Enclosure 2

REDACTED VERSION OF 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 2

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REQUEST FOR ADDITIONAL INFORMATION (RAI) 1-12:

License Amendment Request (LAR) Attachment N states that "an editorial error is also being corrected in Technical Specifications (TS) Bases 3.10.1, Standby Shutdown Facility." It is unclear from the description and the TS markup what editorial error is being corrected. Clarify.

RAI 1-12 RESPONSE:

The editorial error correction in Attachment N refers to the addition of the article 'a' in place of the 10 CFR 50 Appendix R deletion such that the bases will read grammatically correct. Reference Attachment N marked-up bases pages B 3.10.1-1, B 3.10.1-6, and B 3.10.1-9.

RAI 1-13:

Attachments N and S of the LAR provide the implementation schedule for committed plant modifications being made to bring Oconee Nuclear Station (ONS) into compliance with National Fire Protection Association (NFPA) 805. The implementation schedule for upgrading modifications range from February 2012 to the fall of 2015. Provide additional information for modifications that extend more than 2 years after issuance of the Nuclear Regulatory Commission (NRC) safety evaluation (SE) .

Specifically provide:

- a discussion of the complexity of each modification and associated schedule drivers,
- a justification of why the modifications cannot be implemented by the second outage or within 2 years following issuance of the NRC SE, whichever is sooner,
- a discussion of, and justification for, the "appropriate compensatory measure" for those modifications extending beyond the 2 years following the NRC SE.

RAI 1-13 RESPONSE:

In response to bullets 1 & 2:

Modifications listed in the submittal dated April 14, 2010,Table S-1 that may extend into 2012 (and beyond) include Items 2, 3, 4, & 5. Items 2, 3, and 4 pertain to fire barrier modifications. The scoping for the fire barrier modifications was provided in the submittal dated May 31, 2009. The fire barrier modifications are expected to require a significant amount of field work involving scaffolding erection/disassembly and asbestos concerns. Item 5 pertains to installation of additional fire detection. A brief description of scope and complexity is provided as follows:

Based on an initial review of the scope of the modification, the installation of the significant number of fire detectors will require cable installation throughout a large portion of the plant to

route the detector outputs to the panel(s) for operator monitoring. The installation is expected to require a significant amount of elevated work (scaffolding, etc.) and wall/floor penetrations (core drills). As such, detailed modification planning will be necessary to avoid interactions with station systems and equipment. Preliminary reviews have already identified interactions which will require work to be performed during outage periods due to the proposed location of certain detectors. Based on the complexity of the significant number of cable routings and detector installations, careful planning is necessary to minimize plant risk from engineering errors. To allow sufficient time for adequate review, it is proposed that the normal modification template be utilized for the fire detector modifications. These modifications are Major Design Changes. The Business Rules for determining the slotting for these Design Changes is similar to the below except the duration. The time between the creation of the Engineering Change Request to having planned work orders that meet the Work Control Milestone is 36 months.

After the scope was finalized for the modifications, slotting was determined using the following set of Business Rule assumptions:

1. All design and planning are "on template" meaning it takes 15 months from start of design to planning complete. This duration is applicable to Minor Design Changes.
2. Design and planning are staggered as much as possible to prevent resource overloads.
3. All Work Control milestones for modification approval and work order planning are met.
4. For items 2 and 3, there will be a dedicated maintenance team that will work each of the unit's sequentially.

Following this process resulted in the modifications listed below to not be completed within 2 years of the issuance of the NRC safety evaluation:

- Item 2 - Unit 1 to be completed by June 30, 2013.
- Item 3 - Unit 1 to be completed by October 31, 2013; Unit 2 by February 28, 2014; Unit 3 by June 30, 2014.
- Item 5 - Units completed by 1EOC28; 2EOC27; 3EOC27, respectively.

In response to bullet 3:

The compensatory measures for those areas in which a fire barrier modification will be performed (Table S-1, Items 2, 3, & 4) are station fire surveillances. Post-transition, the fire surveillance interval would be determined in accordance with ONS Site Directive (SD) 3.2.14, Fire Protection Program Compensatory Measures. SD 3.2.14 includes a specific flow chart for determining compensatory measures for fire barrier impairments. The flow chart asks a series of questions regarding the conditions relevant to the impaired fire barrier. Depending on the activities, risk, and available suppression and detection in the area a series of compensatory measures are outlined. Compensatory measures include fire surveillances of various durations from set hours to continuous and staging of additional manual suppression sources.

The compensatory measures for those areas in which fire detection will be installed as part of the modifications outlined in Table 4-4 and S-1, Item 5 are station fire surveillances. Post-transition, the fire surveillance interval would be determined in accordance with ONS SD 3.2.14, Fire Protection Program Compensatory Measures, post-transition. SD 3.2.14 includes a specific flow chart for determining compensatory measures for fire detection impairments. The flow chart asks a series of questions regarding the conditions relevant to the impaired fire detection. Depending on the activities, risk, and suppression available in the area a series of

compensatory measures are outlined. Compensatory measures include fire surveillances of various durations from set hours to continuous and staging of additional manual suppression sources.

These are the standard compensatory measures when a fire protection feature is impaired. For fire barriers, the fire barrier will be considered impaired until the modification is complete. For fire detection, the areas in which the new fire detection is to be installed will be considered impaired (as if the detectors were installed but not operable) until the new detection is installed and placed in service.

RAI 1-14:

Attachments N and S of the LAR provide the implementation schedule for committed plant modifications being made to bring ONS into compliance with NFPA 805. The attachments provide the discussion of the breaker coordination study and provide an end date for the study. Provide the details and schedule for the modifications required to resolve breaker coordination study issues.

RAI 1-14 RESPONSE:

The following are the details and schedule for the modifications required to resolve coordination issues that have been identified to be risk significant according to the Fire Probablistic Risk Assessment (FPRA) model:

Four specific breaker cubicles have been identified in the FPRA as having an overall risk increase due to their lack of coordination with the upstream protective device. In order to maintain the current FPRA risk profile, these four breakers must be modified to ensure adequate breaker coordination with the upstream breaker or other means utilized to ensure coordination with the upstream breaker. The breakers are:

<u>MCC# 3XB</u>

- Breaker 1BB
- Breaker 1BT
- Breaker 2BT

<u>250/125VDC Distribution Center# 3DP</u>

- Breaker F3CL

The recommended modification requires that the above breaker compartments each have a new fuse and holder installed and wired.

Other means to ensure coordination may be substituted provided that it does not impact the results of the FPRA model.

This modification will be installed during the next Unit 3 EOC26 refueling outage (spring 2012).

RAI 2-16:

The protected service water (PSW) Fire Area in Table B-3 of the LAR identifies that detection is not required and states that it will be installed with the PSW modification. The introduction to Table B-3 where the PSW modification is discussed (page C-5) makes no mention of fire detection as part of the modification. However, Table 4-4 (page 54) indicates for the PSW fire area that fire detection is required for defense-in-depth but does not identify that a modification is required. Clarify this discrepancy and describe any fire detection that is required and any modifications that are being performed.

RAI 2-16 RESPONSE:

For the PSW Fire Area (PSW building and duct banks), Table B-3 is correct that detection is not required. Detection is being installed in the PSW Fire Area, as part of the overall PSW modification, for property protection/insurance purposes. Table 4-4 (page 54 of the LAR submittal) is incorrect and should not have indicated that detection is required for defense-in-depth. Page 54 of the LAR has been updated and is provided below.

Note that cables and equipment related to PSW located in areas other than the PSW structure and duct banks are considered part of the fire area in which they are located (e.g. PSW cables and equipment located in the Auxiliary Building (AB) is considered part of the AB Fire Area).

Table 4-4 Summary of NFPA 805 Compliance Basis and Required Fire Protection Systems and Features

Fire Area	Fire Zone	Description	NFPA 805 Regulatory Basis	Required Suppression System (E, R, D, S)	Required Detection System (E, R, D, S)	Required[3] Fire Protection Feature (E, R, D, S)	Required Fire Protection Feature and System Details[1]
KEO	**KEO**	**Keowee Hydro Station**	**4.2.3.2**			**None**	
PSW	**PSW**	**Protected Service Water Building**	**4.2.3.2**			**None**	
RB1	**122**	**U1 Reactor Building – Basement thru 4th Floor**	**4.2.4.2[4]**		**D**	**None**	**Detection – all (both ionization and heat)**
RB2	**123**	**U2 Reactor Building – Basement thru 4th Floor**	**4.2.4.2[4]**		**D**	**None**	**Detection – all (both ionization and heat)**
RB3	**124**	**U3 Reactor Building – Basement thru 4th Floor**	**4.2.4.2[4]**		**D**	**None**	**Detection – all (both ionization and heat)**
SSF	**SSF**	**Standby Shutdown Facility**	**4.2.4.2[4]**		**D**	**None**	**Detection – all (Honeywell panel detection)**
TB		**Turbine Building**	**4.2.4.2[4]**				
TB	1	Unit 3 Lube Oil Purifier Area			R	None	Detection – all
TB	2	Unit 3 EHC Area			R	None	Detection – all
TB	3	Unit 3 Heater Bay Area			R	None	Detection – over heater drain pumps[2]
TB	4	Unit 3 Turbine Driven EFDW Pump Area			R	None	Detection – all
TB	5	Unit 3 Condensate Booster Pump Area				None	
TB	6	Unit 3 Main Feedwater Pump Area		R	R	None	Suppression – Main Feedwater Pump 3A and 3B water spray systems Detection – over SWGR[2]
TB	7	Unit 3 Motor Driven EFDW Pump Area			R	None	Detection – all
TB	8	Unit 3 Hotwell Pump & TB Sump Area			R	None	Detection – all
TB	9	Unit 3 Powdex/LSPW Pump Area			R	None	Detection – all
TB	10	Unit 2 Lube Oil Purifier Area			R	None	Detection – all
TB	11	Unit 2 EHC Area			R	None	Detection – all
TB	12	Unit 2 Heater Bay Area			R	None	Detection – over heater drain pumps[2]
TB	13	Unit 2 Turbine Driven EFDW Pump Area			R	None	Detection – all
TB	14	Unit 2 Condensate Booster Pump Area				None	
TB	15	Unit 2 Main Feedwater Pump Area		R	R	None	Suppression – Main Feedwater Pump 2A and 2B water spray systems Detection – over SWGR[2]
TB	16	Unit 2 Motor Driven EFDW Pump Area			R	None	Detection – all
TB	17	Unit 2 HW Pump, LPSW Pump - B Area			R	None	Detection – all

RAI 2-17:

The PSW Modification is credited in the FPRA for NFPA 805 transition. Describe the fire protection design features and modifications being implemented specifically to meet fire protection criteria. For example, describe the design constraints on cable routing, location of credited fire barrier installation/upgrades, and detection requirements. Include any credited detection and suppression system modifications/new construction being made to conform to the assumptions and limitations of the FPRA and/or to meet NFPA 805.

RAI 2-17 RESPONSE:

PSW is designed as a standby system for use under emergency conditions when plant systems in the Turbine Building (TB) are lost. PSW is a dedicated system, not reliant on other normal plant systems. Fire protection design features incorporated into the PSW modification include fire detection and fire barriers. There are no suppression systems included as part of the PSW modification.

Fire detection includes installation of a fire detection system in the PSW Blockhouse that will alarm to the Unit 3 Control Room and additional detectors installed in the AB. The fire detection system in the PSW Blockhouse is not required, but installed for insurance/property protection. New PSW cables are being routed in the AB where existing detection is not deemed sufficient and electrical cabinets that meet the definition of a NUREG-6850 ignition source exist. As a result, additional detectors are being added in affected AB fire zones. Detection modifications are identified in Table 4-4 of the April 14, 2010 LAR and identified as required to meet the risk criteria for NFPA 805, Section 4.2.4.

PSW cables include a new connection from the Keowee cable trench to the PSW Blockhouse, cables from the PSW Blockhouse to the AB and Standby Shutdown Facility (SSF) via a new duct bank, and new cables routed within the AB. New cables installed as part of the PSW modification in the AB are located in accordance with DC 3.13 "Oconee Nuclear Station Cable and Wiring Separation Criteria" which provides guidance on cable routing and installation (including physical independence and separation criteria). General criteria for installation of new PSW cables includes: cables are not routed in the TB or near SSF cables. This precludes placement in the West Penetration Fire Area. PSW cables are generally routed in the AB hallways, the Equipment Rooms, Cable Rooms, Cable Shafts, and Control Rooms around elevations 771', 783', and 796'. For those areas, equipment installed by the PSW modification will not be credited for maintaining the plant safe and stable following a fire. Those components whose normal controls and power are located outside of these areas (i.e. pre-transfer) have been evaluated under NFPA 805, Section 4.2.4.

Fire barriers that are part of the PSW system include the PSW Blockhouse, which is a bunkered facility to the south of main plant structures, the underground duct banks/manholes/cable trenches with respect to the yard, and the TB/AB wall which is being upgraded for overall fire area separation, but is also important to PSW. Each of these fire barriers are discussed in ONS Calculation, OSC-9302.

RAI 2-18:

Each of the approval requests identified in LAR, Attachment L provides a very brief conclusion regarding maintaining safety margins and defense-in-depth. These conclusions need to be supported by a detailed evaluation that considers all aspects of fire protection defense-in-depth

and maintaining safety margins, similar to the evaluation performed for the fire risk evaluations and per the guidance in regulatory guide (RG) 1.205, Rev. 1, and the Nuclear Energy Institute (NEI) 04-02, Rev. 2. For each Attachment L approval request, provide an evaluation that considers all aspects of fire protection defense-in-depth and safety margins.

Similarly, a one-statement conclusion is often all that is provided regarding the impact on the radiological release criteria, which is generally that there is no impact. Provide an evaluation for each approval request that supports these conclusions.

RAI 2-18 RESPONSE:

The LAR Attachment L Approval Requests were modified as shown in the following markups using revision tracking to identify the enhanced discussion of safety margin, defense-in-depth, and radiological release conclusions. Note that the Approval Requests that follow are the original requested text. The body of the response to some approval requests may be modified by other RAI's. The following discussions of safety margin, defense-in-depth, and radiological release take into account the responses to other RAI's.

Approval Request 1

NFPA 805 Section 3.3.1.2(1)

NFPA 805 Section 3.3.1.2(1) states:

> *"Wood used within the power block shall be listed pressure-impregnated or coated with a listed fire-retardant application.*
> *Exception: Cribbing timbers 6 in. by 6 in. (15.2 cm by 15.2 cm) or larger shall not be required to be fire-retardant treated."*

Duke Energy Nuclear System Directive (NSD) entitled "Control of Flammable and Combustible Materials" states that untreated wood for concrete forming and dunnage is permitted. The Directive defines dunnage as large dimension timber materials used for support of heavy equipment. Dunnage is not typically impregnated with flame retardant materials, but is allowed on site due to its mass, density, and fire characteristics of being difficult to ignite. This would meet the exception of NFPA 805 Section 3.3.1.2(1) above of 'cribbing'. Therefore the deviation from the requirements of NFPA 805 Chapter 3 is that non-pressure impregnated or fire-retardant (untreated) wood used for concrete forming is permitted in the plant.

In limited cases where wood is installed as a permanent installation by design (e.g. EHC snubber supports), the installation is reviewed for impact of fire protection program per NFPA 805 Section 3.2.3(4).

The basis for the approval request of this deviation is:

- In some cases, the chemicals used in the treatment of fire-retardant wood affect concrete curing.
- A small quantity of untreated wood used for concrete forming is acceptable because the magnitude of the additive combustible material would be insignificant as compared to the total fire load in the area.
- The locations of concrete forming are generally not in close proximity to ignition sources.
- Concrete forming is for temporary use and not for permanent plant installation.

Nuclear Safety and Radiological Release Performance Criteria:

The use of untreated wood as concrete forming does not affect nuclear safety as concrete forming is not frequently used in the plant. When concrete forming is used, it is in small quantities with negligible impact to the in-situ fire load and within the permissible transient fire load. If the quantity of untreated wood exceeds the permissible limits in NSD entitled "Control of Flammable and Combustible Materials", a Fire Protection Engineer (FPE) review would include identifications of any special precautions or limitations on the use of the untreated wood. Therefore there is no impact on the nuclear safety performance criteria.

The use of untreated wood as concrete forming has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the introduction of untreated wood, or other combustible transient materials. The introduction of untreated wood does not change the radiological release evaluation performed that potentially contaminated water is contained and smoke monitored. Untreated wood does not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

Addition of untreated wood as concrete forming is minimal and/or the quantity is reviewed with special precautions or limitations indentified as necessary in order to minimize fire risk. These precautions and limitations on the use of these materials have been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore; the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control, and extinguish fires that do occur; thereby, limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The introduction of untreated wood as concrete forming does not impact fire protection defense-in-depth. Echelon 1 is maintained by the compensatory measures required by directives when introducing a transient combustible material (combustible control program). The introduction of untreated wood does not affect echelons 2 and 3. Introduction of untreated wood is administered under the control of combustibles program. The introduction of untreated wood as concrete forming does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability. Quantities of untreated wood as concrete forming cannot be introduced such that they may challenge any elements of the fire protection program without appropriate compensatory measures being identified during the work review process.

Conclusion:

NRC approval is requested for approval of the use of untreated wood for concrete forming. 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).

Approval Request 2

NFPA 805 Section 3.3.1.3.4

NFPA 805 Section 3.3.1.3.4 states:

"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."

The use of portable heating devices is controlled by Duke Energy NSD entitled "Fire Protection and Impairment Surveillance." The Directive defines a heating device as "A temporary heater that uses greater than or equal to 220 Volts or is powered by a fossil fuel source and is used to heat a general area, enclosure or equipment." Site personnel are responsible for notifying the FPE of any intended use of a heating device(s). The Directive specifies only steam heaters, UL Listed or FM Approved electric heaters, gas (MAPP, LPG, or natural) or oil-fired heaters shall be allowed inside buildings. Gasoline and other fuels are not permitted. Specific instructions for the placement of the temporary heaters are provided in the NSD. Therefore the deviation from the requirements of NFPA 805 Chapter 3 is that fuel-fired heaters are permitted in plant areas without limitation with respect to the location of equipment important to nuclear safety.

ONS SD entitled "Fire Protection Program Compensatory Measures Process" provides a flow chart for compensatory measures to take when temporary heaters are installed in the plant. This flow chart includes consideration of location and risk. In most cases the compensatory action is a fire watch. The group that requests the placement of the temporary heating device shall be responsible to ensure that fire watches are initiated.

The basis for the approval request of this deviation is:

- Site personnel are responsible for notifying the FPE of any intended use of >220V and fossil fuel heating devices. Instructions for the locations and placement are documented.

- ONS SD "Fire Protection Program Compensatory Measures Process" provides a flow chart for compensatory measures to take when temporary heaters are installed. This includes consideration of location and risk. Compensatory measures are generally fire watches at varying frequencies dependent on the location, risk, and existing fixed fire protection features (suppression/detection).

Nuclear Safety and Radiological Release Performance Criteria:

The use of portable fuel-fired heaters does not affect nuclear safety as appropriate compensatory measures are taken when portable fuel-fired heaters are used in the plant. Compensatory actions include the evaluation of the area for risk significance. If equipment

important to nuclear safety is in the area requiring a portable fuel-fired heater, restrictions on the location and an increase frequency of fire watches will be implemented. Therefore there is no impact on the nuclear safety performance criteria.

The use of portable fuel-fired heaters has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the introduction of a portable fuel-fired heater, or any other combustible transient material or ignition source. The introduction of portable fuel-fired heaters does not change the radiological release evaluation performed that potentially contaminated water is contained and smoke monitored. Portable fuel-fired heaters do not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

Compensatory actions are implemented in order to minimize fire risk. These compensatory actions and limitations on the use of portable fuel-fired heaters have been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control, and extinguish fires that do occur; thereby, limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The introduction of portable fuel-fired heaters does not impact fire protection defense-in-depth.

Echelon 1 is maintained by the compensatory measures required by directives when introducing a portable heater. The introduction of portable heaters does not affect echelons 2 and 3. Introduction of portable fuel fired heaters is administered by NSDs and SDs. The introduction of portable fuel-fired heaters does not result in compromising automatic or manual fire suppression functions fire protection for systems and structures, or post-fire safe shutdown capability. Appropriate compensatory measures are implemented under the directives.

Conclusion:

NRC approval is requested for the use of temporary fossil fuel heaters as specified in the NSD entitled "Fire Protection and Impairment Surveillance", and Site Directive entitled "Fire Protection Program Compensatory Measures Process."
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).

Approval Request 3

NFPA 805 Section 3.3.7.1

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."*

NFPA 805 Section 3.3.7.1 states:

> *"Storage of flammable gas shall be located outdoors, or in separate detached buildings, so that a fire or explosion will not adversely impact systems, equipment, or components important to nuclear safety. NFPA 50A, Standard for Gaseous Hydrogen Systems at Consumer Sites, shall be followed for hydrogen storage."*

Bulk storage of flammable gas is located in various areas of the plant by design. This includes storage of flammable gas cylinders for chemistry labs and reference gases for Post Accident Monitoring instrumentation.

ONS maintains various sizes of compressed gas cylinders. A walkdown validated that most cylinders are size A, although some size K and some smaller cylinders were observed. Size A cylinders contain approximately 195 ft^3 of gas and size K cylinders contain approximately 260 ft^3 of gas. Therefore more than 2 cylinders of gas located in one area would constitute bulk storage. A Bulk Hydrogen Compressed Gas System is defined (in NFPA 55) as:

> *"An assembly of equipment, consisting of but not limited to, storage containers, pressure regulators, pressure relief devices, vaporizers, manifolds, and piping, with a storage capacity of more than 400 ft^3 (scf) (11 m3) of compressed hydrogen gas including unconnected reserves on hand at the site. The bulk system terminates at the point where the gas supply, at service pressure, first enters the supply line. The containers are either stationary or movable, and the source gas is stored as a compressed gas."*

Safe permitted use of compressed gases is controlled by operational procedures. The storage of combustible and flammable materials is controlled by administrative procedures. General provisions for the use and storage of chemicals, including gases, are controlled by Duke Energy NSD entitled "Nuclear Chemical Control Program."

In the Response to Appendix A to Branch Technical Position APCSB 9.5-1 dated February, 1982, ONS stated that there is no bulk gas storage in areas affecting safe shutdown equipment. Under NFPA 805, the requirement is that there is no bulk storage permitted inside structures housing systems, equipment or components important to nuclear safety. Therefore the deviation from the requirements of NFPA 805, Chapter 3 is that ONS has flammable gas cylinders meeting the definition in NFPA 55 of bulk within structures (AB) housing equipment important to nuclear safety.

Chemistry labs use hydrogen on a daily basis and have reserve tank(s) staged for continued use. The Chemistry labs and reference gases are located in the AB at ONS which also contains systems, equipment, and components important to nuclear safety. The hydrogen cylinders are stored in accordance with ONS procedures. Hydrogen gas is also used as a reference gas for Post Accident Monitoring equipment located in the Unit 1/2 and Unit 3 HVAC rooms.

The basis for the approval request of this deviation is:

- In some cases, the staging of flammable gas cylinders is required in structures which house systems, equipment, or components important to nuclear safety. Typically one

bottle is connected to the system and the minimum number of required bottles are staged in the area for continued use.

- The flammable gas cylinders discussed in this request are existing in the plant and any new locations will have a fire hazard review prior to installation.
- Gas cylinders staged but not in use are segregated and stored in accordance with ONS administrative procedures and design review processes.
- The flammable gas cylinders are stored in locations that do not impact equipment important to nuclear safety.
  - The Chemistry Labs are located on the 796' elevation of the Unit 2 and Unit 3 AB (Fire Zones 90 and 86). The area is predominantly labs, offices and work areas. These rooms have been analyzed in the current FPRA in their current configuration which includes the presence of the flammable gas cylinders. PRA concluded hydrogen fires in these areas do not impact any targets.
  - The Post Accident Monitoring instrumentation is located on the 838' elevation of the AB in the Unit 1/2 Air Handling Unit (AHU) and Unit 3 AHU rooms (Fire Zones 119 and 116). These rooms contain air handling equipment, gas analyzers/monitors, breathing air compressors, and exhaust fans. These rooms have been analyzed in the current FPRA in their current configuration which includes the presence of the flammable gas cylinders. PRA concluded hydrogen fires in these areas do not impact any targets.

Nuclear Safety and Radiological Release Performance Criteria:

The use of flammable gas cylinders does not affect nuclear safety as the areas have been analyzed by the FPRA in the current configuration which includes the presence of flammable gas cylinders. Hydrogen fires in the areas do not impact any targets. Therefore, there is no impact on the nuclear safety performance criteria.

The presence of flammable gas cylinders has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the introduction of flammable gas cylinders. The introduction of flammable gas cylinders does not change the radiological release evaluation performed that potentially contaminated water is contained and smoke is monitored. Flammable gas cylinders do not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

These rooms have been analyzed in the current Fire PRA in their current configuration which includes the presence of the flammable gas cylinders. Hydrogen fires in these areas do not impact any targets. The use of these materials has been defined by the limitations of the analytical methods used in the development of the fire PRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The introduction of flammable gas cylinders does not impact fire protection defense-in-depth. Echelon 1 is

maintained by the fire hazard review process required by directives when introducing a new compressed gas cylinder. The introduction of flammable gas cylinders does not affect echelons 2 and 3. The introduction of flammable gas cylinders does not result in compromising automatic or manual fire suppression functions fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the storage of flammable gas cylinders permitted for use in plant operations and systems per design.

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).

Approval Request 4

NFPA 805 Section 3.3.5.1

NFPA 805 Section 3.3.5.1 states:

> *"Wiring above suspended ceiling shall be kept to a minimum. Where installed, electrical wiring shall be listed for plenum use, routed in armored cable, routed in metallic conduit, or routed in cable trays with solid metal top and bottom covers."*

ONS has wiring above suspended ceilings that may not comply with the requirements of this code section.

BTP APCSB 9.5-1 Section 4.A.6 has the requirement "Suspended ceilings and their supports should be of noncombustible construction. Concealed spaces should be devoid of combustibles." ONS's response in February 1982 was that "Suspended ceilings and their supports are non-combustible. Combustibles in concealed spaces are minimal".

The majority of areas at ONS currently with suspended ceilings inside the NFPA 805 defined power block areas are offices, labs, elevator lobbies, corridors, and change rooms. Suspended ceilings were identified in the following areas:

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

These areas are not risk significant with the exception of the Control Rooms. The Control Rooms previously identified cabling above the suspended ceiling. An action to address this condition has been entered into the ONS corrective action program. The corrective action disposition for cabling above the suspended ceilings in the Control Rooms is that the possibility

of a fire is very low due to limited combustible loading, discontinuity of combustibles, and the inherent features of the electrical circuit design. In addition, the ventilation in the Control Rooms is a closed loop system which re-circulates the air where either the existing detection or the Control Room operators who are continuously present in the area would identify the smoke. The station has initiated an engineering change (EC) to install detection above the suspended ceiling area in the control room.

The aforementioned areas are assumed to have wiring above the suspended ceilings; including power and control and video/communication/data. In general, power and control cables at ONS are armored. An inspection above the Control Room ceiling identified the presence of power/control cables. Power and control cables at ONS are IEEE-383 or equivalent. NFPA 805, FAQ 06-0022 identified acceptable electrical cable construction tests. Plenum rated cable is tested to NFPA-262. The FAQ concluded that the NFPA-262 test is equivalent to the IEEE-383 test. Therefore, IEEE cable is inherently equivalent to plenum rated cable and acceptable to be routed above suspended ceilings.

There are also video/communication/data cables which have been field routed above suspended ceilings. Video/communication/data cables are low voltage. The existing Local Area Network (LAN) cables are being removed as the system is being upgraded where feasible for business need purposes with Cat 5E or better plenum rated cables. Existing cables for business, telephone, and Health Physics support (video and instrument) networking may or may not be plenum rated. These low voltage cables are not generally susceptible to shorts which would result in a fire.

The basis for the approval request of this deviation is:

- Power and control cables comply with this section (plenum rated equivalent or armored).
- The wiring above ceilings in offices, lobbies, laboratories, etc., do not pose a hazard:
  - Low voltage is not susceptible to shorts causing a fire.
  - Power and control cables are protected (plenum rated equivalent or armored) per this code section.
  - By eliminating cables with the potential shorts, this eliminates ignition sources and therefore, the jacketing of cable is not relevant.
  - There is no equipment important to nuclear safety in the vicinity of these cables.
  - Beginning in 2006, any new cables installed and the replacement of existing cables as part of upgrades are plenum rated.
- The installation of detection above the Control Room ceilings will promptly identify a fire, thereby, enhancing fire brigade response time.
- Power, control or instrumentation cable installed are constructed similar to or superior to the original cable and meets the requirements of IEEE-383.

Nuclear Safety and Radiological Release Performance Criteria:

The location of wiring above suspended ceilings does not affect nuclear safety. Power and control cables comply with this section. Other wiring, while it may not be in armored cable, in metallic conduit, or plenum rated, is low voltage cable not susceptible to shorts that would result in a fire. Therefore there is no impact on the nuclear safety performance criteria.

The location of cables above suspended ceilings has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the type of cables or locations of suspended ceilings. The location of cables

does not change the radiological release evaluation performed that potentially contaminated water is contained and smoke monitored. The cables do not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

Power and control cables meet these requirements. The use of these materials has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur, thereby, limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The prior introduction of non-listed video/communication/data cables routed above suspended ceilings does not impact fire protection defense-in-depth. Echelon 1 is maintained by the current cable installation procedures documenting the requirements of NFPA 805 Section 3.3.5.1. The introduction of cables above suspended ceilings does not affect echelons 2 and 3. The video/communication/data cables routed above suspended ceilings does not result in compromising automatic fire suppression functions, manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the use of non-listed video/communication/data cables routed above suspended ceilings.

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).

Approval Request 5

NFPA 805 Section 3.3.5.3

NFPA 805 Section 3.3.5.3 states:

> *"Electric cable construction shall comply with a flame propagation test as acceptable to the AHJ."*

Video/communication/data cables installed at ONS may not comply with the requirements of this code section.

The IEEE-383 Standard (one acceptable flame propagation test) was developed after original ONS cables were installed; however, the same cable construction has been used at a later Duke Energy plant and was found to meet the IEEE-383 requirements. The NRC states in the August 11, 1978 SER that the cables used at ONS are acceptable.

The current ONS practice is that all new power, control or instrumentation cable installed will be constructed similar to or superior to the original cable and meeting the requirements of IEEE-383.

The video/communication/data cables are not necessarily tested in accordance with the flame propagation tests outlined in the NFPA 805 FAQ 06-0022 as endorsed by the NRC. These low voltage cables are not generally susceptible to shorts which would result in a fire, therefore self-ignited fires are not a concern. An exposure fire could potentially ignite the cables, although the same fire would result in damage to other cables in the vicinity.

The basis for the approval request of this deviation is:

- Power and control cable installed is constructed similar to or superior to the original cable and meets the requirements of IEEE-383.
- All new power, control or instrumentation cable installed will be constructed similar to or superior to the original cable and meeting the requirements of IEEE-383 or plenum rated.
- Video/communication/data cables are low voltage and not susceptible to shorts and fires.

Nuclear Safety and Radiological Release Performance Criteria:

Power and control cables comply with this section (construction complies with acceptable flame propagation tests). Video/communication/data cables are low-voltage cable not susceptible to shorts that would result in a fire. Therefore there is no impact on the nuclear safety performance criteria.

The flame propagation testing of electrical cable construction has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the flame propagation tests of cables. The construction of cables does not change the radiological release evaluation performed that potentially contaminated water is contained and smoke monitored. Cables do not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

Cables responsible for power and control meet these requirements. The use of these materials have been defined by the limitations of the analytical methods used in the development of the fire PRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur, thereby, limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The introduction of non-listed video/communication/data does not impact fire protection defense-in-depth. Echelon 1 is maintained by the electrical discipline design criteria manual. Cable construction/flame propagation rating does not affect echelons 2 and 3. The video/communication/data cables construction does not result in compromising automatic fire suppression functions, manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the use of video/communication/data cables that are not tested to the flame propagation tests as endorsed by the NRC.

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).

Approval Request 6

NFPA 805 Section 3.3.12(1)

NFPA 805 Section 3.3.12(1) states:

"The oil collection system for each reactor coolant pump shall be capable of collecting lubricating oil from all potential pressurized and nonpressurized leakage sites in each reactor coolant pump oil system."

The ONS oil collection system is designed and was reviewed in accordance with 10 CFR 50 Appendix R, Section III.O to collect leakage from pressurized and nonpressurized leakage sites in the reactor coolant pump oil system. This may not include collection of oil mist as result of pump/motor operation. Oil misting is not leakage due to equipment failure, but inherent occurrence in the operation of large rotating equipment. It is normal for large motors to loose some oil through seals and the oil to potentially become 'atomized' in the ventilation system. This atomized oil mist can then collect on surfaces in the vicinity of the reactor coolant pump as the pump design is not completely sealed to permit airflow for cooling. The oil mist resulting from normal operation will not adversely impact the ability of a plant to achieve and maintain safe shutdown even if ignition occurred. There are redundant reactor coolant pumps to achieve and maintain safe shutdown and there is detection provided in the immediate vicinity of the pumps in order to detect a fire should one occur.

In addition, Generic Letter 86-10, Response to Industry Questions, dated April 24, 1986; Question 6.2 (shown below) discussed oil dripping. The response concluded that there was no concern with oil consumption (which is an oil misting phenomena) but the concern was with an oil fire started from a pressurized leakage point and/or spilled leakage.

QUESTION 6.2
It would appear that a literal reading of Section III.0 regarding the oil collection system for the reactor coolant pump could be met by a combination of seismically designed splash shields and a sump with sufficient capacity to contain the entire lube oil system inventory. If the reactor coolant pump is seismically designed and the nearby piping hot surfaces are protected by seismically designed splash shields such that any spilled lube oil would contact only cold surfaces, does this design concept conform to the requirements of the rule? If the reactor coolant pump, including the oil system, is seismically designed and the nearby hot surfaces of piping are protected by seismically designed splash shields such that any spilled lube oil would contact only cold surfaces, and it could be demonstrated by engineering analysis that sump and splash shields

would be capable of preventing a fire during normal and design basis accident conditions, the safety objective of Section III.0 would be achieved. Such a design concept would have to be evaluated under the exemption process. The justification for the exemption should provide reasonable assurance that oil from all potential pressurized and unpressurized leakage points would be safely collected and drained to the sump. The sump should be shown capable of safely containing all of the anticipated oil leakage.

RESPONSE
The reactor coolant pump, including the oil system, is seismically designed and the nearby hot surfaces of piping are protected by seismically designed splash shields such that any spilled lube oil would contact only cold surfaces, and it could be demonstrated by engineering analysis that sump and splash shields would be capable of preventing a fire during normal and design basis accident conditions, the safety objective of Section III.0 would be achieved. Such a design concept would have to be evaluated under the exemption process. The justification for the exemption should provide reasonable assurance that oil from all potential pressurized and unpressurized leakage points would be safely collected and drained to the sump. The sump should be shown capable of safely containing all of the anticipated oil leakage. The analysis should verify that there are no electric sources of ignition.

Also, the ONS response to BTP 9.5.1in February 1982 stated,

"Reactor Coolant Pumps, which are not required for safe shutdown, have seismically qualified oil collection systems to prevent oil spillage reaching areas which may be above the flash point of the lubricating oil. Upper and lower oil pots have been modified with a shield to catch oil and carry it to a tank to reduce fire potential."

Historically, there have been no fires attributed to oil misting based on normal operation in the industry. Fires have occurred due to oil leakage from equipment failure such as cracked welds on piping or inadequate collection pan design. ONS does not have a history of significant oil loss from the reactor coolant pumps as a result of oil misting or oil leakage that is not contained by the properly designed and installed oil leakage collection system.

The basis for the approval request of this deviation is:

- The oil collection system is designed to collect leakage from pressurized and nonpressurized leakage sites in the reactor coolant pump oil system.
- Oil misted from normal operation is not leakage; it is normal motor oil consumption.
- Oil misted from normal operation does not significantly reduce the oil inventory. The oil historically released as misting does not account for an appreciable heat release rate or accumulation near potential ignition sources or non-insulated reactor coolant piping.
- The reactor coolant pumps use a synthetic oil of higher flash point, approximately 450 degrees Fahrenheit.
- There are redundant reactor coolant pumps and they are not required to achieve or maintain fire safe shutdown.

Nuclear Safety and Radiological Release Performance Criteria:

The oil mist resultant from normal operation will not adversely impact nuclear safety. There are redundant reactor coolant pumps available as necessary. In addition, the reactor coolant

pumps are not required to achieve and maintain fire safe shutdown. Therefore there is no impact on the nuclear safety performance criteria.

The potential for oil mist from the reactor coolant pumps has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials. The entire Reactor Building in which the reactor coolant pumps are located during power operations is an environmentally sealed radiological area. The oil mist does not add additional radiological materials to the area or challenge systems boundaries that contain such.

Safety Margin and Defense-in-Depth:

The oil mist resultant from normal operation will not adversely impact the ability of a plant to achieve and maintain fire safe shutdown even if ignition occurred. There are redundant reactor coolant pumps, however the reactor coolant pumps are not required to achieve and maintain fire safe shutdown. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur, thereby, limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The potential for oil mist from reactor coolant pumps does not impact fire protection defense-in-depth. Echelon 1 is maintained by the oil collection system and reactor coolant pump design. The introduction of a small amount of oil misting does not affect echelons 2 and 3. The potential for oil mist from the reactor coolant pumps does not result in compromising automatic fire suppression functions, manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the potential of oil misting from the reactor coolant pumps due to normal motor consumption not captured by the oil collection system designed for pressurized and non-pressurized leakage and spillage.

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).

Approval Request 7

NFPA 805 Section 3.5.3 (NFPA 20 Section 5.11)

The High Pressure Service Water (HPSW) pumps and the Keowee fire pump are not provided with circulation or automatic relief valves.

NFPA 20 Section 5.11 outlines the requirements for circulation relief valves, and in particular, Section 5.11.1, the requirements for automatic relief valves. The purpose of circulation and automatic relief valves is to provide a flow of water to prevent the pump from overheating when operating with no discharge. The specific sub-section elements in Section 5.11 are related to the settings, configuration, and size of the relief valves.

The two electric driven HPSW pumps are provided with motor cooling lines designed to cool the HPSW pump motor and also provide some flow to prevent the pump from overheating. The HPSW pumps are not standard fire pumps but large industrial pumps and subsequently built to different standards. Due to the size (6,000 gpm) and nature (not only fire protection use) of the HPSW pumps, the pump vendor recommends a minimum flow of 680 gpm but an optimum minimum flow of 1,450 gpm. In addition, the HPSW pumps do not generally operate with no flow. The HPSW pump(s) automatically start upon a low water level signal from the elevated water storage tank. The pump(s) automatically shut off at a designated fill level. Lastly, when the HPSW pump operates in a fire event, the ONS fire brigade response procedure instructs that a deluge system or hydrant be opened if the flow on the system is assessed less than 1,450 gpm in order to maintain greater than the manufacturer's minimum flow.

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

The basis for the approval request of this deviation is:

- The HPSW pumps have procedures in place to ensure there is acceptable flow to prevent overheating therefore circulation relief valves are not necessary.
- The Keowee fire pump has an automatic shut down feature to prevent overheating therefore circulation relief valves are not necessary.

Nuclear Safety and Radiological Release Performance Criteria:

The omission of circulation relief valves on the HPSW and Keowee fire pumps does not affect nuclear safety. The pumps are functional and measures are in place to ensure the pumps do not overheat. Therefore there is no impact on the nuclear safety performance criteria.

The omission of circulation relief valves on the HPSW and Keowee fire pumps has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on the features associated with the fire pumps. The fire pumps provide radiologically clean water to the high pressure service water system and do not cross-tie to contaminated water piping.

Safety Margin and Defense-in-Depth:

The circulation relief valves are provided to prevent the pumps from overheating. Alternative measures are provided to ensure the HPSW and Keowee fire pumps will not overheat. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated

cable, success path remains free of fire damage, recovery actions). The omission of circulation relief valves on the HPSW and Keowee fire pumps does not impact fire protection defense-in-depth. Echelon 1 is maintained by procedures to ensure the HPSW pumps do not overheat by manually opening an excess flow path. The Keowee fire pump auto-stops on loss of flow, thereby preventing overheating when low/no flow. Echelon 2 is maintained by the availability of a redundant HPSW pump. There is not a redundant Keowee fire pump, but the pump is maintained free of overheating by the auto-stop feature. The lack of circulation relief valves on the fire pumps does not affect echelon 3. The pumps are functional and measures are in place to ensure the pumps do not overheat. Therefore this does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the omission of circulation relief valves on the HPSW and Keowee fire pumps.

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).

Approval Request 8

NFPA 805 Sections 3.5.3, 3.5.16, 3.6.1, & 3.6.2 (NFPA 14 Section 7.8.2.1 and NFPA 20 (all))

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 3.5.16 states:

"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."

NFPA 805 Section 3.6.1 states:

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

NFPA 805 Section 3.6.2 states:

> *"A capability shall be provided to ensure an adequate water flow rate and nozzle pressure for all hose stations. This capability includes the provision of hose station pressure reducers where necessary for the safety of plant industrial fire brigade members and off-site fire department personnel."*

NFPA 14 Section 7.8.2.1 states:

> *"Pipe schedule designed standpipe systems shall have piping sized in accordance with the pipe schedule in Table 7.8.2.1 to provide the required waterflow rate at a minimum residual pressure of 100 psi (6.9 bar) at the topmost 2 ½ in. (65 mm) hose connection and 65 psi (4.5 bar) at the topmost 1 ½ in. (40 mm) hose station."*

The Reactor Building hose stations/standpipes are supplied via the Low Pressure Service Water (LPSW) system. The LPSW system is supplied via high volume/low pressure pumps which are not designed or installed to the requirements of NFPA 20. In addition, the pressure at the highest standpipe elevation does not meet the minimum pressure prescribed by NFPA 14. The hose stations in the Reactor Buildings were installed to support the approval of Appendix A to BTP 9.5-1 SER. ONS committed to install hose stations in the Reactor Buildings supplied via the LPSW system. The hose stations are classified as Class II hose stations.

The Reactor Building hose stations are located in the vicinity of the east and west stairways at Elevations 797'-6", 825'-0", and 861'-0". Fire standpipe hose stations are provided in the Unit 1, 2, and 3 Reactor Buildings on an isolated system, which is activated to a charged system during shutdown. Hose stations in the Reactor Buildings are supplied from the LPSW system which was already installed in the buildings. The HPSW system which supplies the balance of the ONS fire protection water system including the yard loop, suppression systems and the remainder of the standpipes is not piped to the Reactor Buildings.

The Reactor Building hose stations are sized in accordance with NFPA 14 Table 7.8.2.1 which requires a minimum nominal pipe diameter of 2-1/2" for piping 50 to 100 feet from the furthest outlet. The highest elevation standpipe is fed by 2-1/2" pipe for approximately 55 feet from the 4" header. The supply for Reactor Building hose stations cannot meet the system demands of pressure (65 psi for 1-1/2" hose stations). In addition, station calculations were performed to identify the pressure at the hose stations for Units 1 and 2 at various flows. The LPSW pumps are designed for large flow (15,000 gpm) at a relatively low pressure. The LPSW pumps are only rated for 100 psi (at the pump). Taking into account friction losses, pipe size changes, and elevation changes, it is not reasonable to expect the required pressure for hose stations would be available using these pumps.

Hydraulic calculations indicate that at a flow of 100 gpm, with 2-1/2" pipe (drawings confirm 2-1/2" piping up to a reducing tee to 1-1/2" to feed the actual hose station) the residual pressure is approximately 21 psi at the highest elevation inside the Reactor Building.

	U1	**U2**	**U1**	**U2**
Elevation	Flow 100 gpm	Flow100 gpm	Flow 75 gpm	Flow 75 gpm
865'-6"	21.2 psi @ I-7; 21.4 psi @ I-3	20.7 psi @ G-11; 20.5 psi @ I-11	22.2 psi @ I-7; 22.3 psi @ I-3	21.9 psi @ G-11; 21.8 psi @ I-11
829'0"	-	-	-	-
801'-6"	-	55.6 psi @ C-6; 56.2psi @ J-6	-	57.3 psi @ C-6; 57.8 psi @ J-6

Note that entries with a '-' were not calculated for flow through 2-1/2" pipe. The largest hydraulic demands are those at the highest elevations. In addition, the Reactor Building hose station designs are generally symmetrical and the same approximate results can be expected in Unit 3 as demonstrated above.

The 1985 NRC Inspection Report 85-34 closed this item identifying that the hose stations in the Reactor Building deviate from NRC and industry guidelines to be adequate and also compensated by the reactor coolant pump oil collection systems and the containment spray system.

Lastly, the fire brigade is equipped with multiple low pressure nozzles staged at various fire brigade equipment locations. The low pressure nozzles are designed to flow larger quantities of water when pressures are less than ideally desired.

The basis for the approval request of this deviation is:

- ONS committed to the NRC to use the LPSW to supply the hose stations/standpipes in the Reactor Buildings. The LPSW pumps were never designed to be able to provide the required pressures for the hose stations in Reactor Building.
- The fire hazards in the Reactor Buildings are minimized and sufficient pressure for hose station operations are provided at the lower elevations where there is a higher concentration of combustibles.
- The ONS fire brigade has low pressure nozzles available and is trained on their use.
- There are six additional CO_2 fire extinguishers staged at each Reactor Building Personnel Hatch.

Nuclear Safety and Radiological Release Performance Criteria:

The LPSW pumps are not designed in accordance with NFPA 20 and the limited pressure from LPSW system to the fire hose stations in the Reactor Buildings does not affect nuclear safety as the Reactor Buildings are not accessed during power operation unless in an emergency. Therefore there is no impact on the nuclear safety performance criteria.

The LPSW pumps not being designed in accordance with NFPA 20 and the limited pressure of fire hose stations in the Reactor Buildings has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials. The entire Reactor Building in which the subject fire hose stations are located is in an environmentally sealed radiological area.

Safety Margin and Defense-in-Depth:

The LPSW pumps not designed in accordance with NFPA 20 and the limited pressure at the standpipes does not negate the presence of the standpipes and the ability to supply at least limited water in a fire event. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the fire PRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated

cable, success path remains free of fire damage, recovery actions). Echelon 2 is maintained by the availability of alternate means for suppression within the Reactor Building. The use of the LPSW system for Reactor Building hose stations does not affect echelons 1 and 3. The LPSW pumps not designed in accordance with NFPA 20 and the limited pressure at the standpipes does not compromise automatic fire detection functions, fire protection for systems and structures, or post-fire safe shutdown capability. Fire extinguishers and hose stations outside the reactor building are provided as the primary means of suppression with hose stations inside the reactor building used as an alternative to attack of a fire.

Conclusion:

NRC approval is requested for the use of the LPSW system to supply the Reactor Building hose stations/standpipes and less than required pressure at the Reactor Building hose stations/standpipes.

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).

Approval Request 9

NFPA 805 Section 3.5.6 and NFPA 20 Section 10.5.2.2.3

NFPA 805 Section 3.5.6 states:

> *"Fire pumps shall be provided with automatic start and manual stop only"*

NFPA 20 Section 10.5.2.2.3 states:

> *"There shall be no means capable of stopping the fire pump motor except those on the fire pump controller."*

Both the HPSW pumps and the Keowee fire pumps have an automatic stop function. In addition, the HPSW pumps can be operated from the Control Room.

The HPSW pumps can be stopped automatically from the level switches in the Elevated Water Storage Tank (EWST), manually in the control room, locally at the switchgear breaker, and locally at the pump. The NRC accepted the HPSW system for fire protection use in the August 11, 1978 SE Report, Section 4.3.1. Due to the nature of the dual purpose of the HPSW system and the water supply supplemented by the EWST, the NRC recognized in the SE Report that the HPSW pumps are operated automatically based on the water level maintained in the EWST (SE Report, Section 4.3.1.2).

The HPSW system has a jockey pump to maintain normal system pressure during service water loads. If the pressure falls below the set point at which the jockey pump can not maintain the HPSW system, the altitude valve, located at the base of the EWST, opens to supply system pressure and flow. As tank level lowers to designated level set points, the HPSW pump(s) automatically start to replenish the tank and provide system pressure. The pump(s) stop based upon a set fill level of the storage tank. In addition, a START/RUN/OFF/BASE/STANDBY switch is provided in the Control Room on Auxiliary Control Board, 1AB3, for the HPSW pumps

(both A and B). This permits the pumps to be manually operated in order to avoid pressure disruptions in the system.

The Keowee fire pump does not have a remote manual stop, but the fire pump stops automatically based on sense of low/no water flow. The Keowee fire pump is provided with a time off-delay relay. This relay will stop the fire pump after a required time delay based on sense of no flow in the system. This delay is calibrated via the Keowee Hydro System (KHS) Fire Protection System – Mulsifyre Pressure Switch and Fire Protection Pump Time Delay Relay Testing inspection procedure. In addition, as part of the KHS Fire Protection Pump and Mulsifyre System Wet Surveillance performance test, the fire pump automatic start and automatic stop features are tested by opening each fire hydrant (exterior automatic wet standpipe), verifying the pump operations while the hydrant is open, fully closing the fire hydrant, then verifying the pump shutdown after the hydrant is closed.

The basis for the approval request of this deviation is:

- The NRC previously accepted the use of the HPSW system for fire protection use.
- The Keowee fire pump has an automatic shutoff on low/no flow. The system is routinely tested to demonstrate operability.
- When the pumps are operating, they are monitored by trained operators who can control the pumps as necessary.

Nuclear Safety and Radiological Release Performance Criteria:

The means of remotely/automatically stopping the fire pumps does not affect nuclear safety. The pumps are available and monitored by trained operators. Therefore there is no impact on the nuclear safety performance criteria.

The means of remotely/automatically stopping the fire pumps has no impact on the radiological release performance criteria. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on these features associated with the fire pumps. The fire pumps provide radiologically clean water to the high pressure service water system and do not cross-tie to contaminated water piping.

Safety Margin and Defense-in-Depth:

The fire pumps operate automatically and are monitored and controlled by trained operators. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The means of remotely/automatically stopping the fire pumps does not impact fire protection defense-in-depth. Echelon 1 is maintained by procedures to ensure pumps are available by taking manual control. Echelon 2 is maintained by the availability of a redundant fire pump. The presence of a remote/auto stop feature on the fire pumps does not affect echelon 3. Means are available to ensure fire pumps are functional during a fire event. This does not result in compromising

automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the means of remote/automatic stop feature on the HPSW and Keowee fire pumps.

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).

Approval Request 10

NFPA 805 Section 3.5.7, 3.5.10, & 3.5.15 (NFPA 24 Sections 5.2.1, 7.1.1, & 13.1)

NFPA 805 Section 3.5.7 states:

"Individual fire pump connections to the yard fire main loop shall be provided and separated with sectionalizing valves between connections."

NFPA 805 Section 3.5.10 states:

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

NFPA 805 Section 3.5.15 states:

"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."

NFPA 24 Section 5.2.1** states:

"Private Fire Service Mains. Pipe smaller than 6 in. (152.4 mm) in diameter shall not be installed as a private service main supplying hydrants."

NFPA 24, Section 7.1.1 states:

"Hydrants shall be of approved type and have not less than a 6 in. (152 mm) diameter connection with the mains."

NFPA 24, Section 13.1** states:

"Private Service Mains. Pipe smaller than 6 in. (152 mm) in diameter shall not be installed as a private service main supplying hydrants. ."

** Note that Sections 5.2.1 and 13.1 are the same requirement.

The Keowee Hydro Generating plant is an extension of the ONS plant as Keowee is used as emergency back-up power. Keowee is part of the property and power block. The Keowee fire protection water system is part of the Keowee service water system and is not part of the ONS HPSW that is used for plant fire protection. The Keowee fire protection water system is not designed in accordance with NFPA 24 and Keowee does not have a yard fire main loop or yard fire hydrants.

Keowee has two outside fire hose standpipes that utilize post type hydrants as their controlling devices. The outside fire hose standpipes (fire hydrants) are equipped with two 2-1/2" hose connections (no other/larger connections).

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

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

The Design Basis Specification for Fire Protection will be revised to state that the fire hydrants are not designed, nor intended to function as fire hydrants but to act as external automatic wet standpipes for fire brigade/fire department response as required.

The basis for the approval request of this deviation is:

- A yard fire loop is not required given the fire protection water required in the Keowee yard.
- The fire hydrants are installed as fire hose standpipes for the fire brigade.

Nuclear Safety and Radiological Release Performance Criteria:

The layout of the fire service main at Keowee does not affect nuclear safety as Keowee is not required for the overall nuclear fire safety at ONS; a fire is not simultaneously postulated at ONS and Keowee. Keowee is the emergency power for ONS. In the event of a loss of offsite power, Keowee provides the power to shutdown ONS. If Keowee were unavailable, ONS would proceed on a controlled shutdown using normal power. Therefore there is no impact on the nuclear safety performance criteria.

The layout of the fire service main at Keowee has no impact on the radiological release performance criteria. There are no radiological concerns at the Keowee Hydro Generating plant location.

Safety Margin and Defense-in-Depth:

The layout of the fire service main at Keowee does not impact fire protection and the fire hydrants are functionally automatic wet standpipes for fire brigade/fire department operations. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire

plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The layout of the fire service main at Keowee does not impact fire protection defense-in-depth. Echelon 2 is maintained by inherent design and objectives of the Keowee fire service main. The Keowee fire service main does not affect echelons 1 and 3. Means are available to ensure water for fire operations is available at Keowee. This does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the omission of a fire yard main loop and that the fire hydrant appurtenances at Keowee are used as outside fire hose standpipes.

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).

Approval Request 11

NFPA 805 Section 3.5.16 (NFPA 24 Section 8.7)

NFPA 805 Section 3.5.16 states:

"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.".

NFPA 24 Section 8.7 states:

"Domestic Service Use Prohibited.

The use of hydrants and hose for purposes other than fire-related services shall be prohibited."

The ONS HPSW system is used for dual purposes including fire protection (suppression systems, hose stations, and fire hydrants) and service water uses including supplying bearing lubrication or cooling water to the Condenser Cooling Water (CCW) Pumps and Motors, the Primary Instrument Air Compressor, and the Leak Rate Test Compressors and backup cooling water to the Turbine Driven Emergency Feedwater Pump Oil Coolers and the High Pressure Injection (HPI) Pump Motors. In addition, the hydrants and/or hose stations may be used for other functions such as wash down, truck/tank filling, etc.

The NRC accepted the HPSW system for fire protection use in the August 11, 1978 SE Report, Section 4.3.1. In addition, at least one of the ONS hydraulic calculations for water-based suppression systems includes a minimum 313 gpm, with a maximum 500 gpm non-fire related service water demand.

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

ONS and Keowee utilize the yard hydrants and standpipes for wash down functions and other non-fire protection related functions which is not permitted by NFPA 24 and not clearly specified in the 1978 SE Report. There are steps to notify the Work Control Center (WCC) Senior Reactor Operator (SRO) when using the HPSW system; in addition the Fire Brigade Response procedure includes a step to make a Public Address (PA) announcement to discontinue use of HPSW for non-essential purposes.

There is significant margin in the HPSW system above that required for the suppression system demands. There are two redundant 6,000 gpm HPSW pumps. The largest suppression system demand is the Unit 2 TB Mezzanine system which requires 2,723 gpm, plus 1,000 gpm fire hose allowance and 318 gpm additional service water for a total demand of 4,041 gpm. However using the maximum 500 gpm non-fire related system flow, there is still approximately 1,800 gpm of margin, with just one pump in operation, in excess of the required HPSW system demands.

The basis for the approval request of this deviation is:

- The HPSW system has excess capacity.
- The Keowee SW system has an automatic valve to cease high SW flow demands in the event of the fire pump start.
- The WCC SRO is notified of use of HPSW system.
- Fire Brigade Response procedure includes a step to make a PA announcement to discontinue use of the HPSW system for non-essential purposes.

Nuclear Safety and Radiological Release Performance Criteria:

The HPSW system has excess capacity to supply the demands of the HPSW system above to the greatest sprinkler system demand. The Keowee SW system has a valve that automatically closes upon actuation of the fire pump. Therefore there is no impact on the nuclear safety performance criteria.

The use of the HPSW system for non-fire protection uses, including the use of hydrants and hose for purposes other than fire has no impact on the radiological release performance criteria. There are no radiological hazards at Keowee. The radiological release review was performed based on the manual fire suppression activities in areas containing or potentially containing radioactive materials and is not dependent on use the HPSW system for non-fire protection uses. The HPSW system is radiologically clean and does not cross-tie to contaminated water piping.

Safety Margin and Defense-in-Depth:

The HPSW system has excess capacity to supply the demands of the HPSW system above the greatest sprinkler system demand. The Keowee service water system has a valve that automatically closes upon actuation of the fire pump. The use of this equipment has been defined by the limitations of the analytical methods used in the development of the FPRA. Therefore, the inherent safety margin and conservatisms in these methods remain unchanged.

The three echelons of defense-in-depth are 1) to prevent fires from starting (combustible/hot work controls), 2) to rapidly detect, control and extinguish fires that do occur thereby limiting damage (fire detection systems, automatic fire suppression, manual fire suppression, pre-fire plans), and 3) to provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed (fire barriers, fire rated cable, success path remains free of fire damage, recovery actions). The use of the HPSW system or the Keowee SW system for non-fire protection uses, including the use of hydrants and hose for purposes other than fire does not impact fire protection defense-in-depth. Echelon 2 is maintained by excess capacity, operational guidance, and automatic equipment functions to maintain sufficient fire fighting water. The use of the HPSW system or the Keowee SW system for non-fire protection uses does not affect echelons 1 and 3. The HPSW pumps have the excess capacity to supply the demands of the HPSW SW system in addition to the greatest sprinkler system demand. The Keowee SW system has a valve that automatically closes upon actuation of the fire pump. This does not result in compromising automatic or manual fire suppression functions, fire protection for systems and structures, or post-fire safe shutdown capability.

Conclusion:

NRC approval is requested for the use of the ONS HPSW and Keowee SW system for purposes other than fire protection water supply.

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 2-19:

Approval Request #2 described in LAR, Attachment L, is requesting approval of the use of temporary portable fuel-fired heaters, which is not permitted by NFPA 805 Section 3.3.1.3.4. Provide further justification for this request by addressing the following:

- Provide justification for why portable electric heaters, as allowed by NFPA 805 Section 3.3.1.3.4, cannot be used at ONS in lieu of fuel-fired heaters.

- The Generic Treatments and the bounding Zone of Influence (ZOI) considered only the use of combustible liquids as a potential source of combustible loading and ZOI and did not construct a ZOI for gas (propane, natural, etc.) or oil-fired fuels. The introduction of materials that potentially have larger ZOIs could affect the FPRA. Explain how the use

of fuels not previously considered in the FPRA will be reflected in the risk review conducted prior to the use of portable fuel-fired heaters.

- It is stated that "if equipment important to nuclear safety is in the area requiring a portable fuel-fired heater, restrictions on the location and an increase frequency of fire watches would be implemented." Describe the risk review process and criteria used to set restrictions on location and establish frequency of fire watches.

- The introduction of additional ignition sources, and ignition sources that may have larger ZOIs than previously considered in the FPRA, clearly could impact the nuclear safety performance criteria. Clarify the conclusion that "there is no impact on the nuclear safety performance criteria" because compensatory measures are taken.

RAI 2-19 RESPONSE:

1. Use of portable electric heaters is not always feasible in the ONS power block buildings. Suitable power sources are not always available for electric heaters, especially those that require 220 V. In addition, during winter months with abnormally cold temperatures, heaters are occasionally needed in large open spaces such as the TB, in key areas for emergency heating, and/or at the personnel hatches. Note that use at personnel hatches will be during an outage. In locations of this size, electric heaters are not as effective as fuel-fired heaters (BTU output). Fuel-fired heaters used in the power block are Kerosene (#1 fuel oil) fired heaters. Portable kerosene-fired heaters may be used for emergency heat in key areas in the event of loss of normal plant heating systems. Loss of normal plant heating systems during an extreme cold event could result in an undesirable unit trip. No other types of fuel are utilized in power block buildings. NSD 316 will be enhanced to clarify that only electric or kerosene portable heaters are to be used in the power block. This item is being tracked in the Corrective Action Program.

2. Placement of portable fuel-fired heaters is outlined in NSD 316 and SD 3.2.14. Portable heaters used in the ONS power block buildings are commercial type portable electric or kerosene heaters. Kerosene (#1 fuel oil) is classified as a Class II combustible liquid. In order to place a portable kerosene-fired heater in an area that has equipment important to nuclear safety the Fire Protection Engineer (FPE) will need to perform an analysis of the potential zone of influence of the particular heater. The analysis will postulate the fuel contents spill and subsequently ignite. Factors such as room geometry will be considered in order to determine spill size and subsequent vertical and horizontal radiant heat distances for the zone of influence. This zone of influence will define the appropriate separation distances required for placement of the heater. The safe shutdown engineer will identify the equipment important to nuclear safety in the area which the heater will be placed in order to ensure appropriate separation distances are maintained. Lastly, the FPE will review other fire protection features in the area such as automatic suppression and detection systems to ensure the placement of the heaters do not impact operation of the system. Both directives will be enhanced to provide additional guidance and clarify placement of kerosene-fired heaters and the requirements for clear space. This item is being tracked in the Corrective Action Program.

3. When portable fuel-fired heaters are used in the plant, the placement and compensatory measures are implemented in accordance with NSD 316. It states the following with regards to temporary heating devices:

"The group that requests the placement of the temporary heating device shall be responsible to assure that impairment firewatches are initiated. Responsibility for control of heaters should be assigned to a specific person. An impairment firewatch shall be conducted at least once every six (6) hours but may be more frequent if determined by the FPE. Impairment Firewatches shall be documented by use of the ICM Form (Appendix A).

Only steam heaters, UL Listed or FM Approved electric heaters, gas (MAPP, LP, or natural) or oil-fired heaters shall be allowed inside buildings. Gasoline and other fuels are not permitted.

Heaters shall be located on noncombustible base or floor and arranged so that they are unlikely to overturn.

Heaters should be located so that there will be a minimum clearance of 6 feet above and 2 1/2 feet on all sides separating the heater from combustible material.

The person responsible for the heater or his designee should control refueling of heaters. Refueling operations shall only be conducted after the unit is sufficiently cool for safe refueling".

In addition, SD 3.2.14 includes a specific flow chart for compensatory measures in placement of temporary portable heaters. The flow chart asks a series of questions regarding the planned location of the temporary heater. These compensatory measures are in addition to the requirements specified in the NSD. The most conservative (frequent) fire watch between the NSD and SD is required to be used. Questions in the SD flow chart include:

- Does the heater meet NSD 316 requirements
- Are there other impairments in the area (if yes - get FPE assistance)
- Occupancy full time (if yes - no compensatory action)
- Area questions (office, yard/warehouse, power production)
- Other maintenance activities
    - If No - then path to ask if fire risk high, suppression available, detection available
    - If Yes - then path to ask if suppression available, detection available, high fire risk

Depending on the activities, risk, and available suppression and/or detection in the area, a series of compensatory measures are outlined. Compensatory measures include fire watches of various durations from set hours to continuous and staging of additional manual suppression sources.

Both NSD 316 and SD 3.2.14 include an FPE consultation/review step. The FPE considers location as part of approval and can modify compensatory actions as necessary, and/or stipulate additional precautions or limitations at the discretion of the FPE.

Additionally, refueling (kerosene-fired heaters) practices will be enhanced with written procedures. Refueling may be done outside of the power block (transport of the heater) or in place. If the heater is refueled in place, then the manufacturer requirements for refueling

shall be followed including ensuring the unit is appropriately cooled, fuel will be transported to the location in no larger than 5-gallon approved safety cans, and a fire extinguisher trained technician will be present with a fire extinguisher during refueling.

Both directives will be enhanced to provide additional guidance and clarify refueling of kerosene-fired heaters. This item is being tracked in the Corrective Action Program.

4. There is no impact on the nuclear safety performance criteria as placement of the heaters will be evaluated as outlined in NSD 316 and SD 3.2.14 and as revised in accordance with Item 2 above. This evaluation will ensure that the ZOI of portable fuel-fired heaters will not have any impact on equipment identified to be important to nuclear safety. Compensatory measures will ensure conditions do not change that would adversely affect nuclear safety.

RAI 2-20:

Approval Request #3 described in LAR, Attachment L, is requesting approval of storage of bulk compressed or cryogenic flammable gas storage inside structures housing system, equipment, or components important to nuclear safety. It is stated that flammable gas cylinders are currently stored in locations that do not impact equipment important to safety, that these locations have been considered in the FPRA and do not impact any targets, and that a fire hazard review is performed prior to installation in any new locations. Describe the fire hazard review process and the criteria used in the decision process, how risk information from the FPRA is used in this review, and how the post-transition plant change evaluation process applies to this situation.

RAI 2-20 RESPONSE:

The fire hazard review process is documented in the Engineering Directives Manual, EDM 601, "Engineering Change Manual". The Engineering Change Manual is used when a proposed change to the station requires design evaluation. The Engineering Change Manual documents the Engineering Change Process which is implemented when changes are required to documents or equipment.

There are multiple screening checklists as part of the process. Introduction of new equipment will be a design change. The engineering review screen for design change checklist includes a Fire Protection screen. This screen will identify changes to fire protection features (barriers, suppression, detection, manual fire fighting), life safety features, combustibles, or ignition sources. If the change impacts the criteria or if the screener is unsure, then the Fire Protection sub-screen and FPE is consulted. The Fire Protection sub-screen includes more detailed questions regarding the change. Lastly, there is the Fire Protection / Safe Shutdown Screening Form which includes additional questions to determine if the proposed plant change has the potential to affect the safe shutdown analysis, fire protection, or FPRA preliminary risk screening. If any questions are answered yes, then a review by the safe shutdown / FPE, and potentially the PRA group is required. In the case of addition of new flammable gas cylinders, this would be captured in the review as an impact on combustible loading which the screening form identifies as a PRA fire magnitude concern.

Risk information is not specifically used in the review, but the screening tools indicate when a PRA review will be performed to determine impact and acceptability of the design change with respect to the FPRA. Changes have been made to this process in anticipation of NFPA 805 to include consideration of FPRA effects from plant modifications as described previously.

Therefore, there is no change required from the current plant approved change evaluation process to the post-transition plant change evaluation process for this specific issue.

RAI 2-21:

Approval Request #5 described in LAR, Attachment L is requesting approval of the use of video/communication/data cables that are not tested to the flame propagation tests specified by the NRC staff. The LAR states that the video/communication/data cables are not necessarily tested in accordance with the flame propagation tests outlined in the Frequently Asked Questions (FAQ) 06-0022 (ADAMS Accession No. ML091240278) as endorsed by the NRC. This implies that ONS will not have any flame spread criteria at all for these types of cables in future installations. Provide further justification for this request. Additionally, clearly explain the cable qualification requirements for future cable installations at ONS. Be sure to justify any deviations from the 10 CFR 50.48(c) requirements for cable installations.

RAI 2-21 RESPONSE:

It is not the intent or implication that ONS will not have any flame spread criteria for video/communication/data cables in future installation.

Acceptable cable construction qualifications will be included in the Power Generation Electrical Discipline Design Criteria Manual. A specific line item will be added that video/communication/data cables shall be plenum rated and/or tested in accordance with IEEE 383-1974, IEEE 1202-1991, CSA 22.2 No. 0.3, NFPA 262, UL 44, UL 83, UL 1581, UL 1666, or UL 1685 (Reference NFPA 805, FAQ 06-0022, Revision 3, ADAMS Accession No. ML090830220).

ONS cable installation procedure, "Cable Installation and Removal" is used for all cable routed in a cable tray or attached to a cable tray which are permanent or temporary installations. It was revised to include requirements from NFPA 805, Section 3.3.5.1. The cable installation procedure currently states in Section 5.6 that "Wiring above suspended ceiling shall be kept to a minimum. Electrical wiring, including video, phone, and communications, installed above a suspended ceiling shall be rated for plenum use, routed in metallic conduit, routed in cable tray with solid metal top and bottom covers, or armored cable."

Therefore, all future installations of cable above ceilings will comply with NFPA 805, Section 3.3.5.3 which requires cable construction to comply with an accepted flame propagation test.

RAI 2-22:

Approval Request #8 described in LAR, Attachment L, related to NFPA 805 Sections 3.5.3, 3.5.16, 3.6.1 and 3.6.2 (NFPA 14, Section 7.8.2.1 and NFPA 20), is requesting approval for use of the LPSW system for the Reactor Building fire hose stations. Specific issues include the lack of an NFPA 20 approved fire pump and insufficient pressure at the hose station. Provide further justification for this request by addressing the following:

- Explain each of the elements of the table provided in the approval request. Clarify, for example, that the discharge pressure at the ONS, Unit 1, RB I-7 hose station provides 22 pounds per square inch (psi) at 75 gallons per minute (gpm).

- Identify the hydraulic points in the system where the pressures given in the approval request are located. Specifically, clarify if the reported ~22 psi is at the angle valve for hose stations on the 865'-6" elevation or at some other point upstream in the system.

- The justification for the use of the system identifies the availability of multiple low-pressure nozzles. The approval request states that "The low pressure nozzles are designed to flow larger quantities of water when pressures are less than ideally desired." Clarify the minimum design specified pressure (or pressure range) for which these low pressure nozzles are qualified and whether they are 'listed" or "approved" to function at ~22 psi.

- One of the bases for the approval request refers to a licensee commitment to the NRC staff to implement the in-place configuration, which was in response to NRC Inspection Report 85-34. NRC inspection reports do not represent NRC staff approval. Clarify whether the licensee considers that the in-place configuration has been previously approved by the NRC and, if so, provide documentation for the prior NRC staff approval as required for a B-1 Table element (i.e., submittal and approval excerpts, etc.).

- A discharge pressure of 22 psi at 75 gpm is very low with regard to the minimum residual pressure of 100 psi per NFPA 14. Furthermore, quick detection and suppression of a fire by the fire brigade is an inherent assumption in the FPRA. Provide an evaluation that demonstrates that the in-place configuration provides adequate manual fire fighting capabilities and that reinforces the conclusion in the LAR that "there is no impact on nuclear safety performance criteria."

- Clarify the statement that "the limited pressure from LPSW system to the fire hose stations in the Reactor Buildings does not affect nuclear safety as the Reactor Buildings are not accessed during power operation unless in an emergency." Explain why a fire in the reactor building would not be an "emergency."

- Provide further justification for the conclusion that there "is no impact on the nuclear safety performance criteria" from the in-place configuration. Include in the response an assessment of the change in the core damage frequency (CDF) and large early release frequency (LERF) and whether the increased risk is acceptable.

RAI 2-22 RESPONSE:

The hose stations in the reactor buildings at ONS were installed as a commitment in the August 11, 1978, SE Report. It was committed to connect the hose stations to the LPSW System. The LPSW system has an inherently low pressure (100 psi at pump discharge); therefore, resulting in low hose station pressures in the reactor buildings.

As part of the NFPA 805 transition, a risk informed performance based evaluation was performed and with that a complete FPRA. The containment FPRA does not credit manual suppression. In addition, the non-power MODES evaluation does not credit fire protection features (active or passive) other than fire barriers. Therefore, the fire hose stations in containment are not credited as part of the performance based risk informed analyses performed.

The fire brigade is the only personnel authorized to use the hose stations at ONS. The fire brigade does not use the hose stations located in containment in the event of a significant fire in

the reactor building. 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. 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, and then proceed with fire fighting operations using a charged water supply from outside of containment. As part of the fire brigade training and standard operating procedures, the fire brigade will not enter the reactor building to fight a fire without a charged hose line. The charged hose line would be connected to the HPSW system hose stations in the AB, or if the equipment hatch is open connected to a yard hydrant. Hose stations used from these 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, a larger diameter hose will be used to overcome friction losses.

The fire brigade would only utilize the reactor building hose stations for a small fire such as a trash barrel, rags, or other small localized fire. This type of fire would be controlled with a limited volume of water and the flows and pressures present in the Reactor Building hose stations would be sufficient for such a fire. 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.

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 was at power, it is likely the majority of these hazards if a fire were to occur would 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, it would likely be of the magnitude that the fire brigade would stage outside of the reactor building and use the AB or yard water supply to attack the fire. Therefore the hose stations are sufficient for the fire scenario in which they would be utilized.

The following are the responses to the specific bulleted questions in the RAI:

1. The table presented in Attachment L in the LAR indicated the pressure available at the various reactor building hose stations (I-7, I-3, etc) with flows of 100 gpm and 75 gpm as these flows were calculated in the supporting calculations. NFPA 14, 1978 edition requirements for Class II hose stations at the angle valve is 65 psi at 100 gpm. The 75 gpm calculated value has no NFPA 14 code basis; therefore it is being removed from the table.

   For clarity, the results of the original calculation at 100 gpm are provided. In addition, values that were originally omitted from the table have been provided using the original calculation methods for those hose stations:

U1		U2	
Elevation	Flowing 100 gpm	Elevation	Flowing 100 gpm
865'-6"	21.2 psi @ I-7 21.4 psi @ I-3	865'-6"	20.7 psi @ G-11 20.5 psi @ I-11
829'0"	38.5 psi @ C-5 ** 38.4 psi @ J-6 **	829'0"	38.2 psi @ C-6 ** 38.9 psi @ J-6 **
801'-6"	50.0 psi @ C-5 ** 49.6 psi @ J-6 **	780'-6" (C-6) 781'-6" (J-6)	55.6 psi @ C-6 56.2psi @ J-6

Note that entries with a ** were not calculated for flow through 2-1/2" pipe in the original calculations. These flows were calculated based on information and assumptions provided in the original calculation. The reactor building hose station designs are generally symmetrical and the same approximate range results can be expected in Unit 3 as demonstrated above.

The aforementioned pressures are located at the angle valve of the hose station. The hose provided at the containment hose stations is Angus 1.5" (red) rubber fire hose in 100 foot lengths. The Angus 1.5" rubber hose has a reported 21 psi friction loss per 100 feet of hose at 100 gpm. At 75 gpm there is a friction loss of approximately 12 psi and at 50 gpm there is an approximate friction loss of 5 psi.

2. The hydraulic points in the system for the pressures provided in the above table are at the angle valve of the hose station at the elevation listed in the above table.

3. The preferred nozzles of the ONS Fire Brigade are Elkhart 4000-023 nozzles which are rated for 50 psi. ONS evaluated various nozzles and determined that this model will deliver the required water flows across the spectrum of pressures from 25 psi to 100+ psi. Although the nozzles may not provide sufficient flow to meet the requirements of NFPA 14 (1978) at the uppermost elevation of the reactor buildings, there is sufficient flow at the mid and lower elevations for the anticipated fire fighting uses of the hose stations.

4. In an SER dated August 11, 1978, Section 4.3.1.4, Duke Energy committed to install hose stations inside containment supplied by the LPSW system. The NRC stated in the SER, "that upon completion of the modification the staff finds the interior hose stations acceptable." Duke Energy considers the SER as approval for the use of the LPSW system and it's applicable system flow characteristics to supply the reactor building hose stations. In addition, Duke Energy considers closure of the inspection report as concurrent acknowledgement of the acceptance of LPSW.

   Duke Energy did not previously seek specific approval based on the SER approval and the inspection report closure. There is no specific documentation that discusses the low pressure of the reactor building hose stations. It is inherent that the hose stations connected to the LPSW system will not be able to meet the pressure requirements as the LPSW system has a design pressure of 100 psi with a total head of 140 feet (140 feet of water = 60 psi) and shutoff head of 202 feet at the pump (202 feet of water = 87 psi).

5. While quick detection and suppression of a fire by the fire brigade is generally an inherent assumption in the FPRA, in the case of a fire in the ONS reactor building, no credit for manual suppression is given in the containment FPRA model, because the reactor buildings are not readily accessible. Therefore, since the FPRA does not credit manual suppression there is no impact on the nuclear safety performance criteria resultant from low pressure in reactor building hose stations. The primary purpose of the hose stations in containment is to act as back-up manual suppression during non-power modes.

6. While a fire is an emergency, immediate access to the reactor building is not possible during operation; therefore, the containment FPRA recognizes limited fire brigade capability. In the event of a fire in containment; the unit would be shutdown and accessed after assessment of reactor building conditions.

7. The containment FPRA does not take credit for manual suppression. Therefore, nuclear safety is not affected as it is assumed that hose stations are not available, or immediately accessible during power operations. Likewise, there is no increased risk or change in delta CDF or LERF as manual suppression is not credited in the containment FPRA . In addition, the reactor building fire risk evaluations state that based on the containment configuration, limited exposed combustibles, and volume, fire propagation can be expected to be slow and limited.

RAI 2-23:

The note added to Table B-3 for fire areas AB, BH12, BH3, CT4, KEO (Keowee), PSW, RB1, and others indicates that "If the East Penetration Room is to become a separate fire area from the Auxiliary Building in the future, then the cork expansion joints in the floor and ceiling should be evaluated as an acceptable fire barrier seal." Clarify the meaning and intent for including this note in the LAR.

RAI 2-23 RESPONSE:

The note in Table B-3 for the aforementioned areas regarding the East Penetration Room is not intended to be part of the LAR. The note was from the supporting fire barriers/fire area separation calculation. Part of the fire barrier/fire area separation calculation evaluated the acceptability of separating the East Penetration Rooms into individual fire areas. The results of the evaluation were not implemented. The East Penetration Rooms are part of the AB Fire Area for transition; therefore, this note has no bearing on the transition effort. The ONS transition database will be revised to remove the note such that it will not appear in future versions of Table B-3.

RAI 2-24:

Table 4-4 of the LAR indicates fire detection will be modified and/or installed in numerous areas of the plant for the performance-based approach utilizing the FPRA. Provide the design, coverage, and code of record (i.e., installation fire code and year) that will be used as the basis for the engineering design change.

RAI 2-24 RESPONSE:

The additional fire detection identified in Table 4-4 will be installed in accordance with NFPA 72, National Fire Alarm and Signaling Code, 2007 edition. Existing fire detection equipment is installed to the NFPA 72, 1987 edition (NFPA 72D and NFPA 72E). The recent code compliance review assessed the fire alarm and detection system to NFPA 72, 2007 edition. Therefore, the 2007 edition will be the code of record going forward.

Additional detection is being installed for indication of a fire to support fire PRA assumptions. Depending on the specific area (hazards, configuration) and the extent of current detection, the new detectors may be added for area protection or for hazard protection. The general number and location/coverage of new detection is outlined in the Engineering Change Request Scope Descriptions. Specific design, location, and coverage details will be developed as part of the formal engineering package for the modification as committed in LAR Attachment S, Table S-1, Item #4.

RAI 3-36:

LAR, Section 4.2.3 and Attachment K, identify that previous NRC staff "Approval of the Safe Shutdown System (SSS) Design" is being transitioned. Provide an evaluation of the continued validity of the bases for the previous NRC approval.

RAI 3-36 RESPONSE:

A review of the SSF SE dated April 28, 1983 to ensure that the bases of the original SE remain valid determined that the following issues require clarification in response to this RAI:

<u>Section 2.0 System Description</u>

Section 2.0 of the SE provides a detailed description of the original design of the SSF subsystems. The pertinent portions of the original design information from the SE have been captured in Chapter 9.6 of the Updated Final Safety Analysis Report (UFSAR) which is maintained as the current design basis of the SSF. As such, a line by line validation of the original design details provided in the SE was not performed. Rather, a review was conducted to validate that UFSAR 9.6 described functionality of the following SSF systems credited for compliance in the B-3 table is capable of meeting the NFPA 805 nuclear safety performance criteria.

<u>SSF Reactor Coolant Makeup (RCM) System</u>

The SSF RCM System is designed to supply borated makeup to the Reactor Coolant System (RCS) to provide Reactor Coolant Pump Seal cooling and RCS inventory. The system is designed to ensure that sufficient borated water is available from the spent fuel pools to allow the SSF to maintain MODE 3 with an average Reactor Coolant temperature ≥ 525°F (the initiating event may cause average RCS temperature to drop below 525°F) for all three units for approximately 72 hours. SSF controlled pressurizer heaters support achieving and maintaining RCS natural circulation flow by offsetting pressurizer heat loss due to ambient heat loss from the pressurizer and pressurizer steam space leakage.

SSF Auxiliary Service Water (ASW) System

The SSF ASW System is designed to cool the RCS during a station blackout and in conjunction with the loss of the normal and Emergency Feedwater System by providing steam generator cooling. The SSF ASW pump is the major component of the system and is sized to provide enough flow to all 3 ONS units to adequately remove decay heat from the RCS and maintain natural circulation in the RCS.

SSF Electrical Power

The SSF Electrical Power System includes 4160VAC, 600VAC, 208VAC, 120VAC, and 125VDC power. This system supplies power necessary to maintain MODE 3 with an average Reactor Coolant temperature ≥ 525°F for the reactors of each unit, in the event of loss of power from all other power systems. It consists of switchgear, load center, motor control centers, panelboards, batteries, battery chargers, inverters, a diesel-electric generator unit, relays, control devices, and interconnecting cable supplying the appropriate loads.

SSF Support Systems

Normal and emergency lighting for the SSF is provided. These lighting units are located to provide adequate levels of lighting in all areas of the structure.

The SSF SW System consists of two subsystems: the HVAC SW System and the Diesel Engine SW System. The HVAC SW System, which operates continuously, contains two pumps and supplies cooling water to the HVAC condensers. The Diesel Engine SW System, which normally operates only when the diesel is operating or when system components are being tested, contains one pump and provides service water to the diesel engine jacket water heat exchangers.

The SSF HVAC system consists of two subsystems, a ventilation system and an air conditioning (AC) system. Both systems are powered by the SSF Power System. The SSF HVAC System supports operation of systems and equipment located in the SSF by maintaining temperature in the SSF within design limits.

In addition, the first paragraph of Section 2.0 states that the "The SSF is to have the capability of maintaining hot shutdown conditions in all three units for approximately three days following a loss of normal AC power."

In 1983, the TS definition of hot shutdown was as follows:

The reactor is in the hot shutdown condition when it is subcritical by at least 1 percent $\Delta k/k$ with an average Reactor Coolant temperature at or greater than 525°F.

During the conversion to Improved Technical Specifications (ITS) in 1998 the Hot Standby and Hot Shutdown operating condition definitions were revised to correlate with their respective ITS MODE 3 and MODE 4 and the current TS definition of hot shutdown is as follows:
The reactor is in the hot shutdown condition when Keff is < 0.99 and the average Reactor Coolant temperature is less than 250°F and greater than 200°F.

The SSF is currently designed to maintain the unit in hot standby with an average Reactor Coolant temperature at or greater than 525°F.

Section 4.7 Fire Protection

In the 2nd paragraph, the SE states that "*no repairs or modifications are required to effect hot shutdown utilizing the dedicated shutdown method*".

ONS subsequently determined that the installation of a portable submersible pump may be required to effect hot shutdown utilizing the dedicated shutdown method. Disposition of the SSF submersible pump is being addressed in the ONS response to RAI 5-73.

Section 4.7.2 Areas Where Dedicated Shutdown is Required

The areas where dedicated shutdown is required will change from those credited in the SE submittal; with the transition to NFPA 805, the AB fire area will be the only fire area utilizing the SSF for dedicated shutdown.

Section 4.7.4 Performance Goals

The 1st paragraph of this section states "performance goals for post fire safe shutdown can be met with the exception of meeting the capability of achieving cold shutdown within 72 hours"; the performance goals of NFPA 805 do not require achieving cold shutdown and therefore this ability is not required.

The 2nd paragraph of this section lists instrumentation originally credited for a post fire dedicated shutdown. ONS credits this same list of instrumentation in its NFPA 805 program with the following exceptions: SSF RCMU pump flow, and SSF ASW pump suction and discharge pressure. These instruments are used to verify pump performance during periodic testing only.

Section 4.7.6 Associated Circuits and Isolation

In paragraph six the SE states that "*DPC has shown that the cable routing of each division (including the SSF cabling) is such that degradation of the redundant shutdown division will not occur, nor will spurious valve actuation occur which might cause an inadvertent depressurization of the primary -system in the event of associated circuit interactions*". Inherent within this statement is the NRC's tacit recognition that the originally accepted design concept for SSF operation was based on a 10 minute capability to transfer control to the SSF and that hot shorts or spurious actuation due to fire within the first 10 minutes of the event are not part of the design basis. ONS has requested clarification of this recognition in Attachment T of the April 14, 2010 LAR.

However, during the fire shutdown analysis documented in calculation OSC-9215; EIR 5044354-004; ONS Appendix R Fire Safe Shutdown Analysis, additional valves were identified that could spuriously open and render the SSF ineffective in mitigating a fire event. These valves, the reactor vessel head vent valves, RC loop high point vent valves and the post accident liquid sampling isolation valves, were classified as variances from the deterministic requirements (VFDRs) of NFPA 805 and were satisfactorily dispositioned in calculation OSC-9327, Oconee NFPA 805 Transition Fire Risk Evaluation - Fire Area AB.

Section 4.7.7 Safe Shutdown Procedures and Manpower

This section has the statement "the manpower to operate the SSF will be exclusive of minimum manpower requirements for the control room and the minimum manpower requirements for the

fire brigade". The staffing to man the SSF is no longer exclusive of the minimum staffing requirements as indicated by the SER but have been incorporated into the levels required in SLC 16.13.1, Minimum Station Staffing Requirements.

RAI 3-37:

LAR, Attachment D, Table F-1, identifies Problem Identification Program (PIP) "Corrective Action #TBD" for numerous variances from deterministic requirement (VFDRs) (i.e., OSC-9313-02, OSC-9313-03, and OSC-9313-07). Clarify the meaning of these entries.

RAI 3-37 RESPONSE:

The VFDRs listed are activities associated with the non power operation program implementation as described within calculation OSC-9313, NFPA 805 Transition Non-Power Fire Area Assessments (Pinch Points Analysis). PIP O-10-2039 will track the implementation of these items. PIP action item (previously called corrective actions) numbers are:

OSC 9313-01	Action Item 31
OSC 9313-02	Action Item 32
OSC 9313-03	Action Item 33
OSC 9313-07	Action Item 34
OSC 9313-09	Action item 35

The #TBD will be replaced with the action item number listed above.

RAI 3-38:

Section 3.5.1.3 of NEI 00-01, Rev. 1, states: "Assume that circuit failure types resulting in spurious operations exist until action has been taken to isolate the given circuit from the fire area, or other actions have been taken to negate the effects of circuit failure that is causing the spurious actuation. The fire is not assumed to eventually clear the circuit fault." With regard to this criterion, Section 3.5.1.3 [Duration of Circuit Failures] of Table B-2, Nuclear Safety Capability Assessment Methodology Review states that the ONS methodology "Does not Align, but has previous approval." The basis for this determination is stated to be "Previous design considerations did not assume spurious actuations or hot shorts due to a fire for the first 10 minutes of the event. This was stated in the referenced safety evaluation report for the standby shutdown facility (SSF). Other spurious operations beyond this assumption were postulated to occur until mitigating actions are taken."

Safety evaluations issued by the NRC staff state that the SSF is expected to be manned and communications established with the main control room within 10 minutes after confirmation of an active fire. Specifically, upon confirmation of an active fire in a fire area where the SSF is credited for safe shutdown and where fire damage could potentially impact SSF operation (defined as an "SSF Risk Area"), an operator would be promptly dispatched to the SSF. As described in the ONS response to RAI 3-1, the shutdown procedure would not be entered until it is verified that an "active fire" in an SSF Risk Area has propagated into a "Challenging Active Fire" (defined as a fire that is burning cables which have the potential to affect additional equipment). Although the decision to shutdown the facility may not occur for sometime after discovery and confirmation of a fire in an SSF Risk Area, transfer of control to the SSF would have already occurred. This is consistent with the ONS response to RAI 3-28 which states: "Upon confirmation of an active fire in those areas of the plant where the Standby Shutdown

Facility is the credited method for safe shutdown and where a fire could adversely affect the safe shutdown capability of the SSF (i.e., SSF transfer control cables) an operator will be dispatched to transfer control to the SSF. This transfer will be completed within 10 minutes of confirmation of the fire."

It does not appear to be correct to state that the NRC staff approved a methodology that does not assume the occurrence of cable or equipment fire damage such as spurious actuations or hot shorts for the first 10 minutes of any fire event. The "10-minutes free of fire damage assumption" appears to be directed at achieving reasonable assurance that operation of the SSF will not be impacted by fire in areas where it is relied on to accomplish required shutdown functions rather than a generic assumption regarding the timing of circuit failures. Provide additional justification which clearly demonstrates NRC staff approval of the assumption that ONS does not need to consider spurious actuations or hot shorts due to a fire for the first 10 minutes of any fire event or, alternatively, clarify that the 10 minute assumption is only valid for SSF Risk Areas and has no relationship to the fires in areas where shutdown will be accomplished from the control room.

RAI 3-38 RESPONSE:

Table B-2, Nuclear Safety Capability Assessment Methodology Review, Section 3.5.1.3, 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."

RAI 3-39:

Although the alignment statement may remain unchanged, the alignment bases of several sections have been modified in the licensee's letter dated April 14, 2010, to the extent that it does not address NEI 00-01 criteria as clearly as the submittal dated October 31, 2008. Specifically, the alignment basis of the submittal dated October 31, 2008 typically includes language which addresses each sub-criterion of a specific NEI 00-01 section. However, in several instances the information submitted by letter dated April 14, 2010, does not provide the level of detail necessary to readily conclude that each sub-criterion of the NEI 00-01 section has been satisfied. For example, NEI 00-01 Section 3.1.1.3 states that feedwater flow should be controlled. The submittal dated October 31, 2008, clearly states that feedwater flow rates are controlled. However, such clarity is not evident in Section 3.1.1.3 of the LAR information sent by letter dated April 14, 2010, and no justification is provided for a lack of alignment. As another example, NEI 00-01, Section 3.1.1.4, provides the basis for the use of an alternative shutdown capability for a specific fire area. The submittal dated October 31, 2008, identified the specific fire areas where shutdown is accomplished from the SSF. However, this information is not included in the submittal dated April 14, 2010. Identify all Alignment Bases in the submittal dated April 14, 2010, that differ from the Alignment Bases provided in the submittal dated

October 31, 2008, and, where necessary, provide additional information as necessary to readily conclude that each sub-criterion of the NEI 00-01 section has been satisfied.

RAI 3-39 RESPONSE:

The B-2 Table in the LAR dated April 14, 2010, was revised to reflect changes in the methodology and terminology for demonstrating the ability to maintain the plant safe and stable post-fire, for clarification of details justifying the alignment bases, and to correct typographical or grammatical errors. A review of all the differences between the two tables was performed; the changes were grouped as follows:

- Group 1: Changes in terminology resulting from ensuring the ability to maintain safe and stable conditions associated with NFPA 805 instead of meeting the criteria of Appendix R.
- Group 2: Clarification of details needed to document alignment with NEI guidance, such as changes resulting from ensuring the ability to maintain safe and stable conditions, changes resulting from crediting the PSW modification, or changes to better clarify details documenting alignment.
- Group 3: Editorial changes to correct typographical or grammatical errors.
- Group 4: Changes which reduced the level of detail because it was deemed not necessary to document alignment with the NEI guidance. Note that detailed system and equipment information is contained within the Nuclear Safety Compliance Assessment (NSCA).
- Group 5: The level of detail was reduced below the level necessary to easily document alignment with the NEI guidance; these alignment statements will be revised in a subsequent revision to the B-2 Table calculation to include additional clarifications to document alignment with the NEI guidance. The B-2 revision will be tracked in the corrective action program.

The following table identifies the NEI 00-01 sections in which the Alignment Bases differ between the LAR submittals.

NEI Section Alignment Bases Grouping

Group 1	Group 2	Group 3	Group 4	Group 5
3.3.1.1	3	3.1.3.4	3.1.2.6.2	3.1.1.3
3.3.1.3	3.1 [B]	3.2.1.7	3.2.2.1	3.1.1.4
3.3.1.4	3.1 [C]	3.3.1.5	3.2.2.2	3.1.1.7
3.3.3.3	3.1.1.6		3.2.2.3	3.1.1.11
3.3.3.4	3.1.1.9		3.3.3.2	3.1.2.1
3.3.3.5	3.1.1.10			3.1.2.2
3.4.2.1	3.2.1.6			3.1.2.3
3.4.2.5	3.2.2.4			3.1.2.4
	3.3.1.2			3.1.2.5
	3.3.1.6			3.1.2.6.1
	3.3.3.1			3.2.1.1
	3.4.2.3			3.2.1.2
	3.5.1.1			3.5.1.3
	3.5.1.2			3.5.1.5 [B]
	3.5.2.3 [A]			3.5.2.1
	3.5.2.3 [B]			
	3.5.2.4			
	3.5.2.5			
	3.4.1.2			
	3.4.1.3			
	3.4.1.4			
	3.4.1.5			
	3.4.1.6			
	3.4.1.7			
	3.4.2.2			
	3.4.2.4			

Duke Energy decided during the review that additional details will be added back to the alignment basis for the following referenced NEI guidance statements; for sections addressed in other RAIs, no revised statement is provided in the RAI and the other RAI is referenced:

- 3.1.1.3: The Alignment basis will be revised to reflect that: For a Control Room shutdown, ONS credits use of makeup/charging, pressurizer heaters, and pressurizer safety relief valves to control Reactor Coolant pressure at hot standby conditions. For SSF shutdown, ONS credits use of pressurizer heaters, pressurizer safety relief valves, and makeup/charging to control Reactor Coolant pressure at hot standby conditions. Decay heat removal during hot standby conditions is provided by natural circulation of the RCS through the steam generators utilizing the PSW Pump for feedwater for a Control Room shutdown and SSF ASW for a SSF shutdown. Feedwater flow rates and steam release rates are controlled by the throttling of feedwater control valves.
- 3.1.1.4: The first sentence of the Alignment basis will be revised to reflect that ONS utilizes a dedicated SSF as a dedicated shutdown capability for Fire Area AB.
- 3.1.1.7: See RAI 3-40 response.
- 3.1.1.11: The alignment basis will be rewritten to reflect that: The effects of fire on the ability to achieve and maintain safe shutdown of all three units have been evaluated for all fire areas in the NSCA. The analysis has demonstrated this ability for each unit separately and for all three units collectively where required.

- 3.1.2.1: The alignment basis will be rewritten to reflect that: For SSF shutdown, borated water from the Spent Fuel Pool using the RCM pump is used to maintain shutdown margin. For a shutdown from the Control Room, borated water from the Borated Water Storage Tank using normal HPI is used to maintain shutdown margin.
- 3.1.2.2: The alignment basis will be rewritten to reflect that: For a Control Room shutdown, ONS credits use of makeup/charging, pressurizer heaters, and pressurizer safety relief valves to control Reactor Coolant pressure at hot standby conditions. For SSF shutdown, ONS credits use of pressurizer heaters, pressurizer safety relief valves, and makeup/charging to control Reactor Coolant pressure at hot standby conditions. Decay heat removal during hot standby conditions is provided by natural circulation of the RCS through the steam generators.
- 3.1.2.3: The alignment basis will be rewritten to reflect that: For a Control Room shutdown, inventory control is accomplished by using an HPI pump for makeup and Reactor Coolant Pump seal injection and Reactor Coolant letdown through Reactor Coolant head vent valves. For SSF shutdown, inventory control is accomplished by using the SSF RCM Pump for Reactor Coolant Pump seal injection and Reactor Coolant letdown back to the spent fuel pool.
- 3.1.2.4: The alignment basis will be rewritten to reflect that: For an SSF shutdown, decay heat removal during hot standby conditions is provided by natural circulation of the RCS through both steam generators utilizing the SSF ASW pump for secondary side make up. For a Control Room shutdown, decay heat removal during hot standby conditions is provided by natural circulation of the RCS through both steam generators utilizing the PSW Pump for secondary side make up. NFPA 805 does not have any explicit requirements to achieve cold shutdown, therefore the NFPA 805 criteria for the Nuclear Safety Performance Goals have been applied to ensure the fuel is maintained in a safe and stable condition.
- 3.1.2.5: The alignment basis will be rewritten to reflect that: Process monitoring instrumentation has been analyzed on a functional level. For SSF shutdown, the following system process variables are available: Pressurizer pressure and level, RCS pressure and temperature, SSF RCM flow and suction/discharge pressure, SSF ASW flow, and steam generator level. An exemption for boron sampling in lieu of neutron source range monitoring instrumentation has been granted for the SSF, therefore neutron source range instrumentation has not been provided or analyzed for the SSF. Steam generator pressure instruments are also not provided in the SSF. An exemption for steam generator pressure indication has been granted for the SSF. For a Control Room shutdown, the following system process variables are available: PSW flow to Steam Generators; Steam Generator, Pressurizer, and Borated Water Storage Tank level; HPI header and seal injection flow; Main Steam Header and Reactor Coolant pressure; Reactor Coolant Temperature; Source Range Flux; Letdown Storage Tank level and temperature; Reactor Building Pressure, Pressurizer level, Reactor Coolant pressure, Steam Generator level, and/or Reactor Coolant temperature.
- 3.1.2.6.1: The alignment basis will be rewritten to reflect that: For a Control Room shutdown, emergency onsite power is provided from the KHS through the underground power path to the PSW 4KV SWGR which provides power for the required AC and DC electrical distribution equipment. Batteries and battery chargers are analyzed to be available when needed. For SSF shutdown, electrical power is provided by the SSF Diesel generator, which provides power for the required AC and DC electrical distribution equipment. Batteries and battery chargers are analyzed to be available when needed.

- 3.2.1.1: The alignment basis will be rewritten to reflect that: The dividing of equipment into 2 categories approach was used at ONS; 'primary' components were identified and added to the Safe Shutdown Equipment List (SSEL), 'secondary' components (referred to as subcomponents) were grouped together with the primary components. Although some subcomponents were not individually identified (i.e., relays, fuses, hand switches, etc), the cables which are connected to the subcomponents were identified and assigned to the primary components. In some instances, components were not captured by the cable selection process but were captured within the cascading interlocks analysis as pseudo-components. The affects of fire on these pseudo components was evaluated where appropriate.
- 3.2.1.2: The alignment basis will be rewritten to reflect that: Safe shutdown analysis assumptions for post-fire integrity of mechanical components to function as pressure boundaries are essentially identical. No damage to packing or gaskets is assumed. The analysis assumes no damage to manual valves. The ability to manually operate valves post fire is evaluated in the manual action feasibility calculation, as necessary.
- 3.5.1.3: See RAI 3-38 response.
- 3.5.1.5 [B]: See RAI 3-41 response.
- 3.5.2.1: See RAI 3-48 response.

RAI 3-40:

The submittal dated October 31, 2008, Section 3.1.1.7, states that offsite power has not been analyzed or demonstrated to be free of fire damage for redundant shutdown. The submittal dated April 14, 2010, Alignment Basis indicates that the availability of offsite power has, in fact, been analyzed. Provide additional information (including fire areas where offsite power is credited) to clarify this apparent difference in alignment.

RAI 3-40 RESPONSE:

The alignment basis for NEI 00-01 Ref 3.1.1.7 [Offsite Power] of Attachment B, NEI 04-02 Table B-2, in the ONS Units 1, 2 and 3 NFPA 805 Transition Report, dated April 14, 2010, states:

> *Oconee relies upon Keowee hydro station to provide emergency onsite power. The cascading power supply analysis determines fire impact to offsite power sources and is utilized in the analysis of fire areas for safe shutdown functions to determine availability of all credited power sources. The adverse consequences of Offsite power being available is considered in the NSCA.*

The credited power supplies as depicted in the reference provided, EIR 51-5044354-004, Oconee Appendix R Fire Safe Shutdown Analysis, Rev. 4, 2/26/2009, Section 11.4, are the KHS and the SSF Diesel. Neither the KHS nor the SSF Diesel requires Offsite Power. To clarify, the Safety Related KHS is on site, but remote from plant structures and is not considered offsite power.

For each fire area analyzed in the NSCA, the KHS or the SSF diesel is available to ensure the fire affected unit is maintained in a safe and stable condition.

To ensure clarity, an action item in PIP O-10-2039 has been created to revise calculation OSC-9291, NFPA 805 Transition B-2 Table, Section 3.1.1.7 to reword the alignment basis to clearly state that offsite power is not credited for the deterministic analysis and therefore not analyzed for its availability in the deterministic analysis. The alignment statement will also be revised to ensure the proper relationship with the alignment basis.

RAI 3-41:

In the submittal dated October 31, 2008, the licensee stated that it meets NEI 00-01, [Section] 3.5.1.5[B], with the exception of inter-cable shorts between armored cables. However, from the alignment basis statement, it is not clear how the licensee aligns with the NEI 00-01 criteria with respect to concurrent faults in multi-conductor cables. The alignment basis for Section 3.5.1.5[B] states that the three types of circuit failures identified in NEI 00-01 were considered to occur on each conductor of each safe shutdown cable. This statement could be interpreted to mean that for each cable under evaluation only one of the three types of circuit failures was postulated to occur at any given time (e.g., considering "A" short to ground on conductor #1; then "A" open circuit on conductor #1; then "A" hot short on conductor #1...and repeating this process for each conductor). Inadequacies with this "one-at-a-time" evaluation technique have been identified at other facilities. For example, the application of this approach at a boiling-water reactor resulted in a failure to consider the potential for fire to cause all 16 safety relief valves to spuriously open because it would have required the occurrence of two concurrent circuit failures (i.e., concurrent shorts between conductors of two twisted pairs) within a single multi-conductor instrument cable. In addition, to not being consistent with the results of the cable fire tests sponsored by industry and NRC staff, the "one-at-a-time" methodology does not satisfy the following statements of Section 3.5.1.5 [B] of NEI 00-01:

(a) more than one conductor-to-conductor short will occur in a given cable and

(b) for any individual multiconductor cable (thermoset or thermoplastic), any and all potential spurious actuations that may result from intra-cable shorting, including any possible combination of conductors within the cable, may be postulated to occur concurrently regardless of number.

Other sections of the B-2 table, such as Section 3.5.2.3 A and B, state that multiple failures were considered. However, the B-2 Table does not specifically state that multiple faults were considered to occur concurrently. Confirm that the evaluation of potential fire damage to multi-conductor cables did not impose an a priori limit on the number of circuit failures that may occur concurrently and multiple concurrent circuit faults, involving all possible combination of conductors within the cable, were assumed to occur concurrently regardless of number.

RAI 3-41 RESPONSE:

While performing the Fire Area Analysis for ONS, multiple fire induced failures and multiple spurious actuations were considered to occur concurrently in accordance with the guidance provided in NEI 00-01, Section 3.5.1.5[B]. This methodology is applied at the safe shutdown component level regardless of how many components that may share a common multiconductor cable. Based on this methodology, ONS considers any and all potential spurious actuations that may result from intra-cable shorting, which may occur concurrently regardless of number.

The verbiage in the B-2 Table, Section 3.5.1.5 [B, Cable Failure Modes] for Parts A and B are being revised to add additional clarification. An action item in PIP O-10-2039 has been initiated to revise the B-2 Table with this clarification.

RAI 3-43:

Section 3.4.2.3 (Determine Safe Shutdown Equipment Impacts) of NEI 00-01 states: "Using the circuit analysis and evaluation criteria contained in Section 3.5 of this document, determine the equipment that can impact safe shutdown and that can potentially be impacted by a fire in the fire area, and what those possible impacts are." Thus, alignment to this section would require the analyst to:

(a) identify equipment that can impact safe shutdown

(b) from this list of equipment, identify equipment that can potentially be impacted by a fire in the fire area under evaluation

(c) determine what those impacts are

The Alignment Basis provided in Section 3.4.2.3 of the submittal dated April 14, 2010, Table B-2 states, in part: "based on the safe shutdown cables and components present in the fire area of concern... All postulated safe shutdown cable and component failures were identified and a resolution provided at the cable or component level for the credited train." From this statement, it appears that the licensee is in full alignment with Section 3.4.2.3 of NEI 00-01. However, the Alignment Basis Statement states that ONS "Aligns With Intent" rather than "Aligns." Provide additional information (with specific examples) to clarify the intent of the stated alignment.

RAI 3-43 RESPONSE:

The "Aligns With Intent" compliance statement is based upon two departures from NEI 00-01 methodology that provide the same net results. The first departure from the criteria in section 3.5 of NEI 00-01 is based on the use of armor jacketed cables which precludes the need to analyze for intercable hot shorts. The armored jacket precludes this interaction from occurring. The second departure is the order of analytical operations stated in NEI 00-01 above. Duke Energy performed the order of analytical operations as follows:

(f) Identified a comprehensive SSEL.

(g) Determined impacts on all safe shutdown equipment at the cable or component level in the fire area under evaluation.

(h) Assessed impacts on safe shutdown equipment to determine least impacted train or method in the fire area under consideration. This defines the credited train/method and the associated credited SSEL.

(i) Identified equipment in the credited train/method that can be potentially impacted by the fire in the fire area under evaluation.

(j) Provided assessment of impacted equipment in the credited train/method against deterministic criteria to identify VFDRs.

The methodology departures, while providing essentially identical results, are slightly different than that specified in NEI 00-01; hence, the basis of the "Aligns With Intent" compliance statement.

Clarification of the "Aligns With Intent" compliance statement to describe the information above is tracked through the corrective action program.

RAI 3-44:

Section 3.4.2.3 of the submittal dated April 14, 2010, states: "The Safe Shutdown Equipment List (SSEL) and logics were developed based on potential spurious operations and other plant impacts by their selection from a functional basis." The meaning and intent of this statement is unclear. Provide additional clarifying information.

RAI 3-44 RESPONSE:

The SSEL contains components that can potentially spuriously operate and impact the safe shutdown function(s). The logics were developed to allow representation of the safe shutdown success paths at a functional level rather than system level and to show potential flow blockage and diversion paths that could impact the safe shutdown function independent of the system, the component, or equipment it's associated with. For example: The Decay Heat Removal function includes components from diverse systems such as Main Feedwater, Emergency Feedwater, Main Steam, and PSW. The logics for Decay Heat Removal show all of the available success paths including any potential flow blockage or diversion paths to achieve the safe shutdown function, independent of the system these components may be located in. In this way, a comprehensive SSEL is developed, cables are selected and the impact of a fire in a given area can be assessed.

RAI 3-45:

Section 3.4.2.3 of NEI 00-01 states that the evaluation should determine equipment that can impact safe shutdown and that can potentially be impacted by a fire in each fire area. The corresponding sections of both the submittal dated April 14, 2010, and the submittal dated October 31, 2008, state that ONS "Aligns With Intent" of this guidance. The submittal dated April 14, 2010, Table B-2, Alignment Basis for NEI 00-01 Section 3.4.2.3, is identical to the Alignment Basis stated in the submittal dated October 31, 2008, Table B-2, except the submittal dated April 14, 2010, does not include the following statement: " ... it is not possible to evaluate all the possible combinations of multiple spurious actions that could occur as a result of the fire and the overall affect of these combinations on safe shutdown." Provide the technical basis and rationale for deleting this statement from the Alignment Basis provided in the submittal dated October 31, 2008.

RAI 3-45 RESPONSE:

It should also be noted that the sentence prior to the one referenced was revised for the submittal dated April 14, 2010 to add the words, "*...for the credited train.*" The full sentence reads, "*All postulated safe shutdown cable and component failures were identified and a resolution provided at the cable or component level for the credited train.*" The methodology used by Duke Energy for safe shutdown logic development at a functional level assures no spurious operation interactions are missed and precludes the need to analyze all trains and components in a fire area for impact. It was later determined that the methodology utilized for evaluation of multiple spurious operations was, in fact, 'bounding' and that while all possible combinations could not be determined with certainty, the impacted equipment and overall effect could be evaluated in a conservative manner, hence the changes in the alignment basis statement.

RAI 3-46:

The submittal dated April 14, 2010, Table B-2, Section 2.4.2.2, appears to repeat Sections 3.3.1.7 and 3.3.3.3 at the end of this section. Clarify the reason for these repeats.

RAI 3-46 RESPONSE:

This is an apparent artifact of the software tool utilized to generate the B-2 table reports. The table will be corrected to eliminate the duplications. An action item has been generated in PIP O-10-2039 to update the report and calculation. This will be done prior to implementation.

RAI 3-48:

Because of the extremely high crest voltages that may occur if a current transformer (CT) secondary is open circuited when the primary circuit is energized, the secondary of a CT must either be short-circuited or connected to a burden (i.e., load). This concern is reiterated in product literature published by CT manufacturers, which typically include a CAUTION statement emphasizing that a CT secondary should never be open-circuited while the primary is energized.

The need to consider the potential for fire to cause an open secondary CT is described in Section 2.4.2 of NFPA 805, which requires consideration of fire-induced open circuit failure modes and specifies that circuits which share a common enclosure with circuits required to achieve nuclear safety performance criteria, be evaluated to ensure that such electrical faults will not cause the fire to extend beyond the immediate (i.e., initial) fire area. As discussed in Section B.3.4.2, the evaluation of common enclosure issues should include consideration of the following:

- *Current transformers that are constructed such that an open secondary circuit could cause ignition of the transformer should be considered.*

- *Current transformers that are susceptible to ignition due to open secondary windings and have secondary circuits extending outside the fire area that are not isolated by transducers should have their circuits included in the nuclear safety assessment.*

In addition, Section 3.5.2.1, Circuit Failures Due to an Open Circuit of NEI 00-01, Rev. 1, also describes the need to consider the potential for open secondary CT circuits and states: "Open circuit on a high voltage (e.g., 4.16 kV) ammeter CT circuit may result in secondary damage." The Alignment Basis provided in Table B-2 of the submittal dated April 14, 2010, states:

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

2. CT's are designed to maintain integrity upon a secondary open circuit and they are contained within metal-clad switchgear which should contain any damage should there be an energetic failure.

3. ONS has assumed that an open circuit on the secondary windings of a CT will not cause a fire.

4. This is an open item under section 3.3.3.3 and is to be considered within the scope of the Circuit Coordination Study update.

With regard to Item 1 above:

This statement is clearly not consistent with current NRC staff positions, NFPA 805 or industry guidance provided in NEI 00-01. Section 2.4.2.2.1 of NFPA 805 requires consideration of fire-induced open circuits. Section 2.4.2.2.2 of NFPA 805 states that circuits that share a common enclosure with circuits required to achieve nuclear safety performance criteria shall be evaluated for their impact on the ability to achieve nuclear safety performance criteria, and Section B.3.3 describes current transformer open secondary concern as a special type of common enclosure issue and states that an opening in the secondary circuit causes excessively high voltages in the current transformer secondary circuit which can result in an ignition of the transformer materials.

Confirm that the referenced memorandum is part of the current ONS fire protection license basis or provide a technical justification to support your contention that consideration of fire-induced open circuits in CT secondary circuits is an "overly conservative" position and "lacked substantiation".

With regard to Item 2 above:

(a) As discussed in NFPA 805, one concern a fire-induced open secondary CT circuit is that excessively high voltages may be produced which could ignite the transformer materials. Thereby, extend the effects of a fire outside of the immediate fire area. For example, a control room fire that caused an open ammeter circuit could result in a secondary fire in a switchgear relied on to meet the nuclear safety objectives. Provide information such as manufacture's data which substantiates the claim that CT's in use at ONS are designed to maintain their "integrity" upon a secondary open circuit.

(b) The metal-clad switchgear may or may not contain the fire damage resulting from an open circuited CT. Provide information to confirm the validity of the licensee's stated assumption that the metal-clad switchgear should contain any damage should there be an energetic CT failure.

(c) In addition, an open circuit in a CT secondary winding may impact the operation of metering and relaying circuits and, thereby, cause equipment to maloperate. Based on the assumptions described in Section 3.5.2.1 of Table B-2, it appears that such failures may not have been considered. Provide information to confirm that metering and relaying circuits have been analyzed for the potentially adverse impact of fire damage, including, but not limited to, open CT secondary windings.

With regard to Item 3 above:

Provide a technical basis to support the validity of the licensee's statement that an open circuit on the secondary windings of a CT will not cause a fire.

With regard to Item 4 above:

(a) As discussed above, the potential impact of fire-induced open secondary CT circuits should be considered within the scope of the Common Enclosure analysis. Within the context of fire protection, a circuit coordination study is performed to demonstrate that fire-induced faults on non-essential cables (i.e., cables of equipment that is not required to assure the nuclear safety capability) will not impact the operation of required power supplies and is typically limited to ensuring adequate selectivity between feeder and load breakers of a selected bus. The concern for fire to cause open circuits in current transformer secondary windings is defined in Appendix B of NFPA 805 as a Common Enclosure issue and not a circuit coordination issue. Provide information to support the licensee's statement that fire-induced open CT secondary circuits will be bounded by the currently ongoing Circuit Coordination Study update.

(b) Based on the assumptions contained in the ONS Alignment Basis it is not clear how ONS intends to address the concern for open circuit CT secondary circuits. Provide a detailed description of the methodology that will be used to identify and resolve potential CT secondary circuits of concern.

In addition to the above, the Alignment Basis provided for Section 3.5.2.1 of the B-2 Table is not consistent with Section 3.2.4.3 and Section 3.2.4.3.1 of the licensee's fire protection Design Basis Document (DBD), "NRC Acceptance Criteria for Associated Circuits by Common Enclosure," Revision 8, which states:

- Current transformers (CTs) may induce secondary fires through the fire-induced opening of circuitry associated with the secondary side windings of the CT. Where such circuitry exits in a fire area or provides a common enclosure concern within a fire area, the impact of such secondary fires should be properly considered.

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

Provide a justification for not meeting the criteria specified in the DBD and describe the interim compensatory measures that have been put in place until conformance with the DBD and NFPA 805 is achieved.

RAI 3-48 RESPONSE:

Item 1:

The following statement is being removed from the B-2 Table, Section 3.5.2.1:

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

ONS does not contain the referenced memorandum in the current ONS fire protection licensing basis. This information in the B-2 Table and NSCA will be replaced with the methodology contained in the ONS Appendix R design basis document.

Item 2:

(a) Based on the removal of the referenced memorandum from Item one and the inclusion of the ONS Appendix R design basis document information, the assumption associated with the secondary CT circuits is being removed. This will ensure that ONS has properly evaluated the effects of an open secondary CT circuit for associated circuits as prescribed in NFPA 805, Section 2.4.2 and guided in NEI 00-01, Section 3.5.2.1.

(b) Based on the removal of the referenced memorandum from Item one and the inclusion of the ONS Appendix R design basis document information, the assumption associated with the metal-clad switchgear is being removed. This will ensure that ONS has properly evaluated the effects of an open secondary CT circuit for associated circuits as prescribed in NFPA 805, Section 2.4.2 and guided in NEI 00-01, Section 3.5.2.1.

(c) The following information is being provided to demonstrate that metering and relaying circuits have been analyzed for the potentially adverse impact of fire damage, including, but not limited to open secondary CT windings:

- Cables associated to secondary CT circuits are included in the cable selection of the associated buses as part of the cable selection performed for the NSCA, PRA, and Non Power Operation (NPO) required equipment.
- The cable routing for these circuits are documented in ARTRAK.
- Fire zone/area, the raceways are located in, are documented in ARTRAK.
- The cable impacts for secondary CT circuits are evaluated on a Fire Area basis contained in ARTRAK.

The information above was utilized to determine if the cable faults of the secondary CT circuits can prevent the operation or cause the maloperation of redundant systems used to achieve and maintain hot shutdown as required by NFPA 805 and guidance provided in NEI 00-01, Revision 1.

The results from the analysis of the secondary CT circuits demonstrate that the redundant systems (i.e. SSF or PSW System as credited) used to achieve and maintain hot shutdown are NOT impacted by the failure of the secondary CT open circuit. There are NO redundant system (i.e. SSF or PSW System as credited) cables or components located in the areas of the CT or secondary CT circuit routing. This analysis satisfies the criteria of NFPA 805 and the guidance provided in NEI 00-01, Revision 1.

Based on the removal of the assumptions and the analysis of the secondary CT circuits, the B-2 Table alignment statement will be revised to state that we do align with the guidance provided in NEI 00-01, Revision 1, Section 3.5.2.1 and the NSCA shall be revised to ensure the analysis of secondary CT circuits is carried forward from the cable selection performed in EIR 51-5044354-004.

Item 3:

In order to align with NEI 00-01 Section 3.5.2.1, ONS is removing the assumptions from the B-2 Table and NSCA and ensuring the analysis for secondary CT circuit for associated circuits was complete utilizing the methodology in the ONS design basis document which NEI 00-01, Revision 1, Section 3.5.2.1 provides guidance for. The alignment statement will be revised to say that ONS does align.

Item 4:

(a) ONS is not utilizing the Circuit Coordination Study to determine the potential impact of fire-induced open secondary CT circuits. This statement will be removed from the B-2 Table.

(b) The methodology and conclusion for open circuit CT secondary circuits has been provided above in the response to Item 3. The results from the analysis of the secondary CT circuits demonstrate that the redundant systems (i.e. SSF or PSW System as credited) used to achieve and maintain hot shutdown are NOT impacted by the failure of the secondary CT open circuit. There are NO redundant system (i.e. SSF or PSW System as credited) cables or components located in the areas of the CT or secondary CT circuit routing. This analysis satisfies the criteria of NFPA 805 and the guidance provided in NEI 00-01, Revision 1.

Based on the response to the four items contained in this RAI, an action item in PIP O-10-2039 has been created to ensure that the B-2 Table is updated to state that ONS Does Align with NEI 00-001, Revision 1, Section 3.5.2.1. In addition, the NSCA calculation, OSC-9659, will be revised to remove the reference to the Brookhaven National Lab report and subsequent NRC internal correspondence and include the design criteria stated in the Duke Energy Fire Protection Design Basis Document. All CT cables are analyzed as noted above and any consequences will be included in the NSCA.

RAI 4-2:

LAR Section 4.4 states that, "radiological release examination is limited to that radiation release to unrestricted areas due to the direct affects of fire suppression activities and shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, limits. In the case of ONS, it is currently licensed to a liquid effluent release limit of 10 times that of 10 CFR 20 limits for specific discharge paths specified in the updated final safety analysis report (UFSAR) (License Amendment dated January 6, 1993)." There is however no mention of the limits as they are applied to smoke or air effluent release. Clarify the radiation release criteria for smoke releases to be applied at ONS for NFPA 805 compliance.

RAI 4-2 RESPONSE:

The ONS release limits for gaseous effluents conform to 10CFR20; therefore, the radiation release criteria for smoke releases also conforms to 10CFR20.

RAI 5-33.2:

In the response to RAI 5-33.1, the licensee clarified that the thermoset rather than the thermoplastic ZOI was used, providing the following justification: "OSC-9375 points out that fully enclosed internal sensitive electronics would be protected by the enclosures in which they are located. The thermoplastic ZOIs would result in a larger damage distance, but it is believed that application of the larger ZOI to cabinets housing sensitive electronics would yield potentially unrealistic results." Note that App. H of NUREG/CR-6850 does not imply that solid-state components are exposed "bare" to the cited damaging heat flux or temperature, so it seems reasonable to presume that the cited lower damage thresholds, and, therefore, correspondingly larger ZOIs than even those for thermoplastic cables, would apply to solid-state components as typically "housed" inside cabinets, for which the licensee claims additional protection when applying the thermoset damage thresholds (330°C [625°F] and 11 kW/m2 [1 BTU/ft^2]) and correspondingly smaller ZOIs. Specifically, App. H states: "If a scenario should arise involving solid-state control components as a thermal damage target, the failure criteria to be applied in screening are 3 kW/m2 (0.25 BTU/ft^2) and 65°C (150°F). The criteria for ignition of the components will assume properties similar to thermoplastic cables (0.5 BTU/ft^2 and 400°F)." Thus, it is not clear that even the lower thermoplastic ignition thresholds, and correspondingly larger-than-thermoset ZOIs, adequately represent the damage thresholds and ZOIs for sensitive electronics.

Additional guidance in Appendix S of NUREG/CR-6850 states: "The following approach is recommended for the Fire PRA: ..Assume no damage [to sensitive electronics present in] an adjacent cabinet if (1) there is a double wall with an air gap, and ... (2) the sensitive electronics have been 'qualified' above 82°C." This would seem to imply that damage to sensitive electronics inside a cabinet can be dismissed only if there is a double wall with air gap between the cabinet housing the electronics and the fire-exposing cabinet ("adjacent" cabinet) and the sensitive electronics have been qualified above 82°C (180°F), as opposed to "only" 65°C (150°F). Note that this is much more restrictive than even the thermoplastic threshold for "ignition" (205°C [400°F]) and would correspond to ZOIs even larger than those for thermoplastic cable fires. The licensee has not provided a technical basis for using the thermoset ignition thresholds, and correspondly smaller ZOIs, as damage thresholds for sensitive electronics under the identified conditions. Provide a technical basis, including appropriate citations to the analytical, research, and testing results, for the use of thermoset ignition thresholds, and correspondingly smaller ZOIs, rather than significantly lower recommended damage thresholds,and correspondingly much larger ZOIs, under these conditions. Alternatively, provide a sensitivity analysis or scoping fire modeling calculation using appropriate damage (vs. ignition) thresholds (with appropriate ZOIs, if employed) for sensitive electronics.

RAI 5-33.2 RESPONSE:

A detailed evaluation was performed to support the conclusions of the FPRA relative to sensitive electronics. The approach taken during scenario development relied on the horizontal zone of influence (ZOI) values from the generic fire modeling treatments to identify target damage including cabinets which may house sensitive electronics. The ZOI values were based on the applicable cable damage threshold for qualified cables.

To account for the lower damage threshold associated with sensitive electronics, a review of all electronics credited in fire scenarios within the Fire PRA was performed. While the specific failure temperature for each of the potentially sensitive targets is not known, NUREG/CR-6850

recommends failure criteria for solid-state control components of 3 kW/m2 (versus 11 kW/m2 for IEEE-383 qualified cables) be used for screening purposes. Accordingly, the resulting ZOI based on this input was used to locate potentially sensitive targets for inclusion in the scenario results.

While the evaluation is considered conservative given that short term exposure to elevated temperatures may not result in immediate failure of sensitive electronics, the results of this review identified no additional failures for inclusion within the existing fire scenarios. Many targets were screened based on their location relative to the ignition source(s) while others were either: 1) already failed directly, 2) failed as a result of tray damage, or 3) subsumed by other failures. The analysis to support the conclusions above will be included in a revision to reference 2 (OSC-9375), and tracked in the corrective action program.

RAI 5-69:

Provide a summary of the risk-informed change process that will be implemented as part of the self-approval process after transition to NFPA 805 has been completed (i.e., post-transition self-approval). This type of information was contained in Section 4.5.3 and Appendix X in the submittal dated October 31, 2008, but is not included in the submittal dated April 14, 2010.

RAI 5-69 RESPONSE:

Program documentation established, revised, or utilized in support of compliance with 10 CFR 50.48(c) is subject to Duke Energy configuration control processes that meet the requirements of Section 2.7.2 of NFPA 805. This includes the appropriate procedures and configuration control processes for ensuring that changes impacting the Fire Protection program are reviewed appropriately. The RI-PB post transition change process methodology is based upon the requirements of NFPA 805, and industry guidance in NEI 04-02, and RG 1.205. These requirements are summarized in the following table:

Change Evaluation Guidance Summary Table		
Document	**Section(s)**	**Topic**
NFPA 805	2.2(h), 2.2.9, 2.4.4, A.2.2(h), A.2.4.4, D.5	Change Evaluation
NEI 04-02	5.3, Appendix B, Appendix I, Appendix J	Change Evaluation, Change Evaluation Forms (App. I)
RG 1.205	C.2.2.4, C.3.1, C.3.2, C.4.3	Risk Evaluation, Standard License Condition, Change Evaluation Process, Fire PRA

The Plant Change Evaluation Process consists of the following 4 steps and is depicted in the figure below:

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

Change Definition

The Change Evaluation process begins by defining the change or altered condition to be examined and the baseline configuration as defined by the Licensing Basis (NFPA 805 Licensing Basis post-transition).

1. The baseline is defined as that plant condition or configuration that is consistent with the Licensing Basis (NFPA 805 Licensing Basis post-transition).
2. The changed or altered condition or configuration that is not consistent with the Licensing Basis is defined as the proposed alternative.

Preliminary Risk Review

Once the definition of the change is established, a screening is then performed to identify and resolve minor changes to the fire protection program. This screening is consistent with fire protection regulatory review processes in place at nuclear plants under traditional licensing bases. This screening process is modeled after the NEI 02-03 process. This process will address most administrative changes (e.g., changes to the combustible control program, organizational changes, etc.).

The characteristics of an acceptable screening process that meets the "assessment of the acceptability of risk" requirement of Section 2.4.4 of NFPA 805 are:

- The quality of the screen is sufficient to ensure that potentially greater than minimal risk increases receive detailed risk assessments appropriate to the level of risk.
- The screening process must be documented and be available for inspection by the NRC.
- The screening process does not pose undue evaluation or maintenance burden.

If any of the above is not met, proceed to the Risk Evaluation step.

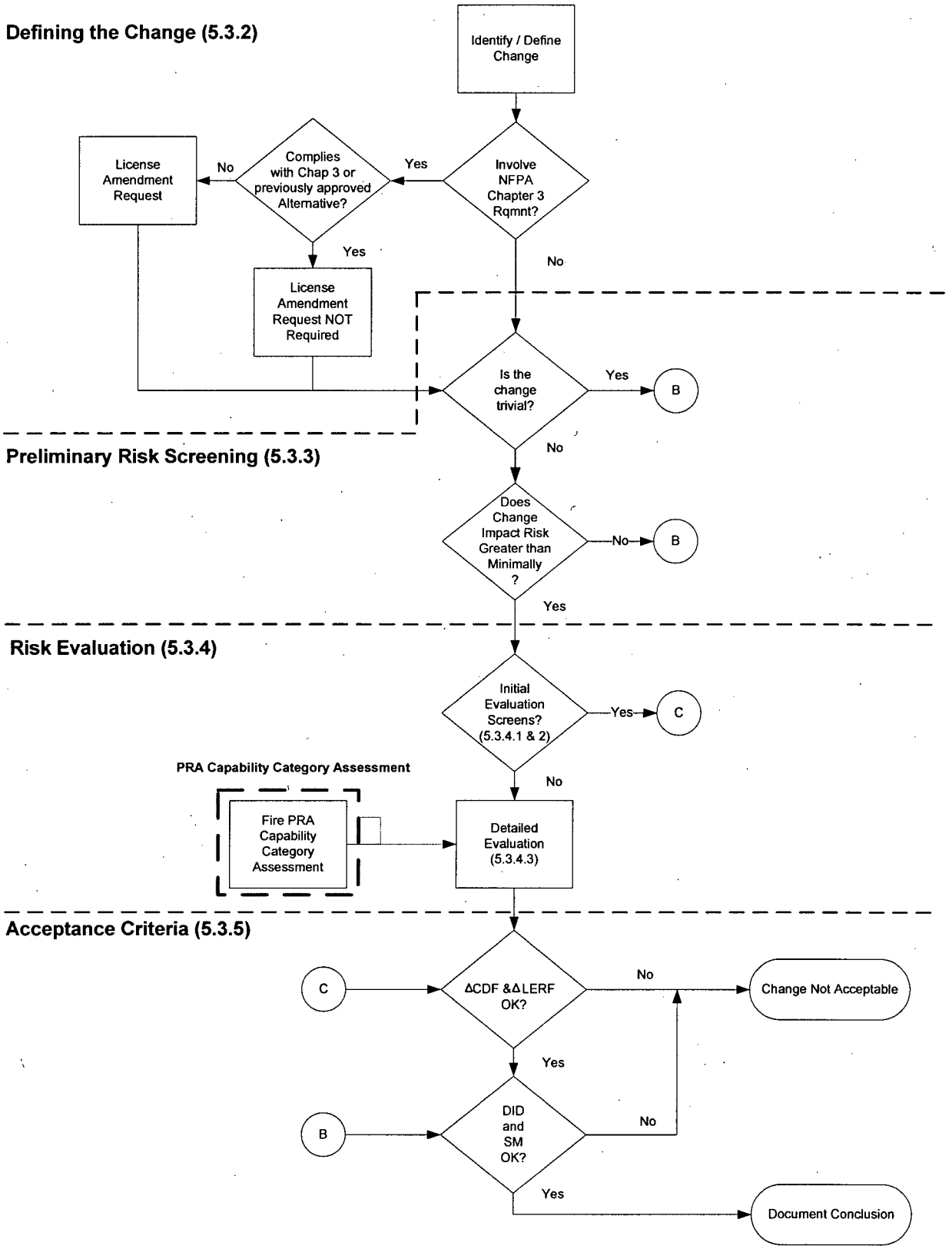
Risk Evaluation

The screening is followed by engineering evaluations that may include fire modeling and risk assessment techniques. The results of these evaluations are then compared to the acceptance criteria. Changes that satisfy the acceptance criteria of NFPA 805 Section 2.4.4 and the license condition can be implemented within the framework provided by NFPA 805. Changes that do not satisfy the acceptance criteria cannot be implemented within this framework. The acceptance criteria require that the resultant change in CDF and LERF be consistent with the license condition. The acceptance criteria also include consideration of defense-in-depth and safety margin, which would typically be qualitative in nature.

The risk evaluation involves the application of fire modeling analyses and risk assessment techniques to obtain a measure of the changes in risk associated with the proposed change. In certain circumstances, an initial evaluation in the development of the risk assessment could be a simplified analysis using bounding assumptions provided the use of such assumptions does not unnecessarily challenge the acceptance criteria discussed below.

Acceptability Determination

The Change Evaluations are assessed for acceptability using the ΔCDF (change in core damage frequency) and ΔLERF (change in large early release frequency) criteria from the license condition. The proposed changes are also assessed to ensure they are consistent with the defense-in-depth philosophy and that sufficient safety margins were maintained.



Plant Change Evaluation [NEI 04-02 Figure 5-1]
Note references in Figure refer to NEI 04-02 Sections

See response to RAI 7-2 for additional details.

RAI 5-70:

During the May 18 and 19, 2010, audit, the licensee clarified that the fault tree and basic event failure data that are used for the auxiliary service water (ASW) system hardware are a reasonable model of the PSW system's function to supply water to the steam generators. The licensee reported that the ASW system will be removed as part of the modification that installs the PSW system. Apparently, the risk decrease associated with the new PSW is included in all change in risk calculations by "toggling on" the PSW in the going forward model and "toggling off" the PSW in the current model. This method does not include the risk increase associated with removing the ASW system.

a) Summarize the PRA models used to model each of the three functions that the PSW will provide. The summary should include a brief description of the system functions modeled, the basic events (both hardware and human actions) used in each functions' models, and the values assigned to these events, and the reliability of the function.

b) Provide change in risk estimates that include the risk increase associated with removing the existing ASW system.

RAI 5-70 RESPONSE:

RAI 5-71:

The risk decrease associated with the installation of the PSW system more than offsets the risk increases associated with retaining the VFDRs and results in a total risk decrease associated with the transition to NFPA-805. The reliability/availability of the PSW is therefore the major assumption relied upon to demonstrate that the transition to NFPA-805 is acceptable. Please describe the process to confirm that the final estimates developed for the as-built, as-operated, PSW are consistent with, or are 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.

RAI 5-71 RESPONSE:

RAI 5-72a:

LAR, Attachment U, identifies facts and observations (F&Os) for internal event supporting the human reliability (HR) requirements HR-A2, HR-A3, HR-D6, HR-G3, HR-G4, HR-G6, and HR-G9 which are related to the development and quantification of human actions. The F&Os have not yet been resolved either by changes to the PRA or by final clarification demonstrating the F&O reflected inadequate documentation instead of non-conforming methods. Human error probabilities (HEP) can have a major impact on PRA results and if inaccurate values were corrected, this might change the results of the change-in-risk analyses. Provide an evaluation and discussion that demonstrates that resolution of the

F&Os is not expected to result in the change-in-risk associated with transition becoming positive instead of negative as currently estimated.

The submittal dated February 9, 2009, described a sensitivity study where the median HEP estimates (identified in HR-G9) were converted to mean HEP estimates. Mean values should be used in PRA calculations. Converting the medians to the mean for the selected HEPs resulted in an increase in the fire CDF by 3E-05, an increase almost as large as the risk decrease due to addition of the PSW. Page 24 of the February 9, 2009 submittal described the sensitivity study as an "update." Subsequent inspection of Table B-1 in ONS calculation ONS-9377 on the share point site indicates that the HEP values used in the reported change-in-risk calculations are the median values. Provide a quantative evaluation that demonstrates that the impact of using mean values instead of median values will not cause the change-in-risk associated with transition to become positive instead of negative. In addition, provide a schedule for when the human reliability analyses and associated HEPs for the internal events and fire PRAs will be upgraded to use the mean values and meet the associated RG 1.200 endorsed standard supporting requirements.

RAI 5-72a RESPONSE:

RAI 5-72b:

Inspection of ONS-9377 also identified some apparent discrepancies. For example, event FEFEFW2DHE is labeled, "operator fails to align emergency feed water from another unit within 23 minutes." In the "Time Avail. For Action in min." column, however, 13 minutes is given. Similar differences appear in many HEPs. Please explain the apparent discrepancy.

RAI 5-72b RESPONSE:

RAI 5-73:

The evaluation for a fire in the Auxiliary Building Fire Area AB, Appendix G, Page G-5 states "Deployment and operation of the SSF Submersible Pump" is an action taken at a primary control station (PCS). The retrieval, assembly, and water body deployment of the portable submersible pump including necessary hose(s) and electrical power are not predominantly conducted in the SSF or deployed during the initial transfer of control from the control room. Furthermore, there are two manual valves (2CCWVA0026 and 2CCWVA0028) in fire area YARD that require repositioning to make the flow path functional. While control of the final deployed system is performed at the SSF, and therefore not a recovery action per Regulatory Position 2.4 of RG 1.205, Rev. 1, the NRC staff disagrees that the submersible pump deployment and activation process is an action performed at the PCS. Provide the change in risk for all submersible pump deployment and activation actions and provide justification why these actions should not be considered recovery actions not previously approved by the NRC.

RAI 5-73 RESPONSE:

RAI 5-74:

Clarify when the breaker coordination study for all three units will be completed. Describe how any lack-of-coordination between breakers will be incorporated into the FPRA and reflected in the change in risk associated with transition to NFPA-805. If the breaker coordination study for all three units will not be completed in a timely fashion so that any lack-of-coordination can be incorporated into the estimate of the transition risk, describe the process the licensee will use to confirm that the final transition risk estimates are consistent with, or are bounded by, the reported estimate. The process should include the actions to be taken if the final estimates cause the transition risk to become positive instead of negative as currently estimated.

RAI 5-74 RESPONSE:

The breaker coordination study was completed on August 26, 2010. For those cases of lack-of-coordination between breakers that caused an increase in the FPRA, the coordination concern will be corrected through the engineering change process to place a fuse in series with the branch circuit breaker to correct the non-coordination concern. Other means to ensure coordination may be substituted provided that the FPRA model is not impacted. The modification will ensure that coordination exists for those cases and will also ensure that the FPRA model remains accurate.

The results of the breaker coordination calculation and subsequent insights from the FPRA model have identified the following breakers that require modification to ensure proper breaker coordination.

Unit 3:

- 3XB-1BB
- 3XB-1BT
- 3XB-2BT
- 3DP-F3CL

Unit 2:

None

Unit 1:

None

The modification scope and schedule is described in RAI response 1-14.

An action item in PIP O-10-2039 has been created to ensure that the breaker coordination results are finalized in the next update to the Fire PRA Model. In addition, an action item has been included in PIP O-10-2039 to ensure the NSCA has incorporated all related non-coordinated information and that the assumptions are still accurate.

Previous responses regarding breaker coordination (RAI 3-3 and 3-3.1) were reviewed to ensure that all responses align. Individual responses to RAI 3-3 and 3-3.1 can be viewed in submittals dated August 3, 2009 and April 14, 2010, respectively.

RAI 5-75:

Table 4-4 in the April 14, 2010, submittal indicates (with an "R") that some fire zones require fire detectors to meet the risk criteria for the Performance-Based Approach based on the risk assessment. In some fire zones or areas, a reference to footnote 2 indicates that plant modifications will be required. 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.

RAI 5-75 RESPONSE:

RAI 5-76:

LAR, Attachment K, states that the finding of a walkdown of the ONS, Unit 1, Reactor Building Pressurizer Level Instruments that "The transmitters' physical location within the Reactor Building is less than the 15 feet described in the exemption documentation" (Appendix R Exemption, "Reactor Building 20 feet separation w/o intervening combustibles," being transitioned) is being resolved under PIP O-08-03241. This deficiency should be classified as a VFDR. Provide the change-in-risk associated with this deficiency and the total change-in-risk for the ONS, Unit 1, Reactor Building. Describe the process the licensee will use to

confirm that the final estimates developed for the actual VFDRs are consistent with, or are bounded by, the initial estimate. The process should include the actions to be taken if the final estimates cause the acceptable change-in-risk guidelines to be exceeded.

RAI 5-76 RESPONSE:

The Fire Risk Evaluation for the Unit 1 Reactor Building (OSC-9315), included VFDR RB1-11 for the pressurizer level instrumentation. The Fire PRA treatment of the VFDR is described in Attachment A (Section A.2.10) of the Fire Risk Evaluation. It includes the basis used for determining the delta risk values. The basis states that pressurizer level is not a critical indication necessary for support of any modeled operator actions and therefore, have not been explicitly modeled. For purposes of the Fire Risk Evaluation, the VFDR was evaluated as having risk contributions of epsilon (ε) for both delta CDF and delta LERF. For clarity, the PRA treatment discussed in the Fire Risk Evaluation will be revised to denote the physical separation aspects of the pressurizer level; however, based on the FPRA model, the risk estimate is independent of the cable separation for this indication. The Fire Risk Evaluation revision will be tracked within the corrective action program.

RAI 5-77:

LAR, Attachment C, Table B-3, has numerous entries of "TBD" for the table entry "FRE/Change Eva/Mod Reference." Clarify the meaning of these entries.

RAI 5-77 RESPONSE:

"TBD" stands for "To Be Determined". The field specified above is used to document the internal reference number of the Fire Risk Evaluation, Change Evaluation, or Modification assigned to the VFDRs. For VFDRs that are being resolved by plant modification, a reference document number was not assigned at the time of the LAR, submitted April 14, 2010; hence, 'TBD' was used.

RAI 5-78:

The fire PRA is built upon the internal events PRA. Provide the CDF and LERF for the internal events PRA for each ONS unit.

RAI 5-78 RESPONSE:

There is only a Unit 3 PRA model available for ONS. This model includes both internal and external events. Considering only the internal events (external events removed from the model), the following is the CDF and LERF:

CDF= 1.69E-05/YR
LERF=2.061E-06/YR

RAI 5-79:

It has been shown that open CT circuits may cause secondary fires in locations away from the primary ignition source. What is the impact of these potential secondary fires on the change in risk estimates?

RAI 5-79 RESPONSE:

ATTACHMENT 1

REGULATORY COMMITMENTS

Attachment 1
Regulatory Commitments

The following table identifies the regulatory commitments in this document. Any other statements in this submittal represent intended or planned actions. They are provided for information purposes and are not considered to be regulatory commitments.

Commitment	Due Date
The recommended modification requires that the breaker compartments below will have a new fuse and holder installed and wired <u>MCC# 3XB</u> • Breaker 1BB • Breaker 1BT • Breaker 2BT <u>250/125VDC Distribution Center# 3DP</u> • Breaker F3CL	Unit 3 Spring 2012 Refueling Outage