capability:

- The PSW pump will provide SSHR capability given loss of Main Feedwater (MFW) and EFW
- This capability will not be affected by a Turbine Building fire
- No FPRA credit for this capability will be taken in the Auxiliary Building until the PSW cables are included in ARTRAK with the requisite circuit analysis
- No outside Control Room (X-CR) actions are needed; all required actions are in the Control Room

Additional SSF Power Source:

- The PSW switchgear will provide additional power to SSF Bus OTS1.
- This capability will be in addition to the power provided to OTS1 from Unit 2 MFB 2 (normal) and the SSF DG (emergency).
- This capability will not be affected by a Turbine Building fire.
- This capability will not be affected in Auxiliary Building fire scenarios where an SSF response may be required (refer to RAI 3-2.2 response for additional related information).
- No operator actions are needed to provide this capability; PSW power is provided to OTS1 via normally closed breaker.

Alternate HPI Pump B Power Source:

- The PSW switchgear will provide alternate power to an HPI pump for each unit
- This capability will not be affected by a Turbine Building fire.
- No FPRA credit for this capability will be taken in the Auxiliary Building until the PSW cables are included in ARTRAK
- No outside Control Room (X-CR) actions are needed; the power transfer is performed in the Control Room.

During Oconee's NFPA 805 transition period, the engineering change process was updated to prompt consideration of potential impacts on the Fire PRA. If any Fire PRA concerns arise, the PRA department is informed, and the above assumptions are verified by the design engineers. This verification is documented in the PSW modification packages that have been approved to date.

Given that PSW represents a significant risk offset for the risk of VFDRs associated with the transition to NFPA-805, once the PSW modifications are installed, the risk benefit of PSW will be confirmed to ensure it continues to bound the cumulative VFDR risk. While it is expected that credit for PSW has been underestimated in the FPRA especially in the Auxiliary Building where limited credit for PSW was taken, the magnitude of the current risk benefit of PSW within the overall FPRA results cannot be guaranteed since part of that benefit may be derived from FPRA treatments that are subsequently refined. For example, PSW may be providing risk benefit for a scenario where a Y3 component was assumed to be failed. If cable routing is subsequently performed for that component and it is no longer failed in the scenario, then the PSW risk benefit may be reduced for that scenario. Consequently, the post-installation confirmation of PSW risk benefit will be limited to ensuring that PSW risk benefit continues to bound the cumulative risk of the VFDRs.

RAI 5-73c:

RAI 5-73 requested the licensee to provide the delta risk for all submersible pump deployment and activation actions.

Licensee's Response

Thus, if modeled, the delta risk associated with the failure to deploy the submersible pump is expected to be epsilon for fire events.

NRC Staff's Comment

The expectation of an impact is unusual. Without a quantitative estimate it is difficult to conclude that the licensee has identified the particular and detailed impact in the PRA and evaluated that impact.

Please provide a quantitative estimate of the delta risk associated with failure to deploy the submersible pump.

RAI 5-73c RESPONSE:

During NFPA 805 implementation, the deployment and operation of the SSF Submersible Pump will be evaluated as part of the scope of a VFDR involving available Condenser Circulating Water (CCW) inventory to support longterm SSF operation. The VFDR will be included in the scope of the FRE for Fire Area AB, Auxiliary Building (OSC-9327). The deployment of the submersible pump is only credited in the PRA for Turbine Building internal floods and is not currently credited for any fire scenarios. The buried Unit 2 CCW piping is the suction source for the SSF ASW pump, SSF HVAC Cooling Water pumps, and the SSF diesel engine water jacket pump. The flood scenario assumes most of Unit 2's buried CCW piping inventory is lost due to a pipe break. An action to deploy the submersible pump is necessary to replenish the buried CCW piping with lake water. The action is already proceduralized. The HRA for this action, NCWSUBPDHE, considers that this action must be completed with the pump submerged, connected, and operating (including the alignment of manual valves required for success of the submersible pump) after the loss of CCW forced and gravity/siphon flow.

In a fire scenario, flow is maintained to the CCW piping if either the CCW pumps or the siphon 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 Essential Siphon Vacuum (ESV) Systems are unavailable to maintain adequate siphon.

Since the action does not contribute to cutsets for fire scenarios, the action to deploy the submersible pump can be equated to failure of the SSF ASW pump to run (NSFPU02APR); the SSF ASW pump can operate after all of the submersible pump deployment conditions (low lake level, loss of CCW, and loss of ESV) are satisfied, so failure to start the SSF ASW pump is not affected. Importance measures for equipment from an integrated FPRA file are typically not used for assessing delta risk since the equipment may be failed by the fire and therefore cannot contribute to as many cutsets. But in this case, fire-induced failure of the SSF ASW pump is fairly uncommon and its importance may serve as a suitable surrogate for the importance of the action in lieu of adding the submersible pump to the fault tree for fire related event sequences.

In other words, the Fussel-Vessely (FV) value for NSFPU02APR represents the need for the SSF ASW pump, but the submersible pump is only needed if the lake level is low and CCW and ESV are both lost. Subtracting the contribution of NSFPU02APR when CCW or ESV is available (not impacted by fire) would be more representative. However, a satisfactory estimate can be derived without the added complexity of specifically identifying when both CCW and ESV are not available.

Accordingly, the delta risk may be conservatively estimated using the following equation:

$\Delta CDF = FV \times (CDF \div APR) \times LLL \times HEP$

Where: FV = 2.7E-04 for NSFPU02APR
CDF = U3 Total Fire CDF = 5.91E-05
APR = SSF ASW Pump random failure probability = 5.7E-04
LLL = Low lake level (NASWKLLDEX probability = 1E-02)
HEP = 8.1E-02 for NCWSUBPDHE

$\Delta CDF = 2.7E\text{-}04 \times (5.91E\text{-}05 \div 5.7E\text{-}04) \times 0.01 \times 0.081 = 2.3E\text{-}08$

The conservatively estimated delta risk is not insignificant but well within the risk acceptance criteria. This value is expected to bound the risk associated with the loss of CCW and ESV under the conditions that may require deployment of the submersible pump. Note that the importance of the SSF ASW pump was deemed a more suitable surrogate for the submersible pump than the importance of the SSF DG (NACSFDGSDR) since the SSF DG appears in cutsets where only the RCMU Pump may be needed (and more time is available before CCW inventory would be a concern) because the PSW installation will become the most likely preferred power source to energize the SSF.

If determined necessary by the FRE, the installation of the SSF submersible pump is expected to be acceptable with respect to manual action feasibility based on the existing procedures and associated periodic validations. Procedures provide detailed instructions on pump deployment and ensure that dedicated equipment and tools are maintained onsite. A Selected Licensee Commitment ensures that qualified individuals are available on each shift. Routine surveillance testing which installs the submersible pump in its expected emergency location and Emergency Planning drills validate the timing of actual pump deployment and simulated pump deployment respectively. During the implementation period, the NFPA 805 documentation will be updated as necessary to include the manual action feasibility associated with the installation of the SSF submersible pump pending the conclusions of the FRE. This is being tracked through the corrective action program.

RAI 5-75c:

RAI 5-75 requested the licensee to explain how fire detection and suppression are modeled in the risk assessment, how fire detectors are credited in this modeling, and why fire detectors are required to meet the risk criteria.

Licensee's Response

Other than a few cases where automatic suppression is credited, no credit for detection is provided in the quantification of fire risk for Oconee. However, the Oconee Fire PRA assumed no damage beyond the initial zone of influence provided the formation of a hot gas layer did not occur until after an assumed fire brigade response time of 20 minutes.

NRC Staff's Comment

The two sentences appear to be inconsistent. Assuming no damage beyond the initial zone of influence does credit detection in the quantification of fire risk. The licensee assumes that fire suppression is always successful based on rapid detection. Successful manual fire suppression is required or damage beyond the initial zone of influence can be expected. Therefore, detection and suppression are credited in the quantification of fire risk because non-suppressed fire scenarios leading to core damage are not included in the risk analysis. The impact of properly including these missing fire scenarios will always increase risk and could increase the risk substantially. Similarly, the change in risk estimates could be unaffected or could increase substantially depending solely upon the particular plant equipment configuration nearby each ignition source. Therefore, these are non-conservative assumptions.

Please identify the potential impact on the change in risk estimates of removing these non-conservative assumptions. If no impact is expected, please provide a discussion on how this conclusion is support based on fire development and other site specific analysis parameters.

RAI 5-75c RESPONSE:

Other than a few cases where automatic suppression is credited, no credit for detection is provided in the quantification of fire risk for ONS. However, the ONS FPRA assumed no damage beyond the initial zone of influence provided the formation of a hot gas layer (HGL) is interrupted due to fire brigade early intervention. Implicit in the initial generic response time applied in the FPRA is an assumption that fires would be detected within 15 minutes even in areas without automatic detection. The challenges inherent in identifying the occurrence of a fire via partial detection or the location of a fire based on equipment loss were discussed during the preparation of the fire risk evaluations. To address specific concerns, the generic response time assumption in the FPRA was replaced with individual fire compartment brigade response times calculated from the fire modeling data based on the specific compartment volumes. During the expert panel process applied to the defense-in-depth portion of the fire risk evaluations, areas were identified where additional fire detection was needed to strengthen the

time to detect the fire to assure the FPRA assumption is met. Accordingly, additional detectors were identified to meet the risk criteria even though no delta risk metric was exceeded.

As discussed in the ONS Fire Scenario Report, ONS fire detection and brigade response is related to room heatup. Room heatup has the potential to create a HGL and increase the zone of influence up to the point of room burnout. The zone of influence extracted from the generic fire modeling treatments would be impacted earlier during the event (HGL formation) than room burnout which occurs only after the critical room temperature is reached (damaging HGL condition). Additionally, HGL is an input to the multi-compartment analysis (MCA).

The evaluation of a potential HGL formation is comprised of a calculation that utilizes the Generic Fire Modeling Treatments report (Hughes report) which computes the HGL formation for constant heat release rate fires for given volumes. The results are shown in terms of the time required for the fire to cause the conditions at the target to exceed the threshold temperature given a fire size, a leakage fraction, and an enclosure volume. Each fire compartment is reviewed for its HGL formation potential. Fire compartments with significant openings are considered incapable of supporting the formation of a damaging HGL for room burnout. Another consideration for HGL formation is the potential impact on zone of influence even if the room opening(s) allows smoke and heat to escape. Fire compartments in open communication with the atmosphere or that are part of significantly larger volume, such as the fire zones within the Turbine Building, are considered invulnerable to HGL formation.

Compartments with no HGL potential are screened in the subsequent multi-compartment analysis (MCA). In these areas, such as the fire compartments within the Turbine Building, multi-compartment interaction would be limited to specific scenarios where the zone of influence extended to an adjacent zone. For compartments which do not screen based on the HGL evaluation, the MCA calculates the barrier breach probability and effectively assumes a conditional core damage probability (CCDP) of 1.0 for the resulting scenario. In this calculation, credit for automatic and manual suppression is applied to the compartment/scenario frequency in order to determine if a multi-compartment scenario should be added to the FPRA for additional refinement and subsequent quantification. A screening value of 1E-7 was used for the MCA calculation. The actual CDF associated with a scenario which breaches a barrier and results in damage to equipment from adjacent zones will likely have a lower CCDP than the 1.0 value assumed in the screening analysis.

Since the MCA is a screening analysis and not part of the actual FPRA quantification, credit for the associated detection is not considered to have a directly quantifiable contribution to fire risk. Similarly, credit for detection in the HGL evaluation is not directly quantifiable. Furthermore, there is considerable margin in the calculated brigade response times which precludes the need to incorporate probabilistic HGL scenarios into the FPRA. First, the peak heat release rate (HRR) values are conservatively assumed to be constant in the HGL evaluation; the peak HRR for a typical electrical cabinet fire occurs after 12 minutes. Second, the ignition source is conservatively assumed to contain sufficient combustible material capable of sustaining the peak HRR for the entire duration. Many fires are self-extinguishing (following removal of power or exhaustion of combustible products). Third, the calculated required response times per

compartment have been evaluated to be considerably longer than the actual fire brigade drill times. Additionally and perhaps most importantly, brigade response is only credited for changing the room conditions to interrupt HGL formation and not for successful fire suppression. Quick suppression is not a concern for limiting HGL; consequently, the non-suppression probabilities in NUREG/CR-6850 are not applicable to the likelihood of failure of brigade response. If a calculated required response time for a given fire compartment was determined to be unachievable or even improbable, then an HGL scenario for that compartment would need to be developed for inclusion and quantification within the FPRA.

The shortest calculated required brigade response time was 33 minutes for the Unit 1 and Unit 2 Equipment Rooms. These required response times are based on the potential increase in the zone of influence and not complete room burnout. Since 33 minutes was considered to be achievable for either the Unit 1 or Unit 2 Equipment Room, no scenario for failure of the brigade to respond was added to the FPRA. It should be noted that an increase in the applied zone of influence has not yet been determined to change the damage state. If a larger zone of influence is likely due to failure of the brigade to respond, then the additional targets, if any, would be identified at that time. The Equipment Room has full detection and is equipped with a fixed manually actuated sprinkler system. In the event of a severe fire in the Equipment Room, it was determined that the first responder could take actions outside the room if necessary to initiate the manual fixed suppression system that would favorably change HGL growth conditions in the room well before 33 minutes. It was also concluded that no additional failures would occur as a result of the manual actuation of the fixed suppression system.

The distribution of HGL formation times based on the more limiting condition of potential impact on the zone of influence is tabulated below:

HGL ZOI Impact Time (minutes)	Number of Fire Compartments	Room Characterization / Comments
33	2	U1 and U2 Equipment Rooms; HGL impact time based on electrical panel fires
40	4	U3 Equipment Room and U1/U2/U3 Battery Rooms; HGL impact time based on electrical panel / transient fires
44	2	U1 and U2 Cable Rooms; HGL impact time based on electrical panel fires
49	3	U1/U2/U3 Cask Decon Rooms; HGL based on transient fires
50 to 59	25	Fire compartments within the Aux Building including U3 Cable Room
≥ 60 or N/A	97	Includes "open" fire compartments (e.g., TB zones) and those which assume room burnout (e.g., CT4, KEO)

It is important to note that FPRA does not assume a fire and then calculate how big it can get. Instead, FPRA assumes a fire with a corresponding zone of influence and tries to come up with the proper frequency. A departure from desired best estimate results has already been incorporated into the FPRA by the use of conservative heat release rate profiles and conclusions based on test results rather than actual observed events. Most fires are not severe. Working with the station to risk-inform fire protection practices and features based on realistic FPRA model results provides the best opportunity for further improvement of fire risk. The HGL and MCA methods and assumptions will be a specific focus of the industry peer review of the updated ONS FPRA model.

RAI 6-2c:

The NRC staff requires follow-up information on your response to RAI 6-2, Monitoring Program. The response did not provide the criteria being used to determine the structure, system, and components (SSCs) included in the monitoring program. Specifically, the response states that performance monitoring goals (PMGs) in high safety significant fire zones will be monitored. Please provide the following:

1. Define "high-safety significant" fire zones and the criteria (e.g., core damage frequency (CDF), large early release frequency (LERF), etc.) used to select SSCs to be monitored.

2. Clarify how SSCs credited to meet the nuclear safety performance criteria but not located in a "high-safety significant" fire zone are evaluated for inclusion in the monitoring program.

3. Clarify how SSCs credited for meeting the radioactive release performance criteria are evaluated for inclusion in the monitoring program.

4. Describe the process for how system failures are evaluated to determine if the target reliability and/or action levels have been exceeded.

5. The response states that Electric Power Research Institute (EPRI) 1006756 will be used to develop availability and reliability criterion for PMGs. With regard to the EPRI document, provide the following:

   a. The EPRI document provides the results of an analysis of FPP system availability for several plants, but does not provide any specific recommendations for target availability. Describe how target availability values are developed for ONS.

   b. The EPRI document provides the recommended target reliability values: 1) valve position checks – 99 percent, and 2) all other activities – 95 percent. Clarify that these are the initial target reliability values being used at ONS. If not, describe how target reliability values are being developed (e.g., is plant-specific data being used).

   c. The EPRI document provides the recommended test/inspection frequencies for various FPP systems. Clarify that these are the initial test/inspection frequencies being used at ONS. If not, describe how test/inspection frequencies are being developed (e.g., is plant-specific data being used). Justify the use of these test/inspection frequencies in lieu of the frequencies specified in applicable NFPA codes (e.g., NFPA 72 for fire detection systems).

   d. The EPRI document provides the recommended action level values: 1) valve position checks – 97 percent, and 2) all other activities – 92 percent. Clarify that these are the initial action level values being used at ONS. If not, describe how action level values are being developed (e.g., is plant-specific data being used).

e. The EPRI document provides failure criteria for the various FPP systems. Clarify if these failure criteria are being used at ONS or provide the alternative failure criteria being used.

RAI 6-2c RESPONSE:

High Safety Significant Fire SSCs are those that support the risk significant fire compartment screening criteria, and those Fire Protection SSC's amenable to risk measurement are included for each fire compartment. The screening thresholds used to determine risk significant (HSS) fire compartments are:

- Core Damage Frequency (CDF) ≥ 1E-7 or
- Large Early Release Frequency (LERF) ≥ 1E-8 or
- Risk Achievement Worth (RAW) ≥ 2

Various structures, systems and components (SSC's, other than fire protection components) perform functions or support assumptions that may be credited in Nuclear Safety Capability Assessment (NSCA), Fire Probabilistic Risk Analyses (FPRA), or other NFPA 805 supporting analyses in order to reduce risk due to fire in a particular fire zone or fire area. The NFPA 805 SSCs credited to meet the Nuclear Safety Performance Criteria can be impacted by fire in multiple fire zones and are more suited to monitoring at a system functional level. These SSCs will be evaluated to determine if additional monitoring of these components is required. Typically, NSCA equipment is monitored as part of Maintenance Rule. SSC's, designated as NSCA equipment that is credited in the NFPA 805 analyses, are being reviewed to validate that the equipment and its NFPA 805 function, including differing methods of control, are monitored as part of the Maintenance Rule and that the criteria is adequate to meet the needs of NFPA 805. New Maintenance Rule functions will be created and applied where necessary to support NFPA 805 and ongoing assessment of fire risk.

Radioactive release performance criteria is programmatic in nature and per FAQ 09-0056 is related to manual fire fighting activities, fire brigade training, engineering controls and fire strategies (Pre-plans). No specific SSC's are credited for meeting the radioactive release performance criteria and, therefore, none are included in the monitoring program. Monitoring of the radioactive release performance criteria will be accomplished via monitoring of Fire Brigade performance programmatically.

A software program still under development will be used to collect real time data from the corrective action program (PIP), Work Order Tracking and Fire Impairment Log to calculate values of availability and identify failures for reliability of SSCs identified as being monitored by NFPA 805 in the Oconee Equipment Database. The program will provide alerts of target values being approached. The appropriate Fire Protection or System engineer will evaluate the data to determine if it is applicable to the performance targets to validate if a target value is being approached or has been exceeded. The ONS NFPA 805 Monitoring Program would require that failure to meet availability and/or reliability criteria results in the initiation of a Problem Investigation Process (PIP), which documents the cause and resolution of the performance shortfall by establishing performance goals and corrective actions to return the component or PMG into compliance with established criteria, similar to the Maintenance Rule.

Initial availability and reliability criteria values for PMGs are established in the ONS NFPA 805 Monitoring Program Scoping Calculation. The calculation review process currently underway is being used as a surrogate for the expert panel process and will determine the final target values. The target and action values for availability will be primarily based on site specific data relative to 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. Reliability data will not be screened out based on compensatory measures. Target and action reliability values will rely primarily on the EPRI 1006756 values provided, tempered by site-specific operating experience, FPRA assumptions, and equipment types (and vendor data when available). The inspection and test frequencies being used to gather data are those currently contained in the Selected Licensee Commitments (SLC's). Inspection and test acceptance criteria (failure criteria) will also be initially based on the system design, manufacturer's criteria, and NFPA code requirements. It is recognized that this criteria may require adjustment by the System/Program Engineer or multidisciplinary review team as the Monitoring Program becomes established and monitoring data is gathered over a period of time. As more performance data is obtained, frequencies may be adjusted using the performance based process described in EPRI 1006756, the NEIL underwriting guidelines, and applicable NFPA codes such as NFPA 72. The target values of availability and reliability for NFPA 805 monitored components will be selected, reviewed and maintained to ensure that the assumptions of the applicable supporting analyses (FPRA, NSCA, etc.) remain valid.