Duke Energy®

DAVE BAXTER
Vice President
Oconee Nuclear Station

Duke Energy
ONO1VP / 7800 Rochester Highway
Seneca, SC 29672

864-873-4460
864-873-4208 fax
dabaxter@dukeenergy.com

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U. S. Nuclear Regulatory Commission
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Subject: Duke Energy Carolinas, LLC
Oconee Nuclear Station, Units 1, 2, and 3
Docket Numbers 50-269, 50-270, and 50-287
Request for Additional Information regarding the License Amendment Request to adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)
License Amendment Request (LAR) 2008-01

In accordance with 10 CFR 50.90, Duke Energy Carolinas, LLC (Duke) proposes to amend Renewed Facility Operating Licenses (FOLs) Nos. DPR-38, DPR-47, and DPR-55. This LAR requests Nuclear Regulatory Commission (NRC) review and approval for adoption of a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a), 10 CFR 50.48(c), and the guidance in Regulatory Guide (RG) 1.205. The LAR was submitted to the NRC on October 31, 2008 and supplemented in letters dated January 30, 2009, February 9, 2009, February 23, 2009, and May 31, 2009.

In order to complete review of the LAR, the NRC issued a preliminary request for additional information (RAI). Duke and the NRC met May 13 & 14, 2009, to review the RAIs to ensure that there was a common understanding of the requirements to address the requests. As a result of that meeting, several RAIs were withdrawn and are noted as withdrawn in Enclosure 1. The remaining RAIs were issued formally on June 18, 2009. Enclosure 1 contains the responses to the RAIs.

Duke provided responses to a majority of the RAIs. Preliminary information was provided concerning Recovery Actions due to ongoing discussions between the NFPA-805 task force and the NRC concerning FAQ 07-30, Operator Manual Action Transition to Recovery Actions. When these issues are finalized, Duke will address these questions. Scoping for one of the modifications listed in Attachment S of the Transition Report is included in Attachment 2. After referenced analyses and calculations are completed, Duke will re-issue a rewrite of affected sections of the LAR and associated attachments.

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Some RAI responses are preliminary information and subject to change as indicated in the applicable RAI responses. These responses pend the outcome of the following ongoing work:

- Completion of Code Evaluations
- Revision to Change Evaluations based on outcome of FAQ 07-30
- Revision of the B-1, B-2, and B-3 Tables and associated documentation based on outcome of FAQ 07-30
- Completion of the Breaker Coordination Study
- Completion of IN 92-18 Study which is potentially affected by the outcome of FAQ 07-30
- Revision to the Fire PRA based on outcome of FAQ 07-30
- Completion of the Recovery Actions Feasibility Calculation which is affected by the outcome of FAQ 07-30

A project schedule has been posted to the shared website for reference. The schedule will be updated to reflect progress as work is completed.

Commitments are provided in Attachment 3.

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on August 3, 2009.

Sincerely,

Dave Baxter, Vice President
Oconee Nuclear Station

Enclosure:

1. Request for Additional Information regarding the License Amendment Request to adopt NFPA 805 Performance-Based Standard for Fire Protection for Light Water Reactor Generating Plants (2001 Edition)

Attachments:

1. AREVA Change No. A0000650-03 Rev. 2.
2. Modification Scoping
3. Commitments

cc: w/o enclosures

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

Mr. Luis Reyes, Regional Administrator
U. S. Nuclear Regulatory Commission - Region II
Atlanta Federal Center
61 Forsyth St., SW, Suite 23T85
Atlanta, Georgia 30303

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

Mr. Eric Riggs
Acting Senior Resident Inspector
Oconee Nuclear Station

Ms. Susan Jenkins, Manager
Infectious and Radioactive Waste Management Section
2600 Bull Street
Columbia, SC 29201

Enclosure 1

REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO ADOPT NFPA 805 PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS (2001 EDITION)

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION REGARDING THE LICENSE AMENDMENT REQUEST TO ADOPT NFPA 805 PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR GENERATING PLANTS (2001 EDITION)

REQUEST FOR ADDITIONAL INFORMATION (RAI) 1-1:

Provide specific commitment dates for the plant modifications and procedure changes required for the NFPA 805 transition to be completed. Enclosure 6 of the LAR identifies a regulatory commitment to provide a schedule for implementation/completion of plant modifications, studies, and/or evaluations identified in Attachment S.

RAI 1-1 RESPONSE:

Procedure changes are addressed under the response to RAI 3-10. In a submittal dated May 31, 2009, Duke provided a partial listing of modifications and associated implementation dates that will bring ONS into deterministic compliance and reduce overall plant risk. The scoping for the spurious operation of Reactor Coolant (RC) head and loop vent valve modification is included in Attachment 2 along with modification commitment dates. The schedule for implementation/completion of the remainder of the plant modifications, studies, and/or evaluations identified in Attachment S cannot be fully developed at this time pending resolution of the following items:

a) Issues related to FAQ 07-30 (OMA Transition to Recovery Actions) need to be resolved. The FAQ feeds into the change evaluation process which impacts the scope of modifications required for implementation of NFPA-805. Until the scope of the modifications is determined it is impractical to develop a schedule. Duke has almost completed scoping potential risk informed modifications; however, the uncertainty associated with this issue prevents submission of these as committed modifications.

b) A common understanding of Regulatory Guide 1.174 (An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis) requirements for utilizing the fire PRA is needed. If the fire PRA results have to be summed with other plant PRA results (e.g. Internal events), the scope of the modifications required for implementation of NFPA-805 may be impacted.

These issues were discussed and feedback requested from the NRC in a conference call on July 10, 2009.

RAI 1-2:

Provide a list and description of the methods and types of changes that ONS intends to use to potentially make future plant changes without NRC approval. Include in the response methods that have previously been approved by the NRC, methods for which ONS is seeking NRC approval in the LAR, and non-risk-informed changes. Provide justification for the use of these methods and changes or, alternatively, provide a crosswalk to the appropriate section in the LAR where this justification is detailed.

Section 2.4.3.3 of NFPA 805 requires the PSA approach, methods, and data shall be acceptable to the AHJ. They shall be appropriate for the nature and scope of the change being evaluated, be based on the as-built, as-operated, and maintained plant, and reflect the operating experience at the plant.

RAI 1-2 RESPONSE:

Please note that this RAI response is preliminary and will be finalized when ONS Calculation, OSC-9518, Oconee Fire PRA Application Calculation, is completed. This response also pends resolution of outstanding issues for self approval currently being addressed as part of the NRC/industry pilot process and is affected by the outcome of FAQ 07-30.

The scope of plant physical or Fire Protection Program changes that could occur that require treatment as part of a Change Evaluation is potentially extensive. However, there are restrictions to the use of the Change Evaluation process as the mechanism to allow self-approval under the provisions to the Post Transition Fire Protection License Condition.

NFPA 805, 2001 states that, for the Fire PRA (FPRA) utilized under NFPA 805, the "approach, methods and data shall be acceptable to the AHJ." The AHJ (Authority Having Jurisdiction) in this case is the NRC. In order to determine acceptability, the parts of the PRA required by the application are to be assessed for technical adequacy. For the purpose of this assessment, the NRC has previously determined that the implementation of Regulatory Guide (RG) 1.200 constitutes an acceptable process adequate to conclude that the PRA approach, methods, and data are acceptable for the PRA to be applied to NFPA 805. Implementation of RG 1.200 obviates the need for staff review of the base PRA.

With respect to the technical adequacy of the FPRA for NFPA 805, NRC has determined that a PRA that achieves Capability Category (CC) II for all technical elements is generally acceptable for NFPA 805 (RG 1.200). For those cases where a particular Supporting Requirement (SR) has been determined to be less than CC II, a justification has been provided. Consequently, any approach, method or data that is determined through the RG 1.200 process to meet the requirements for CC II (or CC I, where that is demonstrated to be sufficient for the application) is effectively deemed to be acceptable to the NRC for application to NFPA 805.

With regards to specific individual Change Evaluations, the aforementioned FPRA is applied to evaluate changes in risk. The manner in which that FPRA model is exercised represents another area subject to the provisions of NFPA 805, 2001. The method(s) by which the cause-effect relationships associated with the Change Evaluation are reflected in the application of the FPRA is subject to approval by the NRC.

The scope of Change Evaluations that could be performed within the limitations of the applicable License Condition is potentially extensive. In order to provide meaningful information in this response, the scope of changes is grouped into broad categories. A summary discussion for each of these categories is provided to characterize the cause-effect relationship in the context of a risk assessment that then leads to a recommended treatment approach.

The approach described here provides the flexibility to 'map' a potentially wide variety of potential plant changes into a finite set of cause-effect relationships. This mapping is important in that the finite scope of cause-effect relationship represents the extent of 'methods' that must be reviewed by the NRC for acceptability for use in support of Change Evaluations that are subsequently self-approved. It is noted that the scope of these cause-effect relationships may be expanded in the future. It is unclear at this time whether such an expansion would require a license action to amend the License Condition or whether alternative means via NEI 04-02 and RG 1.205 could obviate that need.

It is anticipated that the majority of the plant physical or Fire Protection Program changes that occur within the limitations of the License Condition can be described in the context of the categories provided in Table RAI 1-2-1. These categories, the related cause-effect relationship and recommended treatment approach can be aligned with the basic elements of a FPRA. In all cases, it is expected that the specifics of the treatment will have already been applied as part of the base plant fire risk assessment development or are otherwise acceptably addressed by one or more SRs from the PRA Standard as endorsed by RG 1.200. In cases where a particular treatment or approach had not been previously applied in the base plant fire risk assessment, the successful completion of a related Change Evaluation would require that the base model be updated to reflect the in-situ configuration. That update/upgrade of the base fire risk assessment model may require a focused-scope Peer Review to evaluate the technical acceptability of that treatment. The existing provisions and requirements of RG 1.200 are therefore relied upon for establishing the acceptability of the specific treatment as incorporated into the base fire risk assessment.

The categories listed in Table RAI 1-2-1 for Type of Change are not intended to represent a limitation of the scope of Change Evaluations addressed in this RAI response. Instead, it is intended to reflect an overall treatment framework. This framework provides the rigor to satisfy the requirements of 10 CFR 50.48(c), provides a process and mechanism whereby the cause-effect relationship can be bounded so as to establish the extent of Change Evaluation methods approved by the NRC, while also providing the flexibility to accommodate a wide range of potential changes without requiring an exhaustive listing of possible plant physical and Fire Protection Program changes.

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
1	Unprotected Cable	The presence of an unprotected cable due to new discovery or plant modification results in the altering of the scope of cables affected by postulated fire events.	**Target Scope Change** – the quantification of the risk impact is performed using the same methods applied for the base FPRA.
2	Fire Area Boundaries	Changes to Fire Area boundaries can involve performance degradation, addition of new boundaries, or removal of boundaries. The addition or removal of a boundary is directly addressed in the NUREG/CR-6850 Task 1 treatment. The degradation of a boundary can be treated in at least two ways. Both approaches assume that engineering evaluations are unable to demonstrate adequacy of the barrier. 1. A degradation of a barrier can be evaluated using fire modeling analyses consistent with that already applied in the base FPRA. This treatment would assume the specific localized degradation has no barrier worth (equivalent to a through hole). In this case the treatment is the same as item 1. 2. A bounding approach can be used that assumes the entire barrier is 'virtually' removed. In this case the treatment is the same as item 1.	**Target Scope Change** – see No. 1 OR **Partitioning** – changes to plant partitioning is performed using the same methods applied for the base FPRA.
3	Water Curtains	The treatment of a water curtain follows the approach described in item 6 for suppression.	**Target Scope Change** – see No. 1
4	Electrical Raceway Fire Barrier Systems (ERFBS) Barrier Worth	The altering of an ERFBS 'protection worth' can be treated in a FPRA in at least two ways. Both approaches assume that engineering evaluations are unable to demonstrate adequacy of the barrier. 1. A bounding approach can be used in which case the treatment is the same as item 1. 2. The evaluated 'worth' of the barrier is translated to an available time for fire suppression. The crediting of the suppression credit is applied consistent with existing methods used for the base FPRA.	**Target Scope Change** – see No. 1 OR **Change Suppression Credit** - use existing methods from the base FPRA.

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
5	Transients	Anticipated changes are associated with plant administrative controls. Such changes can affect the calculated fire ignition frequencies for the plant locations. Such changes would be evaluated using methods consistent with that used for the base FPRA.	**Initiating Event Frequency** – the calculation of fire ignition frequencies are performed using the same methods as the base FPRA.

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
6	Suppression	Changes to suppression can involve the addition or altering by some means. The scope of potential changes is extensive, but for the purposes of this discussion, does not include the removal of suppression systems or address manual suppression. It also does not include changes that improve the performance of a system as such a change would either have no measure impact on previously approved change or would result in a risk reduction. Automatic System – altering of a system such that its performance is impacted can be addressed in at least three ways. 1. A bounding approach can be used such that the 'target' that was protected would be assumed to be damaged by the fire event. In this case the treatment is the same as item 1. 2. An updated fire modeling analyses can be performed using methods consistent with that already applied in the base FPRA. In this case, the target scope for the postulated fire events is altered to change the scope of fire affected cables. In this case, the treatment is the same as item 1. 3. A change may only affect the reliability of the suppression system. In this case, the numerical treatment (failure probability) can be changed in the risk calculation. This would have a direct proportional change in the associated CDF/LERF results.	**Target Scope Change** – see No. 1 OR **Change Suppression Credit** – see No. 4

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
7	Passive Fire Protection (FP) Features (dikes, curbs, etc)	Changes to fire protection features such as dikes and curbs can affect the spread of combustible fluid fires. As such, the treatment of such events is performed using methods consistent with that applied in the base FPRA. In general, the related fire modeling analyses should be updated using methods consistent with that already applied in the base FPRA. In this case, the target scope for the postulated fire events is altered to change the scope of fire affected cables. In this case, the treatment is the same as item 1. In the case where features are added, the treatment would result in no change in CDF/LERF or a reduction in which case a quantitative treatment is not required.	**Target Scope Change** – see No. 1
8	Embedded Conduit	The treatment assumes that engineering evaluations are unable to demonstrate adequacy of the embedment. In this case, the treatment is the same as item 1.	**Target Scope Change** – see No. 1
9	Floor Drains	The treatment of floor drains should be addressed in the same manner is item 7.	**Target Scope Change** – see No. 1
10	Recovery Actions	The inclusion of a feasible recovery action provides the opportunity to mitigate the scope of a component's fire induced failure/mal-operation. The crediting of such an action should develop a human error probability (HEP) using a method consistent with the base FPRA and internal Events PRA as applicable. If a recovery action is removed (or otherwise not credited), then its assigned credit in the base FPRA should be eliminated.	**PRA Model Change** – A variety of changes can occur that require altering one or more elements of the PRA model or quantification process. Such changes would be performed using methods consistent with that applied for the base FPRA.
11	Nuclear Safety Capability Assessment (NSCA) Equipment and Cables	Changes to NSCA equipment and cables has the effect of altering the scope of predicted damage for existing fire initiating events or creating new fire initiating events. In either case, the basic approach uses methods that are consistent with that applied for the base FPRA.	**Target Scope Change** – see No. 1 AND/OR **Initiating Event Frequency** – see No. 5

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
12	Detection	The altering of plant fire detection features has an effect on manual fire suppression reliability. The addition of such a system would increase the reliability of manual suppression as the time available for those actions would be longer than previously available. The removal or degradation of such a system would have the opposite effect. The specific treatment of manual suppression reliability is performed using method consistent with the base FPRA.	**Detection Credit** – detection is addressed via the manual fire suppression credit applied in the FPRA.
13	Incipient Detection	The addition of an incipient detection system inherently has the effect of reducing the fire risk for all hazards except High Energy Arcing Faults (HEAFs). See NEI 04-02 FAQ 08-0046 for additional discussion.	**Detection Credit** – see item 12
14	Ventilation	Changes to plant ventilation features addressed here is limited to those changes that are unique to the FPRA. All other changes will be addressed using well established processes and methods for update of the internal events PRA model. Changes that are unique to the FPRA are limited to those that are associated with smoke removal and as such, will be addressed using methods consistent with that already applied for the base FPRA. The altering of the smoke removal capability has the effect of altering of the consequences of fire events that are not addressed via the target damage set.	**PRA Model Change** – see item 10

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
15	Fire Brigade	Fire Brigade changes have the effect of altering the manual suppression reliability. Since the available data for this treatment is based on industry historical experience changes that negatively affect brigade performance can only be addressed in a bounding fashion. Such a bounding treatment would be required to removal all credit for manual suppression from the FPRA – this includes both quantitative credit reflected in the CDF and LERF as well as qualitative credit that may have been used to eliminate certain initiating events and sequences.	No specific treatment proposed
16	Emergency Lighting Units	The presence or lack thereof the Emergency Lighting Units can have an effect on credited human actions. Those impacts should be addressed using methods consistent with those applied for the base FPRA and internal Events PRA as applicable. As an alternative, a bounding approach could be applied where any credit for the associated human action is eliminated from the analysis.	**PRA Model Change** – see item 10
17	Feasibility Criteria	The specific feasibility criteria addressed here refers to NUREG-1852. Such changes have no direct impact on the FPRA as the PRA equivalent of feasibility is addressed through the human reliability analysis (HRA).	No specific treatment required
18	Fire Watch	The presence of a fire watch has the practical function of an early fire alarm system for all fire events except HEAFs. As such, it would tend to reduce the fire CDF/LERF. However, if fire watches are being applied as a compensatory measure for some other degradation, then further assessments would be required.	No specific treatment proposed
19	Ignition Source	These changes can affect the calculated fire ignition frequencies for the plant locations. Such changes would be evaluated using methods consistent with that used for the base FPRA.	**Initiating Event Frequency** – the calculation of fire ignition frequencies are performed using the same methods as the base FPRA.

Table RAI 1-2-1 – Summary of Scope of Change Evaluations, Associated Cause-Effect Relationship, and Treatment Method

No.	Type of Change	Treatment Discussion	Cause-Effect Relationship/Treatment Method
20	Surveillance Intervals	The altering of surveillance intervals has the potential to change the reliability of the associated feature – suppression, detection, etc. If separate assessments or justifications have been developed to extend intervals without altering the target reliability of the feature, then no change in the risk assessment has occurred. Otherwise, refer to the item above associated with affected feature.	Refer to affected component

RAI 2-1:

Table B-3 of the TR (page 172) documents an evaluation of the exemption assumptions (RB1-37-O) and request the NRC to disposition the exemption as "prior approval clarification," declaring the exemption still valid. The approved exemption permitted less than 20 feet horizontal distance separation between safe shutdown circuits with no intervening combustibles (redundant pressurizer level instruments in the Unit 1 Reactor Building). In the exemption, the spacing between the safe shutdown circuits was referred to as "about 15 feet."

However, based on recent field measurements, the licensee has found the actual physical separation is approximately 6 feet. They state that the basis for the exemption should be no intervening combustibles and large reactor building volume, and not the specific distance (15 vs. 6 feet) between safe shutdown circuits. Provide additional justification or an evaluation on why the NRC should consider the assumptions supporting this exemption remain valid.

RAI 2-1 RESPONSE:

To address this issue, the risk-informed, performance-based change evaluation process will be utilized. A revision will be made to the change evaluation calculation for the Unit 1 Reactor Building in accordance with the project schedule. The schedule for completion of all change evaluations is dependent upon the resolution of FAQ 07-030.

RAI 2-2:

Provide clarification of what fire barrier upgrades will be provided in accordance with the open items in PIP O-08-2006 and PIP O-08-2520. How will the upgrades meet NFPA 805, where will they be implemented, and when will these upgrades be completed?

In Table B-1 of the TR, pages 72-74, for building separation and fire barriers, open items were identified concerning upgrades to fire barriers. Pages 76-79 indicate: "Duke Letter to the NRC dated May 15, 1981 responded to some modifications indicating completion or insurance company approval of existing fire door configurations." The NRC August 11, 1978, Safety Evaluation Report (SER), Section 4.9.3 states: "Fire dampers have been provided at some locations where ventilation ducts penetrate fire barriers. The licensee has proposed to upgrade ventilation duct penetrations with dampers having fire ratings equivalent to that required by fire barriers." "We find that, subject to implementation of the above modifications, ventilation duct penetrations will satisfy the objectives identified in Section 2.2 of this report and are, therefore, acceptable." Were the fire barrier penetrations ever upgraded and found to be acceptable by the NRC?

RAI 2-2 RESPONSE:

The modifications schedule will be specifically addressed in response to RAI 1-1. Modifications for fire barrier upgrades referred to in PIP O-08-2006 and PIP O-08-2520 include the West Penetration Room (West Penetration Fire Area) to Spent Fuel Pool Room (Auxiliary Building Fire Area) wall and the Auxiliary Building / Turbine Building wall.

ONS Calculation OSC-9298 , Fire Protection Evaluation for West Penetration Room to Auxiliary Building Purge Inlet Room, Revision 1, evaluates the fire area separation between the West

Penetration Rooms (West Penetration Fire Area) to Spent Fuel Pools (Auxiliary Building Fire Area). The barrier to the Spent Fuel Pool requires upgrade as the Purge Inlet Room adjacent to the Spent Fuel Pool was changed to become an extension of the West Penetration Fire Area. The evaluation concludes the barriers require replacement of existing doors with fire rated doors and proper penetration sealing of through penetrations. PIP O-08-2006 includes this item to track to completion. The modification will replace the existing doors with three-hour rated fire door assemblies and seal all penetrations with approved three-hour fire rated penetration designs

ONS Calculation OSC-9302, Fire Protection Evaluation for Balance of Plant (BOP) Fire Area Partitioning, Revision 0, evaluated the fire area partitioning of the BOP fire area into seven new fire areas. The evaluation concluded that the fire barrier (wall) between the Auxiliary Building and the Turbine Building requires upgrade of mechanical penetrations to fire rated penetration designs in some specific wall sections that were beyond the existing committed locations. All other features (wall construction, dampers, doors) support the fire barrier rating of the wall. PIP O-08-2520 includes this item to track to completion. The modification will seal all unrated penetrations with approved three-hour fire rated penetration designs.

Upon completion of these modifications, the fire areas at ONS will be separated by three hour rated designs (wall, floor, ceiling, doors, dampers and/or penetrations) or have other engineering evaluations supporting the adequacy of fire area separation Therefore, the fire barriers required by NFPA 805 Chapter 4 will meet the requirements of Section 3.11 of NFPA 805.

In response to the specific RAI question "Were the fire barrier penetrations ever upgraded and found to be acceptable by the NRC?", a review was performed of the wording where ONS credited prior approval and part of the approval statement included reference to a modification. In this specific case, the modification referenced in the SER was completed. Note that ONS Calculation OSC-9295, NFPA 805 Transition B-1 Table/Report, Revision 1, provided a conclusive statement that "Current fire barrier ventilation duct penetrations are provided with fire dampers." This statement was omitted in Table B-1. All other instances where prior NRC approval compliance was cited and a modification was referenced, the table included a closure statement.

Duke committed to perform a study to review ventilation systems in the initial response to the November 10, 1977 RAI in a letter from Duke to the NRC dated November 22, 1977. Duke responded to the specific RAI in a letter on January 25, 1978, which included attached drawings that were annotated to show HVAC penetrations. The RAI response included drawings that were coded to denote which barriers were provided with dampers and which barriers were proposed for upgrading. Due to the age of the letter, the drawings were not attached in the current document file. However in the February 1982 Branch Technical Position APCSB 9.5-1 comparison document, Section 4.A.10, the station confirmed that fire-rated dampers were installed in fire barrier walls. In addition, as part of the transition, ONS reviewed all fire barriers required for fire area separation and confirmed appropriate barrier elements are provided (rated or equivalent walls, floors, ceilings, doors, dampers, and/or penetration seals). Where rated separation was not confirmed, engineering evaluations document the adequacy of the barrier or modifications were recommended (PIP O-08-2006 and PIP O-08-2520).

Transition Report Attachment A, Table B-1 and ONS calculation OSC-9295, NFPA 805 Transition B-1 Table/Report, Revision 1 will be updated. The update will include enhanced wording for clarity where prior approval is cited and modifications are referenced. In addition, clarification will be provided where modifications for compliance are referenced in order to clearly indicate what modifications will occur.

RAI 2-3:

Provide technical justification for lack of suppression in the rooms identified below, and clarify what elements of safe shutdown separation are being applied for the exemption request, III.G.3 or III.G.2.

The areas (BOP, BH12, and BH-3), previously credited as III.G.3 fire safe shutdown due to lack of III.G.2 separation of redundant safe shutdown cables and electrical equipment, did not have suppression as required by III.G.3. This open item attempts to resolve this lack of a documented exemption to III.G.3 by requesting that the NRC approve this license position as documented in Attachment T as "Prior Approval Clarification Request 2" of the TR (page T-5):

"As part of this LAR submittal and transition to NFPA 805, it is requested that the NRC formally document as a 'prior approval' previous acceptance by the NRC in the early 1980's that installation of fixed suppression and detection systems is not necessary to comply with Section III.G.3 of 10 *CFR,* Part 50, Appendix R for fire areas BOP, BH12, and BH3, which utilize a dedicated shutdown capability." [Attachment T of the TR]

Attachment T of the TR provides no technical justification for the lack of suppression and/or detection in the rooms that may require dedicated shutdown. Additionally Table B-3, page 7 indicates under Suppression Requirements: "Suppression required for III.G.3 area. There is no fixed suppression or fire hose stations in BH12; however, hose stations are provided in adjoining Turbine Building, as are yard hydrants. Approved fire extinguishers are provided in Blockhouses. In Area BH12-25-O, there is no suppression or an approved exemption for the lack of suppression as required by Appendix R, Section III.G.3."

In addition, Table B-3, page 39 indicates under BH3-15-O Fire Detection and Suppression per III.G.3: "This area previously credited an III.G.3 alternative means of shutdown due to lack of III.G.2 separation of redundant safe shutdown cables and electrical equipment. An NFPA 805 evaluation will be performed to ensure a fire will damage only a single train of safe shutdown equipment." Disposition for this open item states: "This open item will be resolved by NRC approval of this license position as documented in Attachment T of the LAR." BH3 disposition indicates the desire for the NRC to accept the Attachment T request for no suppression required by III.G.3, but the above BH3 disposition indicates that only one train is affected for any fire in the room, as per III.G.2.

RAI 2-3 RESPONSE:

Prior to Transition, the BOP (Auxiliary Building and Turbine Building), BH12, and BH3 fire areas were considered III.G.3 areas (SSF shutdown). Following transition to NFPA 805, implementation of the SSF shutdown strategy will only be credited in the Auxiliary Building fire area due to the implementation of the Protected Service Water (PSW) modification. Following transition (a completion of committed modifications), the Turbine Building, BH12 and BH3 YARD E fire areas will not require implementation of the SSF shutdown strategy.

The Auxiliary Building will remain a SSF shutdown area following transition and is not provided with total fixed suppression and detection systems as required by Appendix R Section III.G.3. It is ONS contention that the licensing history provided in Attachment T as "Prior Approval Clarification Request 2" of the TR (pages T-5 and T-6) indicates the NRC accepted the as-built plant fire protection design through NRC and ONS correspondence.

The NRC evaluated the Oconee Fire Protection Program in the Fire Protection Safety Evaluation Report (SER) dated August 11, 1978. The SER was based on docketed information including the ONS "Fire Hazards Analysis and Response to BTP 9.5-1" dated December 31, 1976 and multiple licensee's responses to RAIs and staff positions. The NRC evaluated plant features and specific plant areas. For specific plant areas, the NRC reviewed the combustibles, the consequences of no fire suppression, fire protection systems, adequacy of fire protection and proposed modifications. The specific plant areas evaluated include the Control Rooms (Auxiliary Building Elevation 822), Auxiliary Building Elevation 838, Auxiliary Building Elevation 796, Auxiliary Building Elevations 711 and 783, Auxiliary Building Elevation 758, Cable Spreading Rooms (Auxiliary Building Elevation 809), Battery Rooms (Auxiliary Building Elevation 809), and Penetration Areas (Auxiliary Building Elevations 809 and 822).

As stated in Attachment T, and in the August 11, 1978 SER page 9:

> *"The licensee's approach to fire protection for this plant is different from methods used by other licensees to meet the staff BTP or objectives outlined in Section 2.2 of this report. The difference is that the licensee has proposed to install a dedicated shutdown system which will enable the plant to be shutdown even if other systems are adversely affected. For this reason, the staff has not required as heavy a reliance on the ability to promptly detect or suppress fires as we have requested in other plants. This option is addressed in BTP 9.5-1 as an acceptable alternative to meeting the specific fire protection measures prescribed by BTP 9.5-1. In other plants, the plant can be brought to safe shutdown but an independent system has not been proposed. Therefore, the ability to detect and suppress a fire has greater significance than in this case where we have the dedicated shutdown system."*

The SER goes on to state that for the aforementioned specific fire areas evaluated:

> *"We find that, subject to implementation of the above described modifications and interim measures, fire protection for this area satisfied the objectives of Section 2.2 of this report and is, therefore, acceptable".* SER Section 2.2 states *"The overall objectives of the fire protection program embodied in BTP 9.5-1 and Appendix A, are to:*
>
> *(1) reduce the likelihood of occurrence of fires;*
> *(2) promptly detect and extinguish fires if they occur;*
> *(3) maintain the capability to safely shut down the plant if fires occur; and*
> *(4) prevent the release of a significant amount of radioactive materials if a fires occur."*

The modifications referenced in the specific plant area conclusions included addition or relocation of smoke detectors, sealing unprotected openings, providing portable fire extinguishers, providing a dedicated safe shutdown system independent of the control room, providing manual actuated water spray systems in the equipment and cable rooms, and providing additional hose stations. These modifications have been implemented.
Although the separation and fire protection requirements of III.G.3 are not met, the NRC approved the Auxiliary Building fire protection features as acceptable to maintain the capability

to safely shut down the plant if a fire occurs, which is equivalent to the objective of 10 CFR 50, Appendix R, Section III.G.

The following discussion is provided to describe the fire protection features and hazards located in the Auxiliary Building.

The Auxiliary Building has the following fixed suppression and detection systems:

Suppression

- Unit 1: Hatch (partial area system/automatic); Cable Room (complete/manual); Equipment Room (complete/manual); Cable Shaft (complete/manual)
- Unit 2: Hatch (partial area system/automatic); Cable Room (complete/manual); Equipment Room (complete/manual) ; Cable Shaft (complete/manual)
- Unit 3: Hatch (partial area system/automatic); Cable Room (complete/manual); Equipment Room (complete/manual) ; Cable Shaft (complete/manual)

Detection

- Unit 1: LPI Room, HPI Room, LPI Hatch area, Hatch Area, Battery Room, Cable Room, Equipment Room, Main Purge Exhaust Room, 771 and 783 corridors, East Penetration Room, Cable Shaft
- Unit 1&2: LPI Pump, HPI Pump, RB Component Coolers, Control Room, Control Room HVAC Room
- Unit 2: LPI Room, HPI Room, LPI Hatch area, Chemistry Storage, Chemistry and RP Laboratory areas, Hatch Area, Battery Room, Cable Room, Equipment Room, Main Purge Exhaust Room, 771 and 783 corridors, East Penetration Room, Cable Shaft
- Unit 3: LPI Room, HPI Room, LPI Hatch area, Chemistry Storage, Chemistry and RP Laboratory areas, Hatch Area, Battery Room, Cable Room, Equipment Room, Main Purge Exhaust Room, 771 and 783 corridors, East Penetration Room, Control Room, Control Room HVAC Room, RB Component Coolers, Cable Shaft

The following areas are not provided with detection: Tank Rooms (771' & 783'), HP Pump Hatch Area, Pipe Room, Spent Fuel Cooler Rooms, Locker/Change Rooms (796'), Drumming Area (U1; automatic suppression provided), Decontamination Area (U2), Spent Fuel Pool and Fuel Loading Areas, Ventilation Equipment Rooms, Purge Exhaust Rooms, and the Hot Machine Shop.

According to the Design Basis Specification for Fire Protection, "there are numerous ignition sources and combustible materials in the Auxiliary Building, which are typical of machinery and industrial spaces. However, there are no significant fire hazards in the Auxiliary Building." Based on plant walkdowns, significant numbers of ignition sources are generally located in the Control Rooms, Cable Rooms, and Equipment Rooms. The general areas in the Auxiliary Building have more limited quantities of ignition sources. The combustibles present in the Auxiliary Building generally consist of cabling. Areas with significant quantities of cabling are protected by either automatic smoke detection or automatic smoke detection and fixed suppression.

Attachment T, Prior Approval Clarification Request 2 will be revised to eliminate BOP fire area and discuss the new fire areas (AB and TB). Table B-3 will be reviewed and updated accordingly to clarify the lack of suppression in the III.G.3 areas for the pre-transition configuration.

RAI 2-4

Provide the intended code of record for each code/standard referenced as part of the NFPA 805 Chapter 3 compliance evaluation/statements (Attachment A of the Transition Report (TR)). In addition, provide specific commitment dates for completing the remaining code compliance reviews.

National Fire Protection Association (NFPA) standard NFPA 805, Chapter 3 "...contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features." Transition Report, Attachment A, Table B-1 notes that many actions required to come into compliance with the referenced codes and standards have not yet been completed. Some examples include:

> NFPA 805, Section 3.3.7.1 Bulk Flammable Gas, has identified "Further Action Required" as still needing to perform a NFPA 50A code compliance review.
>
> NFPA 805, Section 3.3.7.2 requires resolution of the propane tank orientation (not yet completed) for code compliance.
>
> NFPA 805, Section 3.6.1 requires an NFPA 14 code compliance review. This item is noted as closed in Table B-1 to a still open PIP O-08-2163. It is not clear when this review will be complete, nor if any modifications to the plant, plant procedures, or a performance based exemption request will be required.
>
> NFPA 805, Section 3.2.2.4 requires that policy documents identify the appropriate AHJ. The plant notes that further action is required to change current documents to identify NRC as the AHJ.
>
> NFPA 805, Section 3.3.12 references an NFPA 701 requirement for plastic sheeting that is not yet met but will be when procedure NSD-313 is revised.

In addition, TR Table B-1 and Table P-2 provide information about codes and standards required for transition to NFPA 805. In some cases, the plant already complies or has an existing deviation/exemption from a code requirement, in other cases compliance with the code will be a new NFPA 805 derived requirement. The codes and standards referenced in this standard refer to the edition of the code or standard in effect at the time the fire protection systems or feature was designed or specifically committed to the authority having jurisdiction (AHJ). The submittal needs to be clear to which edition of the code or standard required to meet NFPA 805 Chapter 3 requirements is specifically being committed.

RAI 2-4 RESPONSE:

The code of record is defined in NFPA 805 Section 1.8 as, "The codes and standards referenced in this standard refer to the edition of the code or standard in effect at the time the fire protection systems or feature was designed or specifically committed to the AHJ."

Generally, a code of record is determined based on the code year that was in effect at the time the fire protection system or feature was designed or the time when the specific commitment was made.

ONS does not have a clear NFPA code of record commitment in previous licensing documentation with the exception of acceptance of deviations to NFPA 72D and a few undated specific cases (e.g. NFPA-4, 6, 7, 8, and 27).

ONS has elected to evaluate fire protection systems and features to the current edition of the NFPA codes for convenience. Evaluation of the fire protection systems and features to the current edition of the NFPA code ensures that the current positions on design, installation, maintenance, and testing are evaluated.

ONS will revise Table B-1 to include the codes years evaluated. A listing of the code reference date evaluated and the pertinent sections of Table B-1 are provided in Table 2-4-1.

The NFPA codes listed in Table 2-4-1 below are currently in the process of development of formal code reviews with the exception of NFPA 13 and 15 for water based suppression systems. The NFPA 13 and 15 water based suppression system code reviews listed in Table 2-4-2 were performed in the early 2000's and include all systems in the Power Block. Some or all of the water based suppression systems listed in Table 2-4-2 will be required by NFPA 805 Chapter 4.

Upon completion of the NFPA code compliance reviews, noncompliances that are not justified in engineering evaluation will be entered into the corrective action program and resolved as needed (e.g., submittal to the NRC for approval (Attachment L of TR), processed as plant modifications, etc.).

Note that when previous NRC approval was cited for compliance with certain code sections in Table B-1 this was based on NRC acceptance of fire protection systems as described in the letters of correspondence. ONS opted to perform detailed NFPA code compliance reviews as applicable for the transition to NFPA-805 to ensure formal uniform documentation going forward. Duke will complete code compliance reviews and identify required modifications in Attachment S by November 30, 2009.

Lastly, based on the closure memo for FAQ 06-08, NFPA Fire Protection Engineering Evaluation, Revision 9, dated March 12, 2009 (ML073380976), ONS will revise the TR to remove the contents of Attachment P and therefore the reference to Table P-2 is unnecessary.

Table 2-4-1: NFPA Code Years Evaluated

NFPA Code	NFPA 805 Section	Code Year Evaluated	Code Evaluation Calculation Number**
10	3.7	2007	OSC-9675
12	3.10	2008	OSC-9676
14	3.6	2007	OSC-9677
20	3.5.3	2007	OSC-9678

Table 2-4-1: NFPA Code Years Evaluated

NFPA Code	NFPA 805 Section	Code Year Evaluated	Code Evaluation Calculation Number**
24	3.5.10	2007	OSC-9679
30	3.3.8	2008	OSC-9680
50A* (55 Chapter 10)	3.3.7.1	2008	OSC-9681
72	3.8	2008	OSC-9682
80	3.11	2007	OSC-9683
90A	3.11	2009	OSC-9684
600	3.4	2005	TBD
13	3.9	1999	See Table 2
15	3.9	1996, 2001	See Table 2

*Note that NFPA 50A has since been superceded. The requirements previously in NFPA 50A are now incorporated in NFPA 55, Chapter 10.

**Note these calculations have not been completed at the time of this response

Table 2-4-2: Water-based Suppression System Adopted Code of Records

Calculation Number	System	Code Year
KC-2167	Keowee Oil Storage Gravity Tank NFPA 15 Code Compliance	1996
KC-2136	Keowee Main Transformer Water Supply System NFPA-15 Code Compliance	1996
KC-2140	Keowee Main Lube Oil Storage Room NFPA 15 Code Compliance	1996
OSC-7816	CT-2 Transformer NFPA-15 Code Compliance	1996
OSC-7847	Transformer 3X NFPA-15 Code Compliance	1996
OSC-7849	Unit 2 EFW and TO Tank NFPA 15 Code Compliance	1996
OSC-7851	Unit 2 MTOT NFPA 15 Code Compliance	1996
OSC-7853	Unit 2 Under Mezzanine NFPA 13 Code Compliance	1999
OSC-7857	Unit 2 Under Turbine Operating Floor NFPA 13 Code Compliance	1999
OSC-7859	Unit 2 Main FDWP A NFPA 15 Code Compliance	1996
OSC-7861	Unit 2 Equipment Room NFPA-13 Code Compliance	1999
OSC-7863	Unit 2 Equipment Hatch NFPA 13 Code Compliance	1999
OSC-8003	Unit 2 Cable Room NFPA 13 Code Compliance	1999
OSC-8005	Unit 2 Cable Shaft NFPA 13 Code Compliance	1999
OSC-8010	CT-4 Transformer NFPA 15 Code Compliance	1996
OSC-8012	Unit 3 Under Mezzanine NFPA 13 Code Compliance	1999
OSC-8296	Transformer CT-1 NFPA 15 Code Compliance	2001

Table 2-4-2: Water-based Suppression System Adopted Code of Records

Calculation Number	System	Code Year
OSC-8297	Unit 1 Turbine Lube Oil Purifier NFPA 15 Code Compliance	2001
OSC-8298	Unit 1 Seal Oil Unit NFPA 15 Code Compliance	2001
OSC-8299	Transformer 2T NFPA 15 Code Compliance	2001
OSC-8300	Unit 1 Emergency Feed Water Pump NFPA 15 Code Compliance	2001
OSC-8301	Unit 2 Turbine Lube Oil Purifier NFPA 15 Code Compliance	2001
OSC-8302	Unit 2 Main Transformer T2 NFPA 15 Code Compliance	2001
OSC-8303	Unit 2 Seal Oil Unit NFPA 15 Code Compliance	2001
OSC-8333	Unit 3 Transformer 3T NFPA 15 Code Compliance	2001
OSC-8334	Unit 3 Transformer 3Y NFPA 15 Code Compliance	2001
OSC-8335	Unit 3 Transformer 3Z NFPA 15 Code Compliance	2001
OSC-8336	Transformer CT-3 NFPA 15 Code Compliance	2001
OSC-8337	Unit 3 Emergency Feed Water Pump NFPA 15 Code Compliance	2001
OSC-8338	Unit 3 Turbine Lube Oil Purifier NFPA 15 Code Compliance	2001
OSC-8339	Unit 3 Seal Oil Tank NFPA 15 Code Compliance	2001
OSC-8340	Unit 3 Spare Transformer NFPA 15 Code Compliance	2001
OSC-8341	Unit 1 Transformer 1T NFPA 15 Code Compliance	2001
OSC-8342	Unit 1 Main Transformer T1 NFPA 15 Code Compliance	2001
OSC-8343	Unit 3 Equipment Room NFPA 13 Code Compliance	1999
OSC-8462	Unit 1 Main Turbine Oil Tank and Oil Piping NFPA 15 Code Compliance	2001
OSC-8463	Unit 1 Equipment NFPA 13 Code Compliance	1999
OSC-8464	Southmount Shunt Reactor "X" Transformer NFPA 15 Code Compliance	2001
OSC-8465	Southmount Shunt Reactor "Y" Transformer NFPA 15 Code Compliance	2001
OSC-8466	Southmount Shunt Reactor "Z Transformer NFPA 15 Code Compliance	2001
OSC-8467	Asbury Shunt Reactor "X" Transformer NFPA 15 Code Compliance	2001
OSC-8468	Asbury Shunt Reactor "Y" Transformer NFPA 15 Code Compliance	2001
OSC-8474	Asbury Shunt Reactor "z" Transformer NFPA 15 Code Compliance	2001
OSC-8666	Unit 1 Mezzanine & Pump Crane Aisle (Area 1 Sys 2) NFPA 13 Code Compliance	1999
OSC-8667	Unit 1 Turbine Building Basement (Area 1 Sys 1) NFPA 13 Code Compliance	1999
OSC-8668	Unit 3 Turbine Building Under Mezzanine Floor NFPA 13 Code Compliance	1999

Table 2-4-2: Water-based Suppression System Adopted Code of Records

Calculation Number	System	Code Year
OSC-8693	Unit 1 Turbine Building Basement (Area 2 Sys 3) NFPA 13 Code Compliance	1999
OSC-8694	Unit 1 Mezzanine & Pump Crane Aisle (Area 2 Sys 4) NFPA 13 Code Compliance	1999
OSC-8830	Unit 1 Cable Room NFPA 13 Code Compliance	1999
OSC-8831	Unit 1 Cable Shaft NFPA 13 Code Compliance	1999
OSC-8832	Unit 1 Drumming Area NFPA 13 Code Compliance	1999
OSC-8833	Unit 1 Hatch Area NFPA 13 Code Compliance	1999
OSC-8834	Unit 3 Cable Room NFPA 13 Code Compliance	1999
OSC-8835	Unit 3 Cable Shaft NFPA 13 Code Compliance	1999
OSC-8836	Unit 3 Hatch Area NFPA 13 Code Compliance	1999
OSC-8837	Unit 1 Feedwater Pump "A" NFPA-15 Code Compliance	2001
OSC-8838	Unit 1 Feedwater Pump "B" NFPA 15 Code Compliance	2001
OSC-8839	Unit 2 Feedwater Pump "B" NFPA 15 Code Compliance	2001
OSC-8840	Unit 3 Feedwater Pump "A" NFPA 15 Code Compliance	2001
OSC-8841	Unit 3 Feedwater Pump "B" NFPA 15 Code Compliance	2001
OSC-8842	Transformer CT-5 NFPA 15 Code Compliance	2001
OSC-8843	Unit 3 Main Turbine Oil Tank and Oil Piping NFPA 15 Code Compliance	2001
OSC-8844	Auto Transformer "X" NFPA 15 Code Compliance	2001
OSC-8845	Auto Transformer "Y" NFPA 15 Code Compliance	2001
OSC-8846	Auto Transformer "Z" NFPA 15 Code Compliance	2001
OSC-8847	Auto Transformer "Spare NFPA 15 Code Compliance	2001

RAI 2-5

Provide clear indications in TR Attachment A, Table B-1, as to the plant's compliance status (e.g., if the plant "complies" or "complies with previous NRC approval").

In some cases, further action is listed as the Compliance Statement and the included compliance basis is unclear as to whether the plant complies with the stated condition. For example:

- NFPA 805, §3.3.5.1 requires "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." Table B-1 Compliance Basis notes, "Combustibles in concealed spaces are minimized." This makes no statement as to whether wiring is installed above suspended ceilings and, if so, if it complies with §3.3.5.1.

- NFPA 805, §3.3.7.2 requires "Outdoor high-pressure flammable gas storage containers shall be located so that the long axis is not pointed at buildings." Table B-1 Compliance statement notes that they comply (for hydrogen tanks), but it appears to only be in part – there is also a note that further action is required for the propane tanks (for both anchor bolts in an unreferenced SER and for orientation).

RAI 2-5 RESPONSE:

ONS will review TR Attachment A, Table B-1 to ensure clear indication as to the compliance status is identified and make more positive statements of compliance to eliminate questions as to if the plant complies or not. Previously, if ONS was in partial compliance with the fundamental requirement in Chapter 3 with further action required for the non-fulfilled requirement, the section was provided a Compliance Statement of Comply and Further Action Required. Further Action Required items are tracked in the corrective action program under PIP O-08-2163. There are multiple sections where ONS complies with the functional requirement, but station documentation requires revision to document compliance going forward. These sections were marked as Comply and Further Action Required.

In response to the examples cited in the RAI:

- NFPA 805 Section 3.3.5.1 – Some wiring is installed above suspended ceilings. The wiring, in general, is armored type. PIP O-09-04105 documents that power and control cables were recently observed located in the Control Room ceiling. As part of the Table B-1 update, a review of all areas with suspended ceilings will be performed and wiring, if present, dispositioned.
- NFPA 805 Section 3.3.7.2 – It has been determined that the propane tank is not classified as a high-pressure flammable gas storage container and therefore the reference to the propane tank and its orientation will be removed from Table B-1.

All 'Further Actions Required' statements will be reviewed to ensure that those actions that are required to demonstrate compliance are retained as include in LAR 'Yes' and include the commitment in the LAR. Where necessary the compliance statement and compliance basis sections will be updated.

Note that the term "Open Item" as used in the transition report refers to:

1. Items which are variances from the deterministic requirements of NFPA 805 for which change evaluations were performed,

2. Items requiring clarification of prior NRC approvals, and

3. Items requiring approval in the LAR

These actions will be commitments in the LAR. Where necessary, the compliance statement and compliance basis sections will be updated.

TR Attachment A, Table B-1 and ONS calculation, OSC-9295, NFPA 805 Transition B-1 Table/Report, Revision 1 will be updated. The update will include clear indication as to ONS compliance status. In addition, more positive statements of compliance will be provided to eliminate any ambiguity in the response.

RAI 2-6:

NFPA 805, Section 2.2.7, describes the application of Existing Engineering Equivalency Evaluations (EEEE's) when using a deterministic approach during the transition to an NFPA 805 licensing basis. One type of EEEE, commonly referred to as a "Generic Letter 86-10 (GL 86-10) evaluation," allows licensees who have adopted the standard fire protection license condition to make changes to the approved fire protection program without prior NRC approval if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. A Generic Letter 86-10 evaluation is one acceptable means of meeting the NFPA 805 EEEE acceptance criteria of "an equivalent level of fire protection compared to the deterministic requirements."

NEI 04-02, Section 4.1.1, "Transition Process Overview," notes that the licensee will review EEEE's during the transition process to ensure the quality level and the basis for acceptability is still valid. Satisfactory results from this review will provide an adequate basis to transition EEEE's for the deterministic requirements of Chapter 4 of NFPA 805. In accordance with 10 CFR 50.48(c) and NFPA 805, EEEE's which evaluate deviations from NFPA 805, Chapter 3, requirements must be submitted to the NRC for approval.

Evaluations of ONS's compliance with several NFPA standards are referenced in a number of Table B-1 elements. A detailed summary of the results of each of these compliance evaluations should be provided in the LAR and referenced in Table B-1.

At a minimum, each summary should include:

1. A description of all evaluated conditions determined to be acceptable based on an engineering, or other type of, evaluation, including:
   - a summary of each condition;
   - a summary of the evaluation of each condition; and
   - a summary of the resolution of each condition.

2. A description of all apparent code deviations, including:
   - a summary of each deviation;
   - a summary of the evaluation of each deviation; and
   - a summary of the resolution of each deviation.

Unless specifically limited by the Chapter 3 element, these evaluations should be completed, at a minimum, for all power block areas. (Note that certain standards, such as NFPA 600, apply to the entire plant.)

RAI 2-6 RESPONSE:

The scope of the Engineering Equivalency Evaluations (EEEs) required to be included in the LAR has changed since ONS submitted the LAR. Figure 1 below depicts the timeline and bases for this conclusion.

FAQ 06-0008, NFPA Fire Protection Engineering Evaluation, Revision 9, as endorsed by the NRC in the closure memo dated March 12, 2009 (ML073380976), provides guidance on treatment of engineering evaluations. FAQ 06-0008 concludes that functional equivalency evaluations for all section of NFPA 805 Chapter 3 and "adequate for the hazard" analyses for sections 3.8, 3.9, 3.10, and 3.11 of NFPA 805 are allowed and do not require NRC approval following transition to NFPA 805. Since NRC approval is not required for these types of evaluations following transition, it is proposed that these evaluations do not need to be summarized/included in the LAR. This concept has been discussed with NRC staff members at Pilot Plant RAI meetings and in a meeting with the NEI 04-02 Writing Team after the initial ONS RAIs were developed. This supersedes the summary in the closure memo for FAQ 07-0033 dated August 28, 2008 (ML082380395), which indicates that "adequate for the hazard" evaluations must be summarized in the licensee's transition submittals.

FAQ 07-0033, Transition of Existing Engineering Equivalency Evaluations, Revision 1, as endorsed by the NRC (ML082380395), provides guidance to licensees to review engineering evaluations included in their new licensing bases for appropriateness, quality, applicability to the current configuration, and compliance with their existing (pre-transition) licensing basis. ONS has reviewed all engineering evaluations referenced in TR Attachment A, Table B-1 to the criteria set forth in FAQ 07-0033, Revision 1, and determined that those engineering evaluations are acceptable for transition.

With respect to evaluations of ONS's compliance with NFPA standards as referenced in TR Attachment A, Table B-1, ONS will review the NFPA code compliance engineering evaluations against the requirements of FAQ 07-0033 when the evaluations are completed. These evaluations will be referenced in a revised Table B-1 by calculation number and the calculations (code compliance evaluations) will be available for NRC review. Upon ONS review of the calculations, it will then be determined if there are evaluations outside of the bounds of FAQ 06-0008 that would require summary in the LAR and require submittal to the NRC for approval.

ONS will revise the TR, specifically Attachment J – EEEE Transition. Attachment J – EEEE Transition will be revised to remove all engineering evaluations currently listed. The engineering evaluations currently listed in Attachment J are allowed and do not need to be submitted in the LAR for NRC approval.

Figure 1 - Timeline

FAQ 06-0008
(Rev 0 – 8)
Originally Submitted
9/26/08

Premise:

- No performance-based analyses (adequate for the hazard) are allowed under NFPA 805 Chapter 3 and requires approval of the performance-based method in the LAR and a license condition to allow these types of evaluations

FAQ 07-0033
(Rev 0 and 1)
Originally Submitted
6/14/07

Premise:

- Since performance-based alternatives are not allowed under NFPA 805 Chapter 3 – those existing FPEEs that are 'adequate for the hazard' do not meet NFPA 805 Chapter 3 and will be submitted to the NRC during the transition

FAQ 07-0033
(Rev 1)
Closed
8/28/08

FAQ 06-0008
(Rev 9)
Originally Submitted
2/19/09

Premise:

- Functional Equivalency Evaluation for all sections of NFPA 805 Chapter 3 allowed.
- Adequate for the Hazard Evaluations allowed for Sections 3.8, 3.9, 3.10 and 3.11 (since these are based on the results of the analyses performed in accordance with NFPA 804 Chapter 4)

FAQ 06-0008
Closed
03/12/09

Affect on LAR Submittals:

- Functional Equivalency Evaluation for all sections of NFPA 805 Chapter 3 are allowed and **do not** need to be included in LAR (include a reference to the Evaluation in the appropriate location B-1/ B-3)
- Adequate for the Hazard Evaluations are allowed for Sections 3.8, 3.9, 3.10 and 3.11 (since these are based on the results of the analyses performed in accordance with NFPA 805 Chapter 4) and **do not** need to be included in LAR (include a reference to the Evaluation in the appropriate location B-1/B-3)

RAI 3-1

The TR uses the phrases *"confirmed active fire"* and *"confirmed challenging active fire."* However, these phrases are not well defined. For example, Attachment T requests the NRC to formally document as a "prior approval," recognition that during the first 10 minutes following a *"confirmed active fire"* no spurious valve operations or loss of offsite power conditions will occur. Attachment C also states that shutdown from the SSF is initiated in response to a "*confirmed active fire*" in the turbine building, cable rooms, equipment rooms, and control rooms. However, Attachment G states that the decision to staff the SSF is tied to confirmation of a "*challenging active fire*."

During the May 13, 2009 telephone conference, the applicant agreed to provide the following:

- Documented definitions of the phrases and a description of how they are applied in the ONS post-fire safe shutdown strategy.
- Results of its review of the consistency of these phrases, as they are used in Attachments C, G, and T of the TR.
- A description of training provided to plant personnel responsible for confirming the existence of a "*confirmed active fire*" or "*confirmed challenging active fire.*"
- Clarify apparent discrepancy between TR Attachments T and G. Specifically, Attachment T states that no spurious <u>valve</u> actuations are assumed to occur in this time period, whereas, Attachment G states that no spurious <u>equipment</u> actuations are assumed to occur in this time period. Clarification is needed because assuming no spurious valve operations would involve a much smaller number of components than assuming no spurious equipment operations.

Section 1.5 of NFPA 805 states that fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. Requirements for the analyses used to support the performance-based fire protection design that fulfills the goals, objectives, and criteria provided in Chapter 1, are established in Chapter 2 of NFPA 805. Section 2.4 of NFPA 805 states that the effectiveness of the fire protection features shall be evaluated in relation to their ability to detect, control, suppress, and extinguish a fire and provide passive protection to achieve the performance criteria and not exceed the damage threshold defined in Section 2.4 for the plant area being analyzed.

NEI 04-02 Section 4.3.2 sets out a systematic process for evaluating the existing post-fire safe shutdown analysis against the methodology requirements provided in Chapter 2 of NFPA 805. RG 1.205 endorses the deterministic post-fire safe shutdown analysis methodology provided in Chapter 3 of NEI 00-01.Section 3.4.1.2 of NEI 00-01 states that a fire should be assumed to affect all unprotected cables and equipment within the fire area, without consideration of fire size or intensity.

RAI 3-1 RESPONSE:

The following definitions are used in the quantification of fires for the purpose of determining the appropriate fire response:

- A Confirmed Active Fire is defined as a locally observed fire with smoke and either radiant heat or visible flame.

Since it may be unnecessary or undesirable to take significant mitigation actions beyond what is reasonably necessary for the size of the fire at the time of discovery, ONS further qualifies the size of the confirmed active fire to determine the appropriate operator response as follows:

- A Localized Active Fire is defined as a locally verified fire where a combination of smoke and heat or flame is observed in a small localized area or affecting only a single component (failed pump bearing, single breaker cubicle, etc.). Fire damage is limited to only the affected component.

- A Challenging Active Fire is defined as a locally verified fire that is spreading/growing in size and is burning cables (bundles/trays which have the potential to affect additional equipment) outside of load center, switchgear, control board, termination cabinet or other pieces of equipment.

- An SSF Risk Area is defined as an area of the plant where the SSF is the credited method for safe shutdown and where a challenging active fire could adversely affect the safe shutdown capability of the SSF (i.e., main steam branch line isolation valves, RCS inventory isolation valves).

The safe shutdown procedures have not yet been developed. The guidance for safe shutdown may be incorporated into existing emergency operating procedures, or new procedures may be developed. However, the intent of the procedures will be to take the following actions based on the size and location of the fire:

If a fire is determined to be a "Localized Active Fire in a non-SSF Risk Area", the following actions will be taken on the Unit(s) affected by the fire:

- The station fire response procedure is entered. The fire response procedure directs activation of the fire brigade, and provides for the call-in of additional site personnel, operation of the HPSW pumps in support of fire fighting activities, and request for support from county fire departments, as necessary.

If a fire is determined to be a "Challenging Active Fire in a non-SSF Risk Area", the following actions will be taken on the Unit(s) affected by the fire:

- The station fire response procedure is entered if it is not already in progress.
- The safe shutdown procedure for the affected fire area is entered.

If a fire is determined to be a "Localized Active Fire in an SSF Risk Area", the following actions will be taken on the Unit(s) affected by the fire:

- The station fire response procedure is entered.

- Control of the RCS inventory isolation valves is transferred from the main control room to the SSF within 10 minutes of confirmation of the fire.

If a fire is determined to be a "Challenging Active Fire in an SSF Risk Area", the following actions will be taken on the Unit(s) affected by the fire:

- The station fire response procedure is entered if it is not already in progress.
- Control of the RCS inventory isolation valves is transferred from the main control room to the SSF within 10 minutes of confirmation of the fire if not already completed or in progress.
- The safe shutdown procedure for the affected fire area is entered.

TR, Attachments C, G and T of the October 31, 2008 LAR have been reviewed for consistency. The text containing the term "confirmed active fire" located in the open item of the B-3 Table of TR, Attachment C was an excerpt from a problem description out of the ONS corrective action program and is only intended to provide background information for the open item. It is not meant to express any future compliance position. Similarly, the excerpt from the NRC inspection report in TR, Attachment T containing the term "confirmed active fire" was intended to provide background information on ONS current license basis and is not meant to express any future license compliance. The term challenging active fire was used incorrectly in Attachment G. Attachment G should have stated ... "Embedded in the compliance strategy is the assumption that the 10 minute time frame does not start until confirmation of an active fire." Attachment G will be corrected in the next submittal of the transition report.

Only fire brigade qualified Operations personnel are dispatched to determine the size of a fire. The definitions for localized and challenging fires are provided in the fire alarm response guide and can be communicated to the operator at the location of the fire. The definition of challenging active fire and the basis for transferring control to the SSF is provided to licensed and non-licensed operators in lesson plan EAP-SSF, Standby Shutdown Facility.

In reference to the apparent discrepancy of using the terms "spurious valve actuations" in TR, Attachment T and the term "spurious equipment actuations" in TR, Attachment G, the use of the term "spurious valve operations" was an incorrect characterization of the request.

TR, Attachment T will be revised as necessary to convey the following request for transition to NFPA 805:

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

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

RAI 3-2:

ONS assumes that no spurious equipment operations occur, and no loss of offsite power will occur in the first ten minutes of a fire event. Further, the 10-minute time frame does not start until confirmation of an active fire. This assumption used throughout the analyses could be a key assumption. RG 1.200 defines a *key assumption* as one that is made in response to a key source of model uncertainty in the knowledge that a different reasonable alternative assumption would produce different results, or an assumption that results in an approximation made for modeling convenience in the knowledge that a more detailed model would produce different results. Provide a discussion and a sensitivity study evaluating the affects of this assumption on the results.

RAI 3-2 RESPONSE:

The 10 minute delay was not credited in the ONS analysis (FPRA). The FPRA assumed applicable spurious equipment operations coincident with other demand failures for each associated scenario. Therefore, a sensitivity analysis would not be applicable.

RAI 3-3:

The TR assumes that sufficient coordination exists for all required power supplies. However, there is currently no documented evaluation of sufficient scope to support this assumption. A coordination study has been performed to meet the current fire protection program requirements under Appendix R. However, this study requires revision in order to meet the nuclear safety capability requirements of NFPA 805 Section 2.4.2.2.2, "Other Required Circuits."

With respect to the ongoing Coordination Study, the applicant agreed to provide the following:

- Scope (e.g., what is not done in the current study and what must be done; including the identification of all <u>sensitive scenarios</u> which are impacted by an incomplete coordination study).
- Schedule for completion of this study.
- Status (any findings).
- Commitment for final product delivery. Provide a commitment, under oath and affirmation, that the coordination study will meet the requirements of NFPA 805 for circuits which share a common power supply with circuits required to achieve nuclear safety performance criteria. Specifically, provide a statement that the analysis is "sufficient to meet NFPA 805" or a commitment to complete analyses and any required modifications needed to achieve compliance within a specified schedule to be agreed upon by the staff.
- Make the existing analysis available for staff review (place on SharePoint).

Section 1.5 of NFPA 805 states that fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. Requirements for the analyses used to support the performance-based fire protection design that fulfills the goals, objectives, and criteria provided in Chapter 1, are established in Chapter 2 of NFPA 805. Section 2.4.2.2.2 of NFPA 805 states that circuits that share common power supply and/or 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.

NEI 04-02 Section 4.3.2 sets out a systematic process for evaluating the existing post-fire safe shutdown analysis against the methodology requirements provided in Chapter 2 of NFPA 805.

RG 1.205 endorses the deterministic post-fire safe shutdown analysis methodology provided in Chapter 3 of NEI 00-01. As discussed in RIS 2005-30, and NEI 00-01, all circuits whose fire-induced failure could prevent safe shutdown must be protected from the effects of fire, even non-safety circuits.

RAI 3-3 RESPONSE:

The current Appendix R Breaker Coordination is contained in ONS calculation, OSC-3120, Oconee Relay Setting and Breaker Coordination. ONS calculation, OSC-3120, was performed utilizing an exclusionary methodology that supports the current Appendix R licensing basis. Further refinement is required to meet the needs of a finite NFPA 805 analysis as stated in Section 2.4.2.2.2 of NFPA 805.

The Breaker Coordination study in support of NFPA 805 is being performed in a phased approach. The first phase "Pass 1" is described in Attachment 1, AREVA Change No. A0000650-03 Rev. 2 (included to convey preliminary scope only). Pass 1 includes all voltage levels of safe shutdown related power supplies contained in AREVA Engineering Information Record (EIR) 51-5044354, Oconee Appendix R Fire Safe Shutdown Analysis, and/or the associated Appendix R Database Management System (ARTRAK). All power supplies required by PRA and Non-Power Operations (NPO) are inclusive to the current breaker coordination study scope of "safe shutdown related" power supplies.

Pass 1 will determine if the protective device is coordinated in general, for all credible levels of fault downstream taking into account the maximum available short circuit current adjacent to the bus. Therefore, Pass 1 does not take into account any specific cable length. If Pass 1 is successful then coordination is validated. If Pass 1 is not successful additional passes may be required to justify coordination in a given plant fire area.

The second phase, Pass 2 through 6, will address any analysis beyond Pass 1 as needed to support the breaker coordination validation. If the results from Pass 1 demonstrate that coordination potentially does not exist on a selected power supply then further analysis will be performed for those feeders which are shown to be uncoordinated. The cables associated with these uncoordinated feeders will be identified and routed by Fire Area in order to determine the impact to the associated Fire Areas/Scenarios. The cables associated to the uncoordinated feeders will be documented in ARTRAK and utilized as an input to the Nuclear Safety Compliance Assessment (NSCA), FPRA Model, and the NPO Pinch Point Analysis.

ONS calculation OSC-9518, Oconee Fire PRA Application Calculation, states that "Breaker coordination impacts are not a contributing factor for top risk contributing scenarios which involve loss of 4 kV power of Control Room abandonment." This statement is based on the top 50% of CDF contributors. Only those scenarios which are not within the top 50% of risk contributors have the potential to be sensitive to uncoordinated breaker coordination; therefore, an increase in risk for many of the less risk significant scenarios will not impact overall risk.

The schedule for phase one of the breaker coordination study is as follows:

TASK DESCRIPTION	START	FINISH	STATUS
Provide DBD to ONS for review	22MAY09	22MAY09	COMPLETE
ONS to return comments to AREVA	23MAY09	26MAY09	COMPLETE

TASK DESCRIPTION	START	FINISH	STATUS
Provide list of breakers with no settings	23MAY09	03JUN09	COMPLETE
ONS to return breaker settings to AREVA	04JUN09	08JUN09	COMPLETE
Provide 25% design review to ONS		14AUG09	ON SCHEDULE
ONS to return comments to AREVA	15AUG09	24AUG09	NOT STARTED
Provide 50% design review to ONS		01OCT09	NOT STARTED
ONS to return comments to AREVA	02OCT09	07OCT09	NOT STARTED
Provide 75% design review to ONS		02NOV09	NOT STARTED
ONS to return comments to AREVA	03NOV09	06NOV09	NOT STARTED
Provide final design review to ONS		10DEC09	NOT STARTED
ONS to return comments to AREVA	11DEC09	18DEC09	NOT STARTED

The phase two schedule will be dependent upon the scope identified during phase one of the ONS breaker coordination study. The preliminary phase two schedule is:

TASK DESCRIPTION	SCHEDULE DATE
Project Kickoff	9/1/2009
DBD review	10/1/2009
25% Design Review	12/15/2009
50% Design Review	2/1/2010
75% Design Review	3/15/2010
95% Design Review	4/30/2010
Final Design Review	5/15/2010
Project Complete	5/31/2010
Final ONS acceptance	6/15/2010

The breaker coordination study is being performed as safety-related in accordance with the AREVA Quality Assurance Program. The study will be formatted in accordance with Duke procedures.

The breaker coordination study is currently on schedule to meet the Phase one 25% completion milestone and there are currently no findings at this time. For any uncoordinated breakers identified during phase one of the study a listing of circuits that are not properly coordinated will be prepared. Phase two will document the impact of the circuits that are not properly coordinated and make the following recommendations:

(a) Demonstrate coordination by refining the available short circuit current and/or trip device characteristics.

(b) Identify protective device setpoint changes (including changes in fuse size and/or clearing characteristics) that establish coordination.

(c) Incorporate the cables of concern into the nuclear safety analysis as required circuits for the affected power supply. This procedure will ensure that, on a fire area basis, the impact of common power supply circuits is evaluated.

The breaker coordination study described above will meet the requirements of NFPA 805, Section 2.4.2.2.2, for circuits that share a common power supply with circuits required to achieve nuclear safety criteria. Based on estimated phase two scope, the breaker coordination study will be issued by June 30, 2010. Any required modifications identified during the breaker coordination study will be entered into the corrective action program, compensatory actions shall be in place until the item is resolved, and the modifications will be scoped and scheduled per the modification process.

RAI 3-4:

Table B-2 of the TR (page 24), does not appear to meet the intent of the guidance provided in NEI 00-01 with respect to process monitoring and diagnostic instrumentation. Specifically, the scope of instrumentation described in NEI 00-01 is broader than Table B-2, which appears to be limited to ensuring indication for required shutdown functions only. To address this concern, provide the following:

- Provide assurance that sufficient process monitoring and diagnostic instrumentation will remain available in the event of fire in any plant area.
- Provide assurance that the credited instruments are appropriate for the shutdown strategy and the type of procedures used (e.g., event based or symptom based).
- Provide assurance that sufficient and suitable diagnostic instrumentation will be available to preclude the need for operators to enter potentially complex troubleshooting activities. Sufficient diagnostic instrumentation should be available so that operators may (i) rapidly identify/detect critical plant transients and/or spurious equipment actuations caused by fire and (ii) assess the results of their mitigating actions. As described in NEI 00-01, Chapter 3, the specific instruments required may be based on operator preference, safe shutdown procedural guidance strategy (symptomatic vs. prescriptive), and systems and paths selected for safe shutdown.

Section 1.5 of NFPA 805 states that fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. Requirements for the analyses used to support the performance-based fire protection design that fulfills the goals, objectives, and criteria provided in Chapter 1, are established in Chapter 2 of NFPA 805. Section 2.4.2.2.2 of NFPA 805 states that circuits that share common power supply and/or 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.

NEI 04-02 Section 4.3.2 sets out a systematic process for evaluating the existing post-fire safe shutdown analysis against the methodology requirements provided in Chapter 2 of NFPA 805. RG 1.205 endorses the deterministic post-fire safe shutdown analysis methodology provided in Chapter 3 of NEI 00-01.

RAI 3-4 RESPONSE:

Process Monitoring and Diagnostic Instrumentation

The following list of process monitoring and diagnostic instrumentation is credited for safe shutdown from the main control room to provide the operator with adequate indication for establishing hot standby safe shutdown conditions and for transitioning from hot standby to cold shutdown conditions:

- 'A' and 'B' loop wide range Reactor Coolant (RC) hot leg temperature, 0-700°F
- 'A' and 'B' loop wide range RC cold leg temperature, 0-700°F
- 'A' and 'B' loop wide range RC pressure, 0-2500 psig
- Pressurizer pressure, 0-2500 psig
- Pressurizer level, 0-400 inches
- 'A' and 'B' steam generator (SG) level, 0-388 inches
- 'A' and 'B' SG pressure, 0-1200 psig
- 'A' and 'B' SG emergency feedwater flow, 0-600 gpm
- Upper Surge Tank Level, 0-12 feet
- Borated Water Storage Tank Level, 0-50 feet
- Source range nuclear indication, 1 to 1E5 counts per second

The following list of process monitoring and diagnostic instrumentation is credited for safe shutdown from the standby shutdown facility to provide the operator with adequate indication for establishing hot standby safe shutdown conditions and for transitioning from hot standby to cold shutdown conditions:

- 'A' and 'B' loop wide range RC hot leg temperature, 0-700°F
- 'A' and 'B' loop wide range RC cold leg temperature, 0-700°F
- 'A' and 'B' loop wide range RC pressure, 0-2500 psig
- Pressurizer pressure, 0-2500 psig
- Pressurizer level, 0-400 inches
- 'A' and 'B' SG level, 0-388 inches
- Total unit auxiliary service water flow, 0-600 gpm

As described below, these instruments are appropriate for establishing and maintaining hot standby safe shutdown conditions and for performing a unit cooldown to cold shutdown conditions:

RC Temperature

Post trip RC temperature is maintained at 550°F to 555°F. The operators will monitor RC temperature indication to determine that the post trip hot standby RC temperature has been established. The operator will also use RC temperature, in conjunction with RC pressure indication, to maintain adequate RC subcooling margin. RC temperature will subsequently be used to establish the desired RC pressure/temperature relationship and cooldown rate during transition to cold shutdown conditions.

At least one hot leg and one cold leg temperature indication will be available in every fire area crediting the main control room for safe shutdown except the reactor building. A fire in the west side of the reactor building may result in a loss of all but one credited RC temperature indication. Due to thermal mixing of the RC inside the reactor vessel, one channel of RC temperature indication will provide representative temperature indication for the RC system as depicted in the RC hot leg and cold leg temperature plots in ONS calculation OSC-9347, Thermal Hydraulic Analysis for NFPA 805 Transition, rev. 3.

At least one hot leg and cold leg temperature indication for both 'A' and 'B' loops will be available in those fire areas crediting the SSF for safe shutdown.

RC Pressure

Post trip RC pressure is maintained at approximately 2155 psig. The operators will monitor RC pressure indication to determine that the post trip hot standby RC pressure has been established. The operator will also use RC pressure, in conjunction with RC temperature indication, to maintain adequate RC subcooling margin. RC pressure indication will subsequently be used to establish the desired RC pressure/temperature relationship during the transition to cold shutdown conditions.

'A' loop RC pressure indication will be available in every fire area crediting the main control room for safe shutdown.

'A' and 'B' loop RC pressure indication will be available in every fire area crediting the SSF for safe shutdown.

Pressurizer Level and Pressure

Post trip pressurizer level is maintained at approximately 100"-150". The operators will monitor pressurizer level to determine that the desired level has been established for hot standby safe shutdown conditions. Pressurizer level indication will subsequently be used to maintain the desired inventory in the RC system during the transition to cold shutdown conditions.

Pressurizer pressure will be used by the operator to determine when saturated conditions exist in the pressurizer.

At least one channel of pressurizer level and pressure will be available in every fire area crediting the main control room for safe shutdown.

At least one channel of pressurizer level and pressure will be available in every fire area crediting the SSF for safe shutdown.

SG Level

Post trip SG levels are maintained at 25"-30" with RCPs on and 240" with RCPs secured. The operators will monitor SG level to determine that the desired levels have been established for hot standby safe shutdown conditions. During the transition to cold shutdown the operator will monitor SG level indication to ensure that the desired levels are being maintained in the SGs.

SG level indication is available for both SGs in all fire areas crediting the main control room for safe shutdown except the reactor building fire areas. In the reactor building fire area, level indication may be lost to either the 'A' SG or the 'B' SG, but not both. One SG provides adequate heat transfer to maintain hot standby conditions and ONS calculation OSC-9347 demonstrates that feeding the affected SG at an EFDW flowrate of 50 gpm during the transition to cold shutdown provides adequate RCS and SG heat transfer while ensuring that an overfill condition in the affected SG does not develop.

SG Pressure

For safe shutdown from the main control room the operators will monitor SG pressure to determine that the desired pressure has been established for hot standby safe shutdown conditions. During the transition to cold shutdown the operator will monitor SG pressure indication to ensure that the desired pressures are being maintained in the SGs.

SG pressure indication is available for both SGs in every fire area crediting the main control room for safe shutdown except the reactor building fire areas. A fire in the east side of the reactor building may result in a loss of all steam generator pressure indication. Post trip steam pressure control will be via the main steam relief valves. Permanently installed local SG pressure gauges located in the vicinity of the atmospheric dump valves will be monitored as the atmospheric dump valves are throttled during the transition from hot standby to cold shutdown conditions.

SG pressure indication is not available for either SG in those fire areas crediting the SSF for safe shutdown. Post trip steam pressure control will be via the main steam relief valves. Permanently installed local SG pressure gauges located in the vicinity of the atmospheric dump valves will be monitored as the atmospheric dump valves are throttled during the transition from hot standby to cold shutdown conditions.

Source Range Nuclear Instrumentation

At least one channel of source range nuclear instrumentation will be available in every fire area crediting the main control room for safe shutdown. The operator will monitor source range Nis during hot standby operations and transition to cold shutdown to verify adequate shutdown margin is being maintained.

In lieu of source range instrumentation, periodic sampling of the RCS will be performed to verify adequate shutdown margin is being maintained in those fire areas crediting the SSF for safe shutdown.

RCP Indications

Though not credited for safe shutdown and not included in the safe shutdown analysis, it is desired that the RCPs remain in operation post fire until pump or plant conditions warrant that they be secured. The four RCPs are not equipped with any automatic trip interlocks and operate at a constant speed. Each RCP is equipped with the following indications:

- Pump shaft vibration
- Motor frame vibration
- Motor stator temperature
- Upper and lower motor bearing temperature
- Upper and lower motor oil pot level
- Pump seal return temperature.
- Pump seal return flow.

Post fire the operator will monitor these indications and be directed to trip the affected RCP if any of these indications are lost or indicates an abnormal trend or reading. An exception to this practice may be for a fire in the control complex. Determination of when RCPs will be secured following a fire in the control complex is pending resolution of FAQ 07-0030 and ONS request for approval of the 10 minute rule (see RAI 3-1).

Diagnosing Plant Transients

The procedures being developed for safe shutdown from the main control room will be symptom based and the credited process monitoring instrumentation described above provides the operator with adequate indication to quickly recognize and mitigate abnormal plant transients. Each post fire safe shutdown procedure will contain a list of all credited instrumentation that remains free of fire damage for the affected fire area.

A plant transient results from an imbalance between primary system heat production and secondary system heat removal. The rate of decay heat production in the primary system is relatively constant and decreases at a slow and constant rate. Therefore, a primary to secondary heat imbalance will be caused by either insufficient or excessive secondary heat removal. The two systems that can contribute to a potentially significant plant transients as a result of fire damage are the the main feedwater control system and the steam dump or turbine bypass control system.

Post trip the non-credited main feedwater control system automatically establishes a SG level of

25" with RCPs on and 240" with RCPs off. A transient in the main feedwater system will be quickly identified by an unexpected change in SG level. Alternate methods of diagnosis include unexpected changes in SG pressure, pressurizer level, RC pressure and RC temperature. If the desired SG levels are not being maintained by the main feedwater control system, the operator will be directed to secure the main feedwater pumps and start the credited emergency feedwater pumps (which will automatically start when the MFDW pumps are secured). The emergency feedwater control system automatically establishes a SG level of 30" with RCPs on and 240" with RCPs off. A transient in the emergency feedwater system will be quickly identified by an unexpected change in SG level and emergency feedwater flow. Alternate methods of diagnosis include unexpected changes in SG pressure, pressurizer level, RC pressure and RC temperature. For those fire areas where the automatic level control signal may be affected by fire damage, the operator will be directed to establish the desired SG level with the EFDW control valve in manual if the desired SG levels are not being maintained by the EFDW control system.

Post trip the non-credited main steam pressure control system automatically establishes a SG pressure of 1010 psig via the turbine bypass valves. A SG pressure of 1010 psig corresponds to an RC temperature of approximately 550°F. A transient in the steam pressure control system will be quickly identified by an unexpected change in SG pressure. Alternate methods of diagnosis include unexpected changes in SG pressure, pressurizer level, RC pressure and RC temperature. For those fire areas where the automatic level control signal may be affected by fire damage, the operator will be directed to establish the desired SG pressure with the turbine bypass valves in manual if the desired SG pressure is not being maintained by the MS control system. If the operator is unable to control SG pressure manually he/she will be directed to close the TBVs. If the operator is still unable to control SG pressure manually he/she will be directed to close the main steam isolation valves. The main steam relief valve set point of 1050 psig corresponds to an RC temperature of approximately 555°F.

SG pressure indication is available for both SGs in every fire area crediting the main control room for safe shutdown except the RB fire areas. A fire in the east side of the RB may result in a loss of all SG pressure indication. If SG pressure is lost for either the 'A' or 'B' SG during a fire in the RB, the operator will be directed to close the TBVs for both SGs.

Similarly, adequate process monitoring instrumentation is provided to diagnose spurious component operation and the safe shutdown procedures for each fire area will provide guidance to either prevent spurious component operation or mitigate the effects of spurious component operation that may occur in a given fire area.

For example, a fire in the vicinity of the 4KV switchgear located on the ground floor of the turbine building may result in the spurious start of the 'C' high pressure injection (HPI) pump. 'C' HPI pump discharges into the 'B' HPI header. 'C' HPI pump discharge pressure and 'B' HPI header flow are both credited safe shutdown instruments and, in conjunction with an unexpected increase in pressurizer level would alert the operator to a spurious pump start. In addition, as a precautionary measure, if a fire is detected in the vicinity of the 4KV switchgear, the safe shutdown procedure will direct the operator to close the 'B' HPI header isolation in anticipation of a spurious pump start.

A fire in containment or the west penetration room may result in a loss of RC letdown. A loss of RC letdown would be indicated by an unexpected increase in credited pressurizer level and RC pressure. The safe shutdown procedure will direct the operator to monitor for a loss of RC letdown and provide guidance to mitigate the affects of a loss of RC letdown.

The safe shutdown procedures for the SSF are event based in that the SSF will not be activated until all other credited methods of RC inventory control and decay heat removal are rendered inoperable due to fire damage. However, system components will be de-energized, isolated or otherwise de-activated as required to prevent spurious operation that could challenge the functionality of the SSF.

RAI 3-5 - WITHDRAWN:

The RAI dealt with plant locations where fires may damage equipment and/or cables credited to perform Key Safety Functions (KSF). During the May 13, 2009, telephone conference, the applicant stated that the information needed to address this RAI is contained in calculation OSC-9268 NFPA 805 Transition Non-Power Operations Component Selection, Rev.0. Therefore, ONS RAI 3-5 has been temporarily withdrawn pending review of this calculation.

RAI 3-6 - WITHDRAWN:

This RAI dealt with coordination of Electrical Protective Devices for NPO/KSF Power Supplies. The applicant was requested to provide assurance that the coordination of electrical protective devices associated with NPO/KSF power supplies will be bounded by the ongoing coordination study described in ONS RAI 3-3. In response to ONS RAI 3-3, the applicant states that it will provide a statement that the analysis is either, "sufficient to meet NFPA 805" or "make commitments to complete analyses. By definition, an analysis that is "Sufficient to meet NFPA 805" includes protective devices associated with NPO/KSF power supplies. Therefore, ONS RAI 3-6 (KS) is withdrawn pending review of the applicants' response to ONS RAI 3-3 (KS).

RAI 3-7:

RG 1.205 endorses the deterministic post fire safe shutdown analysis methodology provided in Chapter 3 of NEI 00-01. Section 3.2.1.2 of NEI 00-01 states that heat sensitive piping materials, including tubing with brazed or soldered joints, should be evaluated for exposure fire damage.

Provide assurance that heat sensitive piping materials, including tubing with brazed or soldered joints, have been evaluated for exposure fire damage. In addition, the applicant agreed to provide assurance that the Nuclear Safety Performance Criteria are maintained when fire affects instrument sensing lines, especially any that are constructed of materials having a relatively low melting point (such as copper) and those that include brazed or copper fittings.

RAI 3-7 RESPONSE:

The preliminary NSCA and Oconee Appendix R Fire Safe Shutdown Analysis (SSA) (AREVA EIR 51-5044354-004), consider that instrument sensing lines have the potential for inaccurate instrument indications and/or spurious equipment actuations that could occur as a result of an instrument sensing line being exposed to a fire and increased temperatures. Any instrument sensing lines that could prevent the fulfillment of the safe shutdown performance criteria have been identified and associated with the equipment that it could impact, and is included in the fire area compliance assessment for review. The NSCA and Appendix R SSA also assume that the fluid boundary associated with the sensing lines remains intact.

A review of ONS specifications, "Materials Specification for Pipe, Tubing and Fittings Nuclear Steam Supply System, Piping Installation Specification," and "Instrumentation and Controls Field Installation Standard," specify materials utilized in the sensing line pipes and fittings for process connections. A review of a population of ONS Instrument Detail drawings and Flow Diagrams, referenced below, show that the material classes utilized in the Nuclear Steam Supply Systems complies with the recommendations provided in specifications.

Based on the higher melting temperatures of the materials specified on the Instrument Detail drawings and Flow Diagrams as compared to the potential flame temperatures cited in NUREG-CR-6850, Section 8 Scoping Fire Modeling, the assumption in the NSCA and Appendix R SSA which state that the fluid boundary associated with the sensing lines remains intact is valid.

The use of copper tubing on the secondary plant systems is evaluated as a loss of the instrumentation function (i.e. indication or interlock) and is not considered an inventory loss concern that can prevent the fulfillment of a safe shutdown performance criterion.

The following drawings were utilized to review a sampling of the sensing lines installed at ONS: (The table also provides the transmitter or sensing device that is associated to the sensing lines along with the device description.)

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
1HPIFT0007A	HPI HDR A FLOW RATE	O-422-X-3 O-422-A-5 O-435-C	FD-101A-01-03
1HPIFT0008A	HPI HDR B FLOW RATE	O-422-X-3 O-422-A-5 O-435-C	FD-101A-01-03
1HPIFT0010A	MAKEUP FLOW	O-422-X-7 O-422-A-8 O-439-A O-439-D (K-K)	FD-101A-01-04
1HPIFT0075	TOTAL RCP SEAL INJ FLOW	O-422-X-24 O-422-A-7 O-437-A	FD-101A-01-04
1HPIFT0157	U1 RC MAKE UP PUMP FLOW	O-422-AA-1 O-422-X-43 O-478-E	FD-101A-01-05
1HPIFT0159	1A HPI EMERG FLOW X-OVER	O-422-X-48 O-422-A-8 O-439-A	FD-101A-01-04
1HPIFT0160	1B HPI EMERG FLOW X-OVER	O-422-X-48 O-422-A-8 O-439-A	FD-101A-01-04

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
1HPIPT0016	HPI PMP DISCH HDR PRESS	O-422-X-16 O-422-A-5 O-435-C (H-H)	FD-101A-01-03
1HPIPT0227	U1 RC MAKE UP PUMP DISCHARGE PRESSURE	O-422-AA-1 O-422-X-44 O-478-E	FD-101A-01-05
1RC LT0004P1	PZR LEVEL 1, TRAIN A HI/LO	O-422-AA-2 O-422-BB-4	FD-100A-01-02
1RC LT0004P2	PZR LEVEL 2, TRAIN A HI/LO	O-422-AA-2 O-422-BB-4	FD-100A-01-02
1RC LT0004P3	PZR LEVEL 3, TRAIN B HI/LO	O-422-AA-2 O-422-BB-4	FD-100A-01-02
1RC LT0072	U1 SSF PRESSURIZER LEVEL	O-422-AA-2 O-422-BB-4.01	FD-100A-01-02
1RC PT0021P	RC WR PRESSURE LOOP A	O-422-AA-3 O-422-BB-2	FD-100A-01-01
1RC PT0023P	RC HOT LEG A WR PRESS 2	O-422-AA-3 O-422-BB-3 O-422-BB-3.01 O-422-BB-3.02	FD-100A-01-01
1RC PT0224	U1 PRESSURIZER PRESSURE	O-422-AA-2 O-422-BB-4.01	FD-100A-01-01
1RC PT0225	U1 RC LOOP A PRESSURE	O-422-AA-3 O-422-BB-3 O-422-BB-3.01 O-422-BB-3.02	FD-100A-01-01
1RC PT0226	U1 RC LOOP B PRESSURE	O-422-AA-3 O-422-BB-3 O-422-BB-3.01 O-422-BB-3.02	FD-100A-01-01
1RC RD0005B	REACTOR COLD LEG 1A WR TEMP 2	N/A	FD-100A-01-01
1RC RD0006A	U1 REACTOR COLD LEG A WR TEMP 1	N/A	FD-100A-01-01
1RC RD0007B	REACTOR COLD LEG 1B WR TEMP 2	N/A	FD-100A-01-01
1RC RD0008A	REACTOR COLD LEG 1B WR TEMP 1	N/A	FD-100A-01-01

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
1RC RD0043A	PZR WATER TEMP 1	N/A	FD-100A-01-02
1RC RD0043B	PZR WATER TEMP 2	N/A	FD-100A-01-02
1RC RD0043C	PZR WATER TEMP 3	N/A	FD-100A-01-02
1RC RD0084A	REACTOR OUTLET LOOP 1A	N/A	FD-100A-01-01
1RC RD0084B	REACTOR COOLANT LOOP A	N/A	FD-100A-01-01
1RC RD0085A	REACTOR OUTLET LOOP 1B	N/A	FD-100A-01-01
1RC RD0085B	REACTOR OUTLET LOOP B	N/A	FD-100A-01-01
1RC TT0083	REACTOR OUTLET LOOP A	N/A	FD-100A-01-01
1RC TT0084	REACTOR INLET LOOP A	N/A	FD-100A-01-01
1RC TT0085	REACTOR INLET LOOP A	N/A	FD-100A-01-01
1RC TT0086	REACTOR OUTLET LOOP B	N/A	FD-100A-01-01
1RC TT0087	REACTOR INLET LOOP B	N/A	FD-100A-01-01
1RC TT0088	REACTOR INLET LOOP B	N/A	FD-100A-01-01
1SF PT0223	U1 RC MAKE UP PUMP SUCTION PRESSURE	O-422AA-1 O-422-X-45 O-478-E	FD-101A-01-05
2HPIFT0007A	HPI FLOW 2A HI/LO	O-1422-X-3 O-422-A-5 O-435-C	FD-101A-02-03
2HPIFT0008A	HPI FLOW 2B HI/LO	O-1422-X-3.01 O-422-A-5 O-435-C	FD-101A-02-03
2HPIFT0010A	MAKEUP FLOW TRANSMITTER	O-1422-X-7 O-1422-A-8 O-1439-A O-1439-D (C-C)	FD-101A-02-04
2HPIFT0075	TOTAL RCP SEAL INJ FLOW	O-1422-X-24 O-422-A-7 O-1444 (D-D)	FD-101A-02-04

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
2HPIFT0157	U2 RC MAKE UP PUMP FLOW	O-1422-AA-1 O-1422-X-43 O-1478-F	FD-101A-02-05
2HPIFT0159	2A HPI EMERG FLOW X-OVER	O-1422-X-48 O-1422-A-8 O-1439-A	FD-101A-02-04
2HPIFT0160	2B HPI EMERG FLOW X-OVER	O-1422-X-48.01 O-1422-A-8 O-1439-A	FD-101A-02-04
2HPIPT0016	HPI PMP DISCH HDR PRESS	O-1422-X-16 O-422-A-5 O-435-C (H-H)	FD-101A-02-03
2HPIPT0227	U2 RC MAKE UP PUMP DISCHARGE PRESSURE	O-1422-AA-1 O-1422-X-44 O-1478-F	FD-101A-02-05
2RC LT0004P1	PZR LEVEL TRAIN A HI/LO	O-1422-AA-2 O-1422-BB-4	FD-100A-02-02
2RC LT0004P2	PZR LEVEL TRAIN A HI/LO	O-1422-AA-2 O-1422-BB-4	FD-100A-02-02
2RC LT0004P3	PZR LEVEL TRAIN B HI/LO	O-1422-AA-2 O-1422-BB-4	FD-100A-02-02
2RC LT0072	U2 SSF PRESSURIZER LEVEL	O-1422-AA-1 O-1422-BB-4.01	FD-100A-02-02
2RC PT0021P	RC WR PRESS LOOP A	O-1422-AA-2 O-1422-BB-2 O-1422-KK-2	FD-100A-02-01
2RC PT0023P	RC HOT LEG A WR PRESS 2	O-1422-AA-3 O-1422-BB-3 O-1422-KK-1 O-1422-KK-2	FD-100A-02-01
2RC PT0224	RC PRESSURIZER PRESSURE	O-1422-AA-1 O-1422-BB-4.01	FD-100A-02-01
2RC PT0225	U2 RC LOOP A PRESSURE	O-1422-AA-2 O-1422-BB-3.01	FD-100A-02-01
2RC PT0226	U2 RC LOOP B PRESSURE	O-1422-AA-3 O-1422-BB-3.01	FD-100A-02-01
2RC RD0005B	REACTOR COLD LEG 2A WR TEMP 2	N/A	FD-100A-02-01

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
2RC RD0006A	REACTOR COLD LEG 2A WR TEMP 1	N/A	FD-100A-02-01
2RC RD0007B	REACTOR COLD LEG 2B WR TEMP 2	N/A	FD-100A-02-01
2RC RD0008A	REACTOR COLD LEG 2B WR TEMP 1	N/A	FD-100A-02-01
2RC RD0043A	PZR WATER TEMP 1	N/A	FD-100A-02-02
2RC RD0043B	PZR WATER TEMP 2	N/A	FD-100A-02-02
2RC RD0043C	PZR WATER TEMP 3	N/A	FD-100A-02-02
2RC RD0084A	REACTOR OUTLET LOOP 2A	N/A	FD-100A-02-01
2RC RD0084B	REACTOR COOLANT LOOP A	N/A	FD-100A-02-01
2RC RD0085A	REACTOR OUTLET LOOP 2B	N/A	FD-100A-02-01
2RC RD0085B	REACTOR OUTLET LOOP B	N/A	FD-100A-02-01
2RC TT0083	REACTOR OUTLET LOOP A	N/A	FD-100A-02-01
2RC TT0084	REACTOR INLET LOOP A	N/A	FD-100A-02-01
2RC TT0085	REACTOR INLET LOOP A	N/A	FD-100A-02-01
2RC TT0086	REACTOR OUTLET LOOP B	N/A	FD-100A-02-01
2RC TT0087	REACTOR INLET LOOP B	N/A	FD-100A-02-01
2RC TT0088	REACTOR INLET LOOP B	N/A	FD-100A-02-01
2SF PT0223	U2 RC MAKE UP PUMP SUCTION PRESSURE	O-1422AA-1 O-1422-X-45 O-1478-F	FD-101A-02-05
3HPIFT0007A	HPI HEADER FLOW RATE	O-2422-X-3 O-2422-A-5 O-2435-K	FD-101A-03-03
3HPIFT0008A	HPI HEADER B FLOW RATE	O-2422-X-3.01 O-2422-A-5 O-2435-K	FD-101A-03-03

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
3HPIFT0010A	MAKEUP FLOW	O-2422-X-7 O-2422-A-8 O-2439-A O-2439-D (B-B)	FD-101A-03-04
3HPIFT0075	TOTAL RCP SEAL INJ FLOW	O-2422-X-24 O-2422-A-7 O-2437-A (T-T)	FD-101A-03-04
3HPIFT0157	U3 RC MAKE UP PUMP FLOW	O-2422-AA-1 O-2422-X-43 O-2478-F	FD-101A-03-05
3HPIFT0159	3A HPI EMERG FLOW X-OVER	O-2422-X-48 O-2422-A-8 O-2439-A	FD-101A-03-04
3HPIFT0160	1B HPI EMERG FLOW X-OVER	O-2422-X-48.01 O-2422-A-8 O-2439-A	FD-101A-03-04
3HPIPT0016	HPI PMP DISCH HDR PRESS	O-2422-X-16 O-2422-A-5 O-2435-K (H-H)	FD-101A-03-03
3HPIPT0227	U3 RC MAKE UP PUMP DISCHARGE PRESSURE	O-2422-AA-1 O-2422-X-44 O-2478-F	FD-101A-03-05
3RC LT0004P1	PZR LVL TRAIN A	O-2422-AA-3 O-2422-BB-4	FD-100A-03-02
3RC LT0004P2	PZR LEVEL 2 TRAIN A HI/LO	O-2422-AA-3 O-2422-BB-4	FD-100A-03-02
3RC LT0004P3	PZR LVL TRAIN B	O-2422-AA-2 O-2422-BB-4	FD-100A-03-02
3RC LT0072	U3 SSF PZR LEVEL	O-2422-AA-1 O-2422-BB-4.01	FD-100A-03-02
3RC PT0021P	RC WR PRESS LOOP A	O-2422-AA-3 O-2422-BB-2 O-2422-KK-2	FD-100A-03-01
3RC PT0023P	RC HOT LEG A WR PRESS 2	O-2422-AA-3 O-2422-BB-3 O-2422-KK-2	FD-100A-03-01
3RC PT0224	U3 PZR PRESS	O-2422-AA-1 O-2422-BB-4.01	FD-100A-03-01

SUB-COMPONENT (Trans/Sensing Device)	DESCRIPTION	DRAWINGS	FLOW DIAGRAM
3RC PT0225	U3 RC LOOP A PRESSURE	O-2422-AA-2 O-2422-BB-3.01	FD-100A-03-01
3RC PT0226	U3 RC LOOP B PRESSURE	O-2422-AA-3 O-2422-BB-3.01	FD-100A-03-01
3RC RD0005B	REACTOR COLD LEG 3A WR TEMP 2	N/A	FD-100A-03-01
3RC RD0006A	U3 REACTOR COLD LEG A WR TEMP 1	N/A	FD-100A-03-01
3RC RD0007B	REACTOR COLD LEG 3B WR TEMP 2	N/A	FD-100A-03-01
3RC RD0008A	REACTOR COLD LEG 3B WR TEMP 1	N/A	FD-100A-03-01
3RC RD0043A	PZR WATER TEMP 1	N/A	FD-100A-03-02
3RC RD0043B	PZR WATER TEMP 2	N/A	FD-100A-03-02
3RC RD0043C	PZR WATER TEMP 3	N/A	FD-100A-03-02
3RC RD0084A	REACTOR OUTLET LOOP 3A	N/A	FD-100A-03-01
3RC RD0084B	RC HOT LEG A WR TEMP	N/A	FD-100A-03-01
3RC RD0085A	REACTOR OUTLET LOOP 3B	N/A	FD-100A-03-01
3RC RD0085B	RC HOT LEG B WR TEMP	N/A	FD-100A-03-01
3RC TT0083	REATOR OUTLET LOOP A	N/A	FD-100A-03-01
3RC TT0084	REACTOR INLET LOOP A	N/A	FD-100A-03-01
3RC TT0085	REACTOR INLET LOOP A	N/A	FD-100A-03-01
3RC TT0086	REACTOR OUTLET LOOP B	N/A	FD-100A-03-01
3RC TT0087	REACTOR INLET LOOP B	N/A	FD-100A-03-01
3RC TT0088	REACTOR INLET LOOP B	N/A	FD-100A-03-01
3SF PT0223	U-3 RC MAKE UP PUMP SUCTION PRESSURE	O-2422AA-1 O-2422-X-45 O-2478-F	FD-101A-03-05

RAI 3-8:

With regard to operator manual action (OMA) feasibility evaluations (TR Attachment G), provide a description of how the following elements were addressed (elements categorized according to the criteria identified in Table G-1):

1. Demonstrations
   - How were the cumulative effects of more than one Operator Manual Action (OMA) on the operating staff for a single fire evaluated?
   - As an example of maximum operator burden, provide the fire scenario/ fire area (not alternate shutdown) including the list of OMAs that creates the largest number of actions required of the operators.

2. Procedures
   - Describe how the input information for operating procedures are collected and retained in design verified documentation under Quality Assurance (QA) configuration management. How are operating procedures verified to reflect the OMAs identified in analysis results?

3. Staffing
   - Were brigade members used to perform any manual actions to enable equipment operation to satisfy nuclear safety criteria, or were only non-brigade operators are used? Explain.

4. Actions in the Fire Area
   - Were any fire areas required to be entered while smoke filled and, if so, which fire areas?
   - Did any manual actions require the use of qualified operators with regard to the use of Self-Contained Breathing Apparatus (SCBAs) or were operators required to wait for restored visibility prior to entry? If so, what fire scenarios required SCBAs or wait times for the operator?

5. Time
   - Spurious operating components during a fire scenario create the need for time-constrained responses to ensure control of the plant is maintained.
     a. What software was used to determine plant response(s)?
     b. How is the software maintained?
     c. How were the time requirements developed?
     d. How are the operators notified of time constraints?
   - How was visibility considered in the time constraints applied to manual action timelines?

Attachment G to the TR does not provide sufficient information for the staff to assess the licensee's conclusions regarding the feasibility and reliability of performing time critical manual actions.

RAI 3-8 RESPONSE:

1. Demonstrations

Manual action feasibility was performed on a fire area basis for actions required to be performed within the 1st four hours of the event. The performance of operator manual actions (OMAs) for each fire area was sequenced based on the amount of time available to perform the action prior to a plant operating parameter being challenged (i.e., pressurizer level, RC pressure/temperature). Once the order for performing the actions was identified, Operators were assigned to each action assuming the minimum available complement of qualified personnel. Each OMA demonstration started from the control room, traveled to the location of the manual action, and performed the action. If the Operator was required to perform additional actions, his subsequent actions started from the point of completion of the previous action. The demonstrations took into consideration all of the feasibility criteria contained in preliminary FAQ 07-30, rev. 0Q.

The maximum operator burden results from a west penetration room fire and the following list of actions are currently identified as being required for a fire in the Unit 1 west penetration room as documented in preliminary ONS calculation OSC-9535, Evaluation of Recovery Actions in Support Of Nuclear Safety Capability Assessment, pending resolution of FAQ 07-30 and ONS request for approval of the 10 minute rule (See RAI 3-1):

Step Sequence*	Action
Op1-1	Open breaker 5B on 1EL LX1X8 to de-energize SSF SWGR 1EL MX1XSF.
Op2-1	Open local starter breaker for 1B MDEFDWP discharge valve 1FDWVA0382 to de-energize valve.
Op2-2	Open breaker 6 on 1EL PL1KVIC to fail 1B EFDW header control valve 1FDWVA0316 open.
Op2-3	Open breaker 5 on 1EL PL1KVID to fail 1B EFDW header control valve 1FDWVA0316 open.
Op3-1	Close Valve 0IA VA1117 to isolate instrument air to WP.
Op3-2	Open Aux PZR Spray Block Valve 1HP VA0472.
Op3-3	Throttle Open 1B MDEFDWP discharge valve 1FDWVA0382 to establish EFDW flow to 1B steam generator.
Op1/2-1	Open 1A SG ADV Block Valve Bypass 1MS VA0161 to align 1A ADV for steaming
Op1/2-2	Open 1A SG ADV Block Valve 1MS VA0153 to align 1A ADV for steaming
Op1/2-3	Open 1B SG ADV Block Valve Bypass 1MS VA0163 to align 1B ADV for steaming
Op1/2-4	Open 1B SG ADV Block Valve 1MS VA0155 to align 1B ADV for steaming
Op1/2-5	Throttle open 1A SG ADV Throttle Valve Bypass 1MS VA0162 to initiate steaming of 1A SG for RCS cooldown.
Op1/2-6	Throttle open 1B SG ADV Throttle Valve Bypass 1MS VA0164 to initiate steaming of 1B SG for RCS cooldown
Op1/2-7	Throttle open 1A SG ADV Throttle Valve 1MS VA0154 to continue steaming of 1A SG for RCS cooldown.

Op1/2-8	Throttle open 1B SG ADV Throttle Valve 1MS VA0156 to continue steaming of 1B SG for RCS cooldown
Op4-1	Jumper contacts for RCP motor starting interlocks in Cabinet 1MTC3 per EP/1/A/1800/001.

* The 'Op#' is the operator number performing the action, the '-#' is the sequence step for that operator; i.e. Op1-1 is the 1st action taken by Operator 1. Operating the ADVs requires 2 operators; thus Op1/2-1 represents Operators 1 and 2 performing the 1st action in the sequence.

The shutdown scenario for this fire area utilized 4 operators to perform the time critical actions outside the control room. Preliminary FAQ 07-30, rev. 0Q states that feasibility will be performed for time critical recovery actions that are required to be taken within the first two hours of the event. ONS has chosen to demonstrate feasibility on the recovery actions that must be performed within the 1st four hours of the event. A time period of four hours was chosen as the thermal hydraulic analysis performed in ONS calculation OSC-9347, Thermal Hydraulic Analysis for NFPA 805 Transition, rev. 3 demonstrates that for the fire area with the most limiting time constraints the recovery actions required to establish hot standby conditions and initiate a cooldown to cold shutdown conditions must be accomplished within four hours of the initiating event.

2. Procedures

Operator manual actions for the safe shutdown strategies were identified for every fire area during the development of the NSCA calculation. The NSCA and supporting calculations are inputs to the Fire Protection Program design basis document which is the design deliverable document from which the operating procedures will be developed from. All of these input documents are prepared, verified, and approved in accordance with Duke's QA Category 1 requirements.

The Fire Protection Program design basis document will be the basis for the development of the safe shutdown procedures. The safe shutdown procedures are developed and maintained in accordance with NSD-703, Administrative Instructions for Technical Procedures and OMP 4-02, Verification and Validation Process for APs, EOP and Support Procedures. NSD-703 and OMP 4-02 require that the safe shutdown procedures be verified and validated by Operations personnel, and that the safe shutdown engineer provides a cross disciplinary review of the procedures prior to approval.

3. Staffing

Required Operations staffing is specified in Selected Licensing Commitment (SLC) 16.13.1, Minimum Station Staffing Requirements. SLC 16.13.1 requires that a minimum of four non-licensed operators will be available to perform post fire OMAs. These operators are in addition to those credited for fire brigade response and only non-brigade operators are credited with performing OMAs.

4. Actions in the Fire Area

No fire areas are required to be entered or traversed while smoke filled. However, following a fire in either the Unit 1 and 2 Control Complex or the Unit 3 Control Complex, the east penetration room of the affected unit may require entry to perform post fire OMAs. Both the control complexes and the east penetration rooms are located in the Auxiliary Building fire area. The control complex walls, floors and ceilings provide a 3 hour fire barrier between the control complexes and the remainder of the Auxiliary Building as documented on drawing O-310K-1 and drawing series O-310K-7 through O-310K-15. In addition, the east penetration room can be accessed from the west penetration room, which is a separate fire area, and will be accessible following a fire in either control complex.

5. Time

RETRAN-02/MOD005.2DKE and RETRAN-3D MOD3.1DKE computer codes were used to determine plant response for determining time constrained operator actions. These computer codes are maintained in accordance with software quality assurance procedure SDQA-70098-COM, RETRAN-02 MOD005.2/DKE, rev. 1D and SDQA-30218-NGO, RETRAN-3D MOD3.1DKE rev. F, respectively.

Time requirements for operator actions were developed as follows:

Components that are subject to spurious operation were identified in preliminary ONS calculation OSC-9659, Oconee Nuclear Safety Capability Assessment for Units 1, 2 and 3, for every fire area.

The fire area crediting the main control room (MCR) for safe shutdown that contains the combination of spurious component operation and inoperable components that would result in the greatest number of recovery actions was then identified. This fire area became the bounding case for MCR analysis.

A thermal hydraulic analysis of the scenario containing these inoperable components was then performed using the RETRAN computer codes. The time constrained operator actions identified as a result of this analysis are documented in ONS calculation OSC-9347, Thermal Hydraulic Analysis for NFPA 805 Transition, rev. 3.

The timeline established in ONS calculation, OSC-9347, Thermal Hydraulic Analysis for NFPA 805 Transition, rev. 3 was then used to determine the sequence of the required OMAs and the manual action feasibility study contained in preliminary ONS calculation OSC-9535, Evaluation of Recovery Actions in Support Of Nuclear Safety Capability Assessment, demonstrated that completion of these OMAs would be feasible in the allowed time.

Preliminary ONS calculations OSC-9535, Evaluation of Recovery Actions in Support Of Nuclear Safety Capability Assessment, and OSC-9659, Oconee Nuclear Safety Capability Assessment for Units 1, 2 and 3, are inputs into the Fire Protection Program design basis document. The Fire Protection Program design basis document is the design deliverable from which the operating procedures will be developed from. The timing of the OMAs are validated again during procedure development to ensure that they can be accomplished within the allowed time. The operators are trained on the time constraints associated with these OMAs during initial and continuing training on the procedures. In addition, a note or caution may be included in the procedures to alert the operator to these time constraints.

New time constrained operator responses for the fire area crediting the SSF for safe shutdown were identified from the list of components that are subject to spurious operation from preliminary ONS calculation OSC-9659 for the SSF fire area. Additionally, well established SSF specific time constrained operator actions such as transferring control to the SSF within 10 minutes of a confirmed active fire, establishing SSF Auxiliary Service Water (ASW) flow within 18 minutes, etc., are already incorporated into station abnormal operating procedures and no additional thermal hydraulic analysis was performed for these actions.

Visibility was evaluated during the determination of manual action feasibility. No credit was given for station lighting except for lighting units with fixed battery packs. To ensure that adequate lighting is available for areas of the plant where lighting units with fixed battery packs are not installed, operators will be directed to obtain and carry hand held lanterns when performing manual actions. Fifteen emergency equipment lockers are located throughout the auxiliary building and the turbine building. Each of these lockers contain four lanterns and four miner's lamps. In addition, an emergency equipment locker located outside the Unit 1 and 2 control room contains ten lanterns and ten miner's lamps. There is also an emergency equipment locker located in the SSF control room that contains ten lanterns and ten miner's lamps. These lanterns are inventoried and verified to be functioning properly on a quarterly basis using a procedure entitled, Field Equipment and Procedures Surveillance.

The completion of ONS calculation, OSC-9535, is contingent upon the pending outcome of FAQ 07-30 and its impact on change evaluations, recovery actions, and required plant modifications.

RAI 3-9 - WITHDRAWN:

During the May 13, 2009, telephone conference, the applicant stated that the information needed to address this RAI is contained in Sections 3.1.19, 3.4.1.5, and 3.4.2.2 and Attachment C of the TR. After re-review of the LAR, it was decided that the LAR provided sufficient information to answer this RAI.

RAI 3-10:

During the audit, ONS stated that NFPA 805 compliant (fuel in a safe and stable condition) procedures for fire safe shutdown to Mode 5 (CSD) have not been developed and it will not likely be ready until after the Safety Evaluation is issued. For the development of these procedures:

- Provide a schedule for the development of the required procedures.
- Include a detailed list of recovery actions and instrumentation to be credited.

RAI 3-10 RESPONSE:

Generating a schedule for the development and revision of the safe shutdown procedures will be contingent upon the completion of the design deliverables, the actual completion date of the committed station modifications, and available operator training dates. The completion dates for the design deliverables, in turn, are contingent upon the pending outcome of FAQ 07-30 and its impact on change evaluations, recovery actions, and required plant modifications. Therefore, at this time, ONS is not prepared to commit to procedure development milestones or completion dates other than to confirm that the procedures will be developed, validated, approved and issued within 18 months of issuance of the SER.

A list of recovery actions for each fire area is contained in preliminary ONS calculation, OSC-9535, Evaluation of Recovery Actions in Support Of Nuclear Safety Capability Assessment. The purpose of this calculation is to document the methodology and results of the transition of OMAs credited under the current licensing basis to the new National Fire Protection Association (NFPA) 805 licensing basis. OMAs will be transitioned to Recovery Actions using the guidance from FAQ 07-30, OMA Transition to Recovery Actions.

The following tasks are included in this calculation:

- Establishment of OMA Evaluation Groups – Binning (input to NFPA 805 Change Evaluation Calculations)
- Determination of whether or not a transitioning OMA is a post-transition Recovery Action
- Evaluation of the feasibility of the Recovery Actions credited in the analysis post-transition.

A list of recovery actions and credited instrumentation for each fire area is also contained in preliminary ONS calculation, OSC-9659, Oconee Nuclear Safety Capability Assessment for Units 1, 2 and 3. The purpose of this calculation is to document the methodology and results of the transition of a deterministic NSCA based on the current license basis to a risk informed, performance based nuclear safety capability assessment based on the National Fire Protection Association (NFPA) 805 licensing basis.

The following tasks are included in this calculation:

- Documenting the methodology used for equipment selection and for performing circuit analysis.
- Documenting the results of the fire area-by-fire area evaluations.
- Documenting how the fire PRA, change evaluations, plant modifications, and recovery actions are used to demonstrate compliance.

Completion of ONS calculations, OSC 9535 and OSC-9659 are contingent upon the pending outcome of FAQ 07-30 and its impact on change evaluations, recovery actions, and required plant modifications.

A revised list of recovery actions for each fire area will also be included in TR, Attachment G, Operator Manual Actions Transition, of the next supplemental submittal of the TR which is scheduled to occur in November, 2009. The completion date for this document is also contingent upon the pending outcome of FAQ 07-30 and its impact on change evaluations, recovery actions, and required plant modifications.

RAI 3-11:

Provide a description of the analysis and conclusions performed to ensure ONS cable configurations are bounded by the performance values for Non-IEEE 383 cables, IEEE 383 cables, and flammable liquids in the Generic Fire Modeling Treatments.

The Generic Fire Modeling Treatments delineates the fire models/correlations used to determine the Zone of Influence based on generic values of Heat Release Rate for Non-IEEE 383 cables, IEEE 383 cables, and flammable liquids. It does not however, describe how these bound the values for ONS specific cable and flammable liquids in the plant.

For Material Data Input; The Generic Fire Modeling Treatments [Sections 2.1, 2.2, and 2.3] states: "The assumed performance criteria for IEEE 383 is conservative but does not necessarily bound all types of cables that meet the IEEE 383 flame spread test. It is therefore important aspect of any analysis to assess the types of cables involved to ensure that the assumed performance is applicable." It repeats these statements regarding Non-IEEE 383 cable as well.

RAI 3-11 RESPONSE:

The performance data for ONS cables is described in the ONS fire scenario development calculation, OSC-9375. Specifically, Section 6 of the calculation provides a discussion and justification for the treatment of the ONS cables as IEEE-383 qualified using the generic values from NUREG/CR-6850.

With respect to flammable liquids, the only liquids treated in the Generic Fire Modeling Treatments are combustible fluids. Rather than attempt to identify all possible types of fluids, the treatments and its application are based on the properties of the fluid. In the ONS FPRA and the associated Change Evaluations, the only fluids that required treatment are lubricants. This treatment was previously assessed by the NRC as part of their review of the FPRA. The results of that review did not identify any issues or concerns with respect to this treatment.

RAI 4-1:

Provide a detailed description of the methods for achieving radioactive release performance criteria on a fire area-by-fire area basis, during full and low/non-power operations, and how the potential for cross-contamination (water run-off and smoke from a contaminated area being directed through an uncontaminated area) is addressed. Also, provide the following:

- The methodology used to identify which systems / components / flow paths are used to meet the release criteria (similar to safe shutdown analysis)
- Identification of FPP elements, measures / systems / procedural control actions / flow paths, credited to meet the criteria
- Description of plant programs, such as fire brigade training and equipment maintenance that are relied upon to sustain equipment reliability and fire brigade performance
- A bounding analysis, qualitative risk analysis, or detailed risk analysis that demonstrates the release criteria are met either by crediting the identified FPP elements or by showing the available source term as "too low to challenge" the criteria

Additionally, the ONS radioactive release review determined the current FPP is compliant with the NFPA 805 requirements, with the guidance in NEI 04-02 and RG 1.205, and **with the exception of the 10 *CFR*, Part 20 limits**. ONS has prior NRC approval for the concentration of radioactive material in releases of liquid effluents at anytime from the site boundary to unrestricted areas (denoted in Figure 2.1-4(a) of the ONS UFSAR) that shall be limited to 10 times the effluent concentrations specified in 10 CFR 20. In which fires does this exemption from 10 *CFR*, Part 20 limits occur? Are there any steps being applied in the procedures to measure and control the potential radiological airborne release to 10 *CFR*, Part 20 limits?

Insufficient information is provided to assess if fire areas meet the NFPA 805 radioactive release performance criteria in NFPA 805, Section 1.5.2..

RAI 4-1 RESPONSE:

The radioactive release evaluation was performed to assess the direct effects of fire suppression activities as directed by fire brigade pre-fire plans and instructions in contaminated and potentially contaminated areas per the requirements of NFPA 805, section 1.5.2 and guidance of NEI-04-02 as endorsed by RG 1.205. Fire brigade pre-fire plans for areas which did not contain or are not expected to contain contaminated materials were screened from further evaluation. The reviews were performed by the groupings provided by ONS fire pre-plans. The reviews were not performed on an area-by-area basis since ONS has a limited number of very large fire areas that encompass multiple elevations and compartments that would unnecessarily complicate the evaluation. The evaluation was not performed based on plant operating modes, since fire suppression activities in the way they are defined in the pre-fire plans and fire brigade fire fighting instructions (Standard Operating Guidelines [SOGs]) are written for any plant operating mode. The need for monitoring and control of potentially contaminated run-off into non-contaminated areas was recognized and a new fire brigade instruction, SOG 16 was developed and incorporated into training to address this need.

In response to the RAI:

- **The methodology used to identify which systems/ components / flow paths are used to meet the release criteria (similar to safe shutdown analysis).**

  The guidance provided in NEI 04-02, Revision 1 as endorsed by RG 1.205, Revision 0 defines the analysis of the Radioactive Release performance criteria as evaluation of the direct effects of fire suppression activities. Consequently, there is no methodology used to identify which systems / components / flow paths used to meet the release criteria (similar to safe shutdown analysis). The evaluation did not credit any installed SSC's similar to what would be done in a safe shutdown analysis and relies strictly on fire brigade pre-fire plans and fire fighting instructions for guidance in monitoring and controlling potentially contaminated releases.

- **Identification of Fire Protection Program elements, measures / systems / procedural control actions / flow paths, credited to meet the criteria.**

  As stated above, the Fire Protection Program elements, measures / systems / procedural control actions / flow paths, credited to meet the criteria were the fire brigade pre-fire plans and fire fighting instructions of the SOGs.

- **Description of plant programs, such as fire brigade training and equipment maintenance that are relied upon to sustain equipment reliability and fire brigade performance.**

  As stated above, the plant programs, fire brigade training and instructions are relied upon to sustain fire brigade performance.

- **A bounding analysis, qualitative risk analysis, or detailed risk analysis that demonstrates the release criteria are met either by crediting the identified Fire Protection Program elements or by showing the available source term as "too low to challenge" the criteria.**

  As stated in the NRC letter dated March 18, 2005, "With respect to radiation release performance criteria, in consideration of the very low risk associated with radioactivity release from sources other than the reactor core, the NRC agrees with NEI's proposal as reflected in Revision F of NEI 04-02." The endorsed NEI guidance does not require the performance of a bounding analysis. The bases for the guidance of NEI 04-02 regarding radiological release is that the NSCA performance goals for all modes of plant operation preserve the integrity of the fuel and the RCS boundary such that a source term for radiological release is never created and that the only concern for radiological release occurs as a result of fire fighting efforts in radiologically contaminated areas or fires in contaminated materials themselves.

  Chemistry/RP personnel are part of the responding fire brigade team. Current radiation protection procedures and practices adequately describe how to monitor and control liquid and gaseous effluents from the site. Therefore, the existing gaseous and liquid release limits for the site will be adhered to.

The 10 CFR 20 exemption for liquid effluents has been granted to the site for any liquid release and does not apply to a specific fire or fire area, event or accident. As with all radiological processes, the ALARA principles are utilized to ensure that releases are minimized below site limits.

Therefore, the specific performance goals of ensuring radioactive releases as a result of fire fighting activities stay within the limits of 10CFR 20 as modified for liquid effluents at ONS have been met due to the adequacy of the fire brigade pre-fire plans and fire fighting instructions as maintained and reinforced by the current fire brigade training program.

RAI 5-1:

Peer review finding, SR FSS-G3-1 (page V-3), was closed based on "Addressed via multi-compartment screening analysis added as Attachment D to Fire Scenario Report; no additional scenarios identified for quantification." Provide additional detail to review this resolution.

RAI 5-1 RESPONSE:

Finding SR-FSS-G3-1 noted that the FPRA documentation had not addressed the requirement to screen potential multi-compartment combinations of interest nor defined any multi-compartment fire scenarios beyond those that are inherently captured in the treatment of fire scenarios for the turbine building fire zones. Subsequent to the NRC staff review of the ONS FPRA, a multi-compartment screening analysis was performed and included as Attachment D to ONS calculation OSC-9375, Oconee Fire PRA Scenario Development Report. Previously, the multi-compartment analysis had been addressed in part by the consideration of targets in adjacent compartments (fire zones) during the development of individual fire scenarios. In other words, if a target was within the zone of influence for a given fire scenario, it was included in the scope of equipment assumed to be damaged by the fire even if that target was located or routed in another compartment. To address the potential for multi-compartment interaction due to barrier failure that may have been credited with limiting the target damage, a matrix was developed to calculate revised frequencies for each "exposing" compartment. For the purposes of multi-compartment scenario screening, this frequency was considered equivalent to a revised CDF value for the multi-compartment scenario in lieu of calculating a new CCDP based on the combined failures from the affected (exposing and exposed) compartments. Based on a screening threshold for CDF of less than 1E-07, no additional scenarios were identified for quantification.

RAI 5-2:

Provide documentation that demonstrates the PRA model used as the basis for the fire PRA analyses is acceptable per the requirements used by the NRC when the application was submitted, i.e., RG 1.200, Rev.1. RG 1.200 specifies that an acceptable PRA model is, at a minimum, one that has been reviewed against the current ASME PRA Standards as endorsed in RG 1.200, either directly or through some series of steps that yield equivalent results. Since the PRA model has apparently not been reviewed against the ASME PRA Standard RA-Sb-2005 as endorsed in RG 1.200, Rev.1, the plant must document how the existing model development and review yield an equivalent result.

RAI 5-2 RESPONSE:

The ONS Rev. 3a PRA model is used as the basis for the FPRA analyses. This model, along with previous revisions of the model, has been reviewed against the latest industry guidance documents, as indicated in the table below. Note that the ONS Rev. 3a PRA model has been reviewed against ASME PRA Standard RA-Sc-2007 as modified by RG 1.200, Rev. 1.

Table 5-2
Summary of PRA Model Review Results

Oconee PRA Version	Peer Review Date	Guidance Document	Review Results
Rev. 2	May 2001 (Ref. 1) (B&W Owner's Group)	NEI-00-02, "Industry PRA Peer Review Process," Nuclear Energy Institute, January 2000	*Strengths* • Thorough evaluation of model results and insights • Analysis of uncertainties • Modeling of RCP seal LOCAs and electric power recovery • Small break LOCA modeling • Detailed modeling of Keowee (emergency power) • Process for maintaining the model up to date • Rigorous Level 2 and 3 PRA *Recommended Areas for Improvement* • Better document independent verification of critical inputs, support system initiators and dependencies • Strengthen common cause failure analysis • Improve documentation of human error reliability dependencies and timing • Develop a more rigorous treatment of Steam Generator Tube Ruptures • Provide guidance for application of Unit 3 model to Units 1 and 2
Rev. 3	June 2006 (Ref. 2) (Maracor Software & Engineering)	ASME RA-Sa-2005, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME, 2005.	The review determined that the majority of the SRs in each technical area of the Standard meet the Category II requirements, either completely or partially. The largest overall area that would need improvement to more completely comply with the requirements of Capability Category II is that of PRA documentation. Documentation updates have been performed for some areas of the PRA, but not for all items. Also, the PRA Standard requires significantly more discussion of technical issues, key assumptions, and tests of reasonableness than has been typically developed for industry PRAs.

Table 5-3-1 (continued)
Summary of Recent PRA Model Revisions

Oconee PRA Version	Peer Review Date	Guidance Document	Review Results
Rev. 3a	October 2008 (Ref. 3) (Duke Energy Self Assessment)	ASME RA-Sc-2007, "Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications," ASME, August 2007. "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," RG 1.200, Rev. 1, US NRC, January 2007.	The Oconee PRA fully meets 236 of the 306 ASME PRA Standard Supporting Requirements (SRs), as modified by Reg. Guide 1.200. In addition, 24 of the SRs are not applicable to the Oconee PRA, either because the referenced techniques are not utilized in the PRA or because the SR is not required for Capability Category II. Of the 46 open SRs, 13 are of a technical nature. The remaining open SRs require enhanced documentation. However, none of the open items are expected to have a significant impact on the base case PRA results or insights.

References

1. Oconee Nuclear Station Probabilistic Risk Assessment Peer Review Report, 38-1288512-00, B&W Owner's Group Risk-Informed Assessment Committee, October 2001.

2. An Independent Review of the Oconee PRA Against the Requirements of the ASME PRA Standard, Maracor Software & Engineering, Inc., August 2006.

3. PRA Quality Self-Assessment, DPC-1535.00-00-0013, Rev. 1, October 2008.

RAI 5-3:

Identify, date, and briefly describe modifications made to the PRA for each version of the PRA up to version OR3a (beginning with the version that was peer reviewed). Could any facility changes that have not been modeled in the PRA affect the risk estimates important to the LAR analysis?

The PRA used to support an application should reflect the as-designed and as-operated facility. Therefore, PRAs must usually be periodically modified to update the methods and to incorporate facility changes.

RAI 5-3 RESPONSE:

The ONS PRA is periodically modified to update the analysis methods and incorporate changes in plant configuration or operation, as summarized in Table 5-3-1.

Table 5-3-1
Summary of Recent PRA Model Revisions

Oconee PRA Version	Revision Date	PRA Changes
Rev. 2	December 1996 (Ref. 1)	• Revised the systems analyses to incorporate changes in configuration and operation. This involved system walkdowns and review of plant documents to identify any changes in the composition of the system, changes in the normal or emergency operating state of system components, reviewing the system operating history, re-solving each system model and reviewing the results with the system engineer at the plant. • Updated the data analysis to include more recent industry and plant-specific operating experience. Generic equipment failure rates were obtained primarily from a database that aggregated from many sources, such as other nuclear plant PRAs, WASH-1400, IEEE-500 and INEL's generic database (EGG-SSRE-8875). Plant-specific data was compiled from the Problem Investigation Process (PIP) database, INPO LER database, NPRDS and other sources as needed. The data covered the six-year period from 1988 through 1993 for all three Oconee units and included component failure data, demands or run hours, and maintenance unavailability. Plant-specific and generic data were combined using a Bayesian-update. The human reliability analysis was updated to validate the results of the previous assessments and provide quantification of new human error events. Applicable procedures were reviewed and station personnel contacted as needed. The initiating events analysis was updated to incorporate industry and plant-specific operating experience for the period from 1980 through 1993.

Table 5-3-1 (continued)
Summary of Recent PRA Model Revisions

Oconee PRA Version	Revision Date	PRA Changes
Rev. 3	November 2006 (Ref. 2)	• Revised the systems analyses to incorporate changes in configuration and operation, as described above for Rev. 2. • Updated the data analysis to include more recent industry and plant-specific operating experience. The generic data sources were the same as for the Rev. 2 PRA; Maintenance Rule experience calculations for the period from January 1996 through June 1999 provided plant-specific information on the number of equipment failures, demands, run hours and unavailability hours. Revised the quantification for certain human error events based on operator validation data and procedure changes. Expanded the common cause failure analysis to address peer review items; performed a detailed common cause analysis for the N breakers. Partitioned the small break LOCA initiator from the OPRA Rev. 2 analysis is into the SL (small LOCA), RCPSL (RCP seal LOCA), SAFETY (pressurizer safety valve spurious operation) and PORV (pressurizer PORV spurious operation) initiators. Updated initiator frequencies using data through 2003. • Integrated a simplified LERF model • Included a rigorous treatment of human error dependency (approximately 400 human error event combinations were evaluated) • Incorporated a new (WOG2000) RCP seal LOCA model • Revised the BWST diameter in the tornado analysis

Table 5-3-1 (continued)
Summary of Recent PRA Model Revisions

Oconee PRA Version	Revision Date	PRA Changes
Rev. 3a (interim update)	June 2008 (Ref. 3)	• Revised the Low Pressure Injection system model to incorporate changes in configuration and operation. Specifically, deleted basic event EKSFAS1DHE (Operators Fail to Actuate ES Logic Module Manually) from the fault tree to fix a logic problem; revised the power supplies to several valves based on station modifications; revised the model to reflect the installation of LPI train cross-connect piping, flow restrictors and check valves, thereby eliminating the need to throttle LPI flow to prevent pump run-out. • Revised the High Pressure Injection system model to incorporate changes in configuration and operation. Specifically, deleted basic event EKSFAS1DHE from the fault tree to fix a logic problem; revised the power supplies to piggy-back valves 3LP-15 and -16 based on station modifications. • Refinements to the cut set recovery rule file (to add additional human error combination events and delete Tech. Spec. disallowed maintenance combinations)

As part of the Rev. 3 model integration process, model solution cut sets were compared with the Rev. 2 cut sets. This comparison yielded the following observations:

Reason for new Rev. 3 CDF cut sets and reason some Rev. 2 cut sets no longer appear:

- New LOCA initiators (RCPSL, PORV, SAFETY)
- New seal LOCA model used (WOG 2000) incorporating NRC-required conditions
- New HELB model and success criteria
- Accounting for HRA dependencies
- Expanded modeling of common cause failures
- Simplified LERF model added
- Added SSF submersible pump logic
- Expanded modeling of Tornado Analysis
- System modeling changes on LPSW, ESV, SSW
- Loss of Power from Bus 3KI initiator deleted since loss of 3KI alone will not result in a loss of ICS and main feedwater
- Incorporation of PRA Peer review items
- Resolution of High and Medium PRA impact items
- Lower truncation limit

Reasons for modified cut sets:

- The 18 RCP seal LOCA off-site power recovery events have been replaced by 24 new events that represent specific leakage cases
- RCP seal LOCA sequences are now partitioned into very small, small, medium and high leakage rate cases by multiplying by the appropriate split fraction (T0SMALLDEX, TMEDIUMDEX, T0LARGEDEX)
- Improved SSF & 4kV data
- Improved Common Cause Analysis
- Improved Relief Valve Values
- Revised operator action failure probabilities based on latest operating procedures and latest operator performance data
- Updated initiating events frequency
- With new SGs, greater credit given for feedwater inventory, resulting in more recovery time
- Numerous changes to exposure factors based on input from system engineers

None of these changes are considered model upgrades, which would require an additional peer review. In addition, none of the model changes from the peer reviewed version (Rev. 2) to the current model (Rev. 3a), which formed the basis of the fire PRA, significantly altered the PRA insights.

The following issues have arisen since the issuance of Rev. 3a. They are tracked in the PRA Change Database, have been evaluated for impact on the base model and are evaluated for impacts on future applications as they arise. These changes are not expected to have a significant impact on the risk estimates important to the fire LAR analysis, as discussed in Table 5-3-2 below. Note that the planned installation of Protected Service Water (PSW) has been addressed in the fire PRA model.

Table 5-3-2
Significant Outstanding PRA Changes

PRA Change Form	Description	Impact on Fire LAR
O-03-0051	Incorporate the CE Owners Group RCP seal leakage model.	Use of the newer seal LOCA model is expected to result in a reduction in CDF and LERF for seal LOCA scenarios. However, normal RCP seal leakage rates need to be reduced before the model can be incorporated.
O-03-0052	Incorporate new relay seismic fragilities into the seismic PRA.	No impact on fire sequences.
O-03-0063	Implement several changes to the ONS seismic PRA analysis.	No impact on fire sequences.
O-06-0004	Update the response and execution times for basic event VHP1ASWDHE, "Failure to Align HPI to ASW Switchgear Prior to Core Damage Following HELB."	Applies to high energy line break sequences. Not probabilistically significant with respect to fire sequences.
O-06-0005	Update the response and execution times for basic event BHP0ASWDHE, "Crew Fails to Recover Power To HPI Pump From ASW Switchgear."	Operator action applies to tornado sequences. No impact on fire sequences.
O-06-0006	Basic event VADPRESDHE, "Operators Fail To Depressurize Using The ADVs," is no longer applicable due to EOP changes.	Applies to high energy line break sequences. Not probabilistically significant with respect to fire sequences.
O-06-0026	Revise the quantification of basic event HHPHPR0DHE, "Operators Fail to Initiate High Pressure Recirculation," based on a 2006 Alden Research Report.	The revised basic event quantification results in CDF and LERF reductions. However, fire change evaluations are not expected to be significantly impacted.

Table 5-3-2 (continued)
Significant Outstanding PRA Changes

PRA Change Form	Description	Impact on Fire LAR
O-09-0001	Implement modeling changes in the Oconee PRA system models, HRA, and external event models to reflect the installation of the new PSW system.	The base case PRA model has not yet been revised to reflect the planned installation of the PSW system. However, the fire PRA considers planned modifications and does credit the PSW system.
Draft Change Form: O-09-0004	Draft:The result of a review of the impact of safe shutdown cable routing data changes on the FPRA resulted in a high risk impact on the U2 Fire PRA CDF and a low impact on the U2 LERF. The purpose of this review is to evaluate the impact of these ARTRAK changes on the Change Evaluation conclusions. Even though the delta CDF due to the changes were high, it was determined that the conclusions of the change evaluations remain unaffected.	The impact of this Draft Change Form will be reviewed and addressed by incorporating the cable routing changes in the planned update to the ONS FPRA.

References

1. ONS PRA Revision 2 Summary Report, December 1996 and other supporting documents.
2. Oconee Core Melt Model Integration Notebook, OSC-8863, Rev. 0, November 2006 and other supporting documents.
3. Oconee Core Melt Model Integration Notebook, OSC-8863, Rev. 1, June 2008 and other supporting documents.

RAI 5-4:

Provide justification for identified Fire PRA Supporting Requirements (SR) for which a Capability Category I is deemed sufficient in Attachments V and X to the LAR. Justification cannot be only a concluding statement, but rather must support the assignment of specific SR Capability

Categories. In addition, confirm that ONS has reviewed the application and determined that no SR should be capability category III for this application.

Section 3 in the ASME RA-Sb-2005 standard endorsed in RG-1.200 provides a discussion of a methodology that may be used to identify necessary and sufficient SR requirements for a particular application.

RAI 5-4 RESPONSE:

The assessment of sufficient capability category for the NFPA 805 application is provided in ONS calculation OSC-9518, NFPA 805 FPRA Application. While several supporting requirements were judged to be CC III, Duke is in agreement with the NRC position as stated in Draft Regulatory Guide DG-1218, Proposed Revision 1 of RG 1.205: for PRA Standard supporting requirements important to the NFPA 805 risk assessments, the NRC position is that CC II is generally acceptable. Several supporting requirements which were deemed to be CC I were judged to be sufficient to support the change evaluation process.

The following table was extracted from a preliminary revision to ONS calculation, OSC-9518, NFPA 805 FPRA Application Calculation.

Table 4-2 ONS Fire PRA NRC Review – Category I Summary

SR	Topic	Status
PP-B5	Crediting of active fire barrier elements	Cat 1 is acceptable for the application given that the results are scenario driven (the zone of influence was not arbitrarily limited to the zone boundary). Other than the rollup door between BH12 and CT4 which was credited for deterministic fire area partitioning, active fire barrier elements were not credited.
ES-B4	Number of fire-induced spurious operations considered	Disagree with Category assignment; actually a Cat 3. No limit was placed on the number of fire-induced spurious operations in consideration of potential ISLOCA or containment bypass scenarios.

Table 4-2 ONS Fire PRA NRC Review – Category I Summary

SR	Topic	Status
CS-A10	Identification of cable locations at raceway level (Cat 3) or at compartment level (Cat 2), versus fire area level (Cat 1)	Disagree with Category assignment; actually a Cat 3. Cable location information is available in ARTRAK at the fire area, fire zone, and raceway level. While compliance is on an area basis, the FPRA scenarios included identification of targets (including raceways), where applicable. It is noted that credit by exclusion was only applied to low risk components (designated as 'Y3') for which cable routing information was not assembled. These Y3 components were not the subject of specific change evaluation calculations.
FSS-C2	Use of time-dependent fire growth profiles	Cat 1 is acceptable for the application. While the results are conservative, no changes to the results are anticipated if time dependent fire growth profiles are assumed given the conservatism inherent in the NUREG/CR-6850 maximum heat release rates and available fire growth profiles relative to the ignition frequencies.
FSS-D9	Evaluate potential for smoke damage	Potential for smoke damage to equipment not already failed by fire affects has been added to Fire Scenario Report.
FSS-E3	Statistical analysis of parameters used for modeling fire scenarios	Cat 1 is acceptable for the application; parameters for modeling fire scenarios (such as heat release rates and severity factors) taken from NUREG/CR-6850 which is the consensus method for FPRA development.
FSS-F2	Establish and justify criteria for structural collapse for selected scenarios	Not applicable; no scenarios were selected for quantification (structural steel damage required no further quantitative treatment).
FSS-F3	Quantify risk associated with selected structural collapse scenarios	Not applicable; no scenarios were selected for quantification (structural steel damage required no further quantitative treatment).

Table 4-2 ONS Fire PRA NRC Review – Category I Summary

SR	Topic	Status
FSS-H6	Document technical basis for statistical models applied in the analysis	Conservative scoping fire modeling criteria based on NUREG/CR-6850 which is the consensus method for FPRA development; other than the Bayesian update of fire frequencies and consideration of actual fire brigade response times, no plant specific updates were applied.
QNS-C1	Verify quantitative screening process does not screen high risk fire areas based on 1% (Cat 3) or 10% (Cat 2) CDF/LERF contribution criteria	Not applicable. The screening criteria was not applied; if a building or structure (or an area in the case of the switchyard) contained PRA credited equipment/cables and/or could result in loss of offsite power, it was not screened.

RAI 5-5:

NFPA 805, §2.4.3.1 requires comparison of risk to be done in CDF and LERF. Attachment W of the TR provides overall risk insights and risk change due to NFPA 805 transition. The risk insights are provided in delta CDF, but not delta LERF. Provide LERF values similar to those provided for CDF (e.g., pre-transition, post-transition, change due to fire risk, and change due to internal events risk).

RAI 5-5 RESPONSE:

Please note that this RAI response is preliminary and will be finalized when ONS Calculation OSC-9518, NFPA 805 FPRA Application, is completed.

Both CDF and LERF were considered in the evaluation of fire risk. However, since ONS calculation, OSC-9518, NFPA 805 FPRA Application was made available for NRC review, Duke sought to minimize the level of detail and opted not to include LERF results within the LAR. Only one of the risk significant scenarios based on CDF involve containment bypass and none were within the top 50% of CDF contributors, therefore, few additional insights can be gained by inclusion of the corresponding LERF scenarios. ONS calculation, OSC-9518, contains a listing of the top LERF scenarios. While there are some containment bypass scenarios that contribute to overall LERF risk that are less significant when considering CDF alone, the overall risk results are dominated by CDF.

Accordingly, paragraph W.2 of the Transition Report may be more appropriately worded as follows: *In accordance with the guidance in Regulatory Position C.2.2 of RG 1.205, Revision 0, the total risk increase associated with pre-transition FP program variances that will meet the NFPA 805 performance-based approach (via the Change Evaluation Process) resulted in a collective risk increase of less than 2E-05/year for CDF and 3.5E-07/year for LERF. As allowed by RG 1.205, credit for non-fire related modifications that affect the FPRA results has been* *calculated to be over 3E-05/year and for CDF and 7.5E-07/year for LERF, respectively, resulting in a total risk decrease associated with transition of approximately 1E-05/year (CDF) and 4E-07/year (LERF). It is important to note that the risk reduction is based solely on the scope of fire initiating events. Any additional risk reductions that may result from the internal events PRA have not been included. This change is compared to the total baseline fire risk of ~5E-05/year (CDF) and ~1.65E-06/year (LERF). Therefore these changes are allowable per RG 1.174.*

An excerpt from an in-process preliminary revision of ONS calculation, OSC-9518, NFPA 805 FPRA Application, has been included below. These results are subject to change as the FPRA calculation revisions initiated to address recent changes in the list of committed modifications undergo final approval.

3.1 LERF CONSIDERATIONS

Only one of the risk significant scenarios tabulated above for CDF involve containment bypass and none were within the top 50% of CDF contributors, therefore, few additional insights can be gained from the corresponding LERF scenarios as demonstrated below.

LERF Results Corresponding to Top Risk Contributing Scenarios

Scenario	Description	LERF	CDF	LERF/CDF
U3TB29B1	3TC Switchgear - Severe Fire	6.43E-08	5.66E-06	0.01
CR03W1	MCB Fire - CR Abandonment	6.45E-08	5.29E-06	0.01
AB101B1	Aux BchBd 3AT1 Fire	8.71E-08	4.82E-06	0.02
AB101E	Cabinet EPSL (LS) Fire	5.05E-08	4.07E-06	0.01
AB101B2	Aux BchBd 3AT2 Fire	4.38E-08	3.34E-06	0.01
CR03F	MCB Fire - 3AB1 (MFB 1 & 2)	2.60E-08	2.21E-06	0.01
CR03E	MCB Fire - 3UB2 with HPI & PORV	2.04E-08	1.84E-06	0.01
AB098B	AHU 3-6 Fire	1.27E-08	1.26E-06	0.01
CR03H	MCB Fire - 3EB1 thru 3EB4	1.36E-08	1.10E-06	0.01
AB089C1	LC 3X9 Severe Fire	1.03E-08	8.96E-07	0.01
U2TB15B1	Main FDW Pump 2A Fire - Non Suppression	1.31E-08	8.72E-07	0.02
U2TB15C1	Main FDW Pump 2B Fire - Non Suppression	1.31E-08	8.72E-07	0.02

LERF Results Corresponding to Top Risk Contributing Scenarios

Scenario	Description	LERF	CDF	LERF/CDF
U1TB19C1	Main FDW Pump 1B Fire - Non Suppression	1.31E-08	8.72E-07	0.02
AB089B2	LC 3X8 Non-Severe Fire	8.52E-09	8.20E-07	0.01
RB03D	RB Fire - West Half	7.99E-07	7.96E-07	1.00

It is noted that the top LERF scenarios do not correspond directly to the top CDF scenarios. There are some containment bypass scenarios that contribute to overall LERF risk that are less significant when considering CDF alone. The significant LERF scenarios are tabulated in descending order (or ascending LERF Rank). The LERF Rank and CDF Rank demonstrate that each of the scenarios is important to CDF and LERF, but the order of their significance varies.

Scenario	Description	LERF Rank	CDF Rank	Unique LERF Insights
RB03D	RB Fire - West Half	1	15	Seal return pathway via HP-20 and HP-21 (see below)
AB101X4	Transient Fire (Hotwork) near Cable Shaft	2	9	LDST pathway via HP-3/4 and HP-5 and seal return pathway via HP-20 and HP-21 (see below)
AB101B1	Aux BchBd 3AT1 Fire	3	3	None; no containment bypass; see CDF insights above
CR03W1	MCB Fire - CR Abandonment	4	2	None; no containment bypass; see CDF insights above
U3TB29B1	3TC Switchgear - Severe Fire	5	1	None; no containment bypass; see CDF insights above
AB089Y1	Transient Fire at Column N88	6	10	Containment bypass scenario: Letdown pathway via ES failure due to loss of 3KVIA & KVIB power
AB101E	Cabinet EPSL (LS) Fire	7	4	None; no containment bypass; see CDF insights above
AB101B2	Aux BchBd 3AT2 Fire	8	5	None; no containment bypass; see CDF insights above
CR03F	MCB Fire - 3AB1 (MFB 1 & 2)	9	6	None; no containment bypass; see CDF insights above

An important insight gained from the LERF analysis is that a fire in the reactor building appears to result in failure of both the inboard isolation valves (located in containment) and the outboard isolation valves (located in the East Pen Room) in both the seal return line and the letdown line. The seal return line is isolated by normally open inboard motor operated valve (MOV) HP-20 (fails as is) and normally open outboard air operated valve (AOV) HP-21 (fails closed on loss of air and fails open on loss of power). The letdown return is isolated by normally open inboard MOVs HP-3 and HP-4 (both fail as is) and normally open outboard AOV HP-5 (fails closed on loss of air or power). ARTRAK data indicated that cables affecting both of the outboard valves (HP-5 and HP-21) could fail given a fire in containment. However, in the case of HP-5, the valve would fail closed on loss of air or power so the LERF results for a fire in containment are conservative (flagged for refinement in a future revision). In the case of HP-21, there is the potential to lose power resulting in the failure of the active function to close even for a fire in containment. Therefore, neither of these paths represent a multiple spurious operation (MSO) scenario (one is not real and the other is failure of the active function to close).

RAI 5-6:

Provide justification for only providing listings for fire initiating events representing 50% of CDF in Table W-1 of the TR or justify a level of additional initiators that contribute to risk (CDF and LERF).

RAI 5-6 RESPONSE:

Since there are no existing regulatory requirements or guidance on providing fire risk insights in the LAR TR, a decision was made to provide fire initiating events representing 50% of CDF in Table W-1 of the TR. The 50% threshold decision did not have a documented technical basis and was based on judgment. A revision to ONS calculation OSC-9518, NFPA 805 FPRA Application Calculation, is in process to address the compilation of insights from scenarios that are significant contributors to risk. According to the definition of "significant" in Section 1.2-2 of the ASME Standard relative to CDF (ASME-ANS RA-S-2008), the fire initiating events that sum to 95% of CDF collectively or those that contribute more than 1% of the total fire CDF are considered to represent the significant fire scenarios. There are 50 scenarios comprising 90% of the cumulative fire CDF. Of these, 31 scenarios contribute less than 1% on an individual basis. Therefore, the scenarios comprising 90% of CDF are considered sufficient for developing insights.

The updated table from a preliminary revision of ONS calculation OSC-9518 has been attached. These results are subject to change as the FPRA calculation revisions initiated to address recent changes in the list of committed modifications undergo final approval. As discussed in response to RAI 5-5, insights from the LERF results are also included in the update to ONS calculation OSC-9518.

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
U3TB29B1	3TC Switchgear - Severe Fire	11.4% / 11.4%	In addition to the loss of 3TC, tray damage results in failure to trip the turbine causing overcooling and HPI actuation; operators fail to throttle HPI which leads to a stuck open PORV sequence. Additional contributing sequences involve transient induced spurious opening of the PORV, tray damage resulting in turbine driven EFW pump failure, and loss of all HPI due to loss of power and LPSW failures. Note that 3TD has been excluded from failure via tray damage by crediting reroute of cables 3EPTD105 & 3EPTD1404; without the reroute, failure of 3TD is offset by the planned installation of MSIVs.	1.4E-02	4.1E-04	5.7E-06
CR03W1	MCB Fire - CR Abandonment	10.7% / 22.1%	Scenario takes no credit for Aux Shutdown Panel; failures were assumed which virtually eliminated all success paths other than SSF mitigation. Additionally, due to failure of HPI coupled with spurious operation of the PORV and failure of the PORV block, SSF does not prevent core damage for the controlling scenario.	1.7E-01	3.2E-05	5.3E-06
AB101B1	Aux BchBd 3AT1 Fire	9.7% / 31.8%	Internal cabinet failures include MFB 1 & 2. Other than the additional power source for the SSF, PSW is also assumed to be failed. The risk is dominated primarily by seal LOCA sequences with additional contribution from stuck open PORV sequences. Seal LOCA results from the loss of seal cooling (due to loss of power) and spurious RCP operation. The stuck open PORV sequences result from loss of HPI & EFW (due to loss of power) with random failure of the PORV block valve to close.	1.2E-01	4.0E-05	4.8E-06
AB101E	Cabinet EPSL (LS) Fire	8.2% / 40%	Internal cabinet failures include MFB 1 & 2. Other than the additional power source for the SSF, PSW is also assumed to be failed. The risk is dominated by seal LOCA sequences. Seal LOCA results from the loss of seal cooling (due to loss of power from the fire-induced failure of the N1 & N2 breakers to open) along with random failure(s) of the SSF; the PRA assumes a 20% chance that an RCP seal LOCA (medium) will occur.	5.0E-02	8.1E-05	4.1E-06

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	Description	Contribution[b]	Risk insights	CCDP	IF[a]	CDF
AB101B2	Aux BchBd 3AT2 Fire	6.7% / 46.7%	Internal cabinet failures include MFB 1 & 2. Other than the additional power source for the SSF, PSW is also assumed to be failed. The risk is dominated by seal LOCA and stuck open PORV sequences. Seal LOCA results from the loss of seal cooling (due to loss of power) along with random failure(s) of the SSF; the PRA assumes a 20% chance that an RCP seal LOCA (medium) will occur. The stuck open PORV sequences result from loss of HPI & EFW (due to loss of power and fire-induced failure of 3V-186) with random failure of the PORV block valve to close.	8.3E-02	4.0E-05	3.3E-06
CR03F	MCB Fire - 3AB1 (MFB 1 & 2)	4.5% / 51.2%	Internal cabinet failures include 3TC, 3TD, & 3TE, and 3MS VA0126 & 129. Other than the additional power source for the SSF, PSW is also assumed to be failed. Loss of HPI coupled with loss of CC to the thermal barrier (due to loss of power) and spurious RCP operation along with random failure(s) of the SSF results in a loss of RCP seal cooling; the PRA assumes a 20% chance that an RCP seal LOCA (medium) will occur.	1.1E-01	2.1E-05	2.2E-06
CR03E	MCB Fire - 3UB2 with HPI & PORV	3.7% / 54.9%	This scenario is dominated by fire induced loss of coolant accidents (open RV head vents or pressurizer PORV) and failure of LPI (sump recirculation). Transient (loss of condenser vacuum) induced stuck open pressurizer safety valves with failure to achieve sump recirculation make a small contribution.	2.62E-01	7.02E-06	1.84E-06
AB098B	AHU 3-6 Fire	2.5% / 57.4%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of RCP seal cooling, spurious RCP start, and loss of the RBCUs. Unavailability of the SSF, leads to a consequential RCP seal LOCA. Mitigation fails due to various sump recirculation failures.	7.62E-03	1.65E-04	1.26E-06

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
CR03H	MCB Fire - 3EB1 thru 3EB4	2.2% / 59.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power and PSW supply to HPI. Unavailability of the SSF ASW and the TDEFW pump leads to an unmitigated loss of SSHR. Failure of the SSF RCM with a consequential seal LOCA makes a smaller but meaningful contribution.	5.28E-02	2.08E-05	1.10E-06
AB089C1	LC 3X9 Severe Fire	1.8% / 61.5%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, draining of the BWST, loss of LPSW, instrument air, EFW, PSW, and the PORV block valve. Failure of EFW cross-connect and SSF ASW leads to a total loss of SSHR. Stuck open pressurizer PORV sequences are also important. Mitigation fails due to the loss of ECCS pumps as a consequence of the draining.	8.78E-03	1.02E-04	8.96E-07
U2TB15B1	Main FDW Pump 2A Fire - Non Suppression	1.8% / 63.2%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, the TDEFW pump and EFW cross-connect. Failure of the SSF and PSW leads to a loss of all SSHR, RCP seal LOCA sequences make a small contribution. Mitigation fails due to the loss of power.	5.96E-02	1.46E-05	8.72E-07
U2TB15C1	Main FDW Pump 2B Fire - Non Suppression	1.8% / 65%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, the TDEFW pump and EFW cross-connect. Failure of the SSF and PSW leads to a loss of all SSHR, RCP seal LOCA sequences make a small contribution. Mitigation fails due to the loss of power.	5.96E-02	1.46E-05	8.72E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
U1TB19C1	Main FDW Pump 1B Fire - Non Suppression	1.8% / 66.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, the TDEFW pump and EFW cross-connect. Failure of the SSF and PSW leads to a loss of all SSHR, RCP seal LOCA sequences make a small contribution. Mitigation fails due to the loss of power.	5.96E-02	1.46E-05	8.72E-07
AB089B2	LC 3X8 Non-Severe Fire	1.7% / 68.4%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, spurious draining of the BWST, loss of various ECCS valves, the PORV block valve, and a loss of instrument air. Failure of EFW and PSW lead to a stuck open pressurizer PORV. Mitigation fails due to the BWST draining and consequential loss of the ECCS pumps.	3.23E-03	2.54E-04	8.20E-07
RB03D	RB Fire - West Half	1.6% / 70%	This scenario is dominated by a fire induced reactor trip initiating event, fire induced spurious pressurizer spray, loss of the pressurizer heaters, loss of the SSF RCM, loss of RCP thermal barrier cooling, and inability to isolate the seal return line. Failure of the LDST level transmitters results in a loss of RCP seal injection. An ISLOCA through the seal return line results (significant LERF contributor). Other important sequences include fire induced pressurizer PORV opening or transient induced stuck open PORV with failure of sump recirculation.	1.91E-03	4.17E-04	7.96E-07
CR03D	MCB Fire - 3UB1 with LPI (no EFDW impact)	1.6% / 71.6%	This scenario is dominated by fire induced opening of the pressurizer PORV or transient induced stuck open PORV, loss of the pressurizer PORV block valve, failure of the LDST level transmitters, and LPI valves. The result is a fire induced LOCA with failure to achieve sump recirculation. Transient (loss of condenser vacuum) loss of SSHR with failure to achieve sump recirculation is also important.	1.13E-01	6.99E-06	7.90E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
CR03G2	MCB Fire - 3VB2 & 3VB3	1.5% / 73.1%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, fire induced draining of the BWST, failure of the turbine trip, loss of the RBCUs, spurious RCP restart, and loss of the motor driven EFW pumps. Overcooling transients with failure to throttle HPI result in a stuck open pressurizer relief valve that can not be mitigated. Failure of the TDEFW and SSF ASW leads to a loss of all SSHR or stuck open pressurizer relief valve that can not be mitigated.	3.56E-02	2.08E-05	7.41E-07
CR03J	ICS Cabinet Fire	1.1% / 74.1%	This scenario is dominated by a fire induced loss of MFW initiating event, fire induced loss of LPSW, RCP seal injection, PSW, and spurious RCP restart. Failure to activate the SSF RCM leads to an RCP seal LOCA, failure of EFW and the SSF ASW result in a loss of all SSHR. Failure to cross-connect LPSW leads to a loss of sump recirculation capability.	9.28E-04	5.65E-04	5.24E-07
AB101H	Cabinet SGLC-3 Fire	1.0% / 75.1%	This scenario is dominated by a fire induced loss of MFW initiating event, fire induced loss of instrument air, the EFW pumps, the RBCUs, and PSW. Failure of the EFW cross-connect and the SSF ASW leads to loss of SSHR. Various failures lead to loss of sump recirculation.	4.07E-03	1.21E-04	4.92E-07
AB101B3	Aux BchBd 3AT3 Fire	0.9% / 76.0%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, fire induced loss of LPSW, the RBCUs, PSW,TDEFW pump, and a failure to trip the turbine. Failure of the EFW and LPSW cross-connects, and SSF ASW leads to a loss of all SSHR. Mitigation fails at sump recirculation due to the loss of LPSW. An overcooling transient which initiates HPI and leads to a stuck open PZR safety valve makes a smaller but important contribution.	1.12E-02	4.03E-05	4.51E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	Description	Contribution[b]	Risk insights	CCDP	IF[a]	CDF
U3TB29B2	3TC Switchgear - Non Severe	0.9% / 76.9%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, fire induced loss of LPSW partly due to the loss of power, and loss of instrument air. Failure of the LPSW and EFW cross-connects, failure of the TDEFW pump, SSF ASW, and PSW lead to a loss of SSHR or stuck open PORV. Mitigation fails at sump recirculation. An overcooling transient which initiates HPI and leads to a stuck open PORV also make a contribution.	9.27E-04	4.84E-04	4.49E-07
AB101K1	Keowee ES Ch A Cabinet Fire	0.9% / 77.8%	This scenario is dominated by a fire induced loss of MFW initiating event, fire induced loss of LPSW, PSW, instrument air, and failure of the turbine to trip. Failure of the LPSW cross-connect, failure of EFW cross-connect, and SSF ASW leads to a loss of SSHR. Mitigation fails at sump recirculation.	1.08E-02	4.04E-05	4.36E-07
CR03W2	Non MCB Fire - CR Abandonment	0.9% / 78.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, fire induced loss of all ac power, EFW, and PSW. Mitigation fails due to SSF failure.	1.21E-01	3.58E-06	4.33E-07
U3TB28B	MCC 3XGA Fire	0.8% / 79.5%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, fire induced loss of LPSW, EFW, and instrument air. Failure to cross-connect EFW, of SSF ASW, and PSW lead to a loss of SSHR. Mitigation fails at sump recirculation with failure to cross-connect LPSW.	5.71E-04	7.27E-04	4.15E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
U3TB06C3	Main FDW Pump 3B Fire - Severe	0.8% / 80.3%	This scenario is dominated by a fire induced loss of main feedwater initiating event, loss of all ac power, and EFW. Failure of the SSF and PSW leads to an unmitigated loss of all SSHR.	6.84E-02	5.56E-06	3.80E-07
AB101T1	3ESTC1 Cabinet Fire	0.7% / 81.0%	This scenario is dominated by a fire induced inadvertent engineered safeguards initiating event, loss of LPSW, instrument air, and the RBCUs. Failure to throttle HPI results in a stuck open pressurizer relief valve. Failure to cross-connect LPSW fail mitigation.	8.87E-03	4.04E-05	3.58E-07
U2TB15C3	Main FDW Pump 2B Fire - Severe	0.7% / 81.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, and EFW. Failure of the SSF and PSW leads to a loss of all SSHR. RCP seal LOCAs also make a small contribution.	5.96E-02	5.55E-06	3.31E-07
U2TB15B3	Main FDW Pump 2A Fire - Severe	0.7% / 82.3%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, and EFW. Failure of the SSF and PSW leads to a loss of all SSHR. RCP seal LOCAs also make a small contribution.	5.96E-02	5.55E-06	3.31E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
U1TB19C3	Main FDW Pump 1B Fire - Severe	0.7% / 83.0%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of all ac power, and EFW. Failure of the SSF and PSW leads to a loss of all SSHR. RCP seal LOCAs also make a small contribution.	5.96E-02	5.55E-06	3.31E-07
AB089B1	LC 3X8 Severe Fire	0.7% / 83.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of instrument air, and draining of the BWST to the sump. Failure of a pressurizer relief valve to reseat leads to a transient induced LOCA that can not be mitigated.	3.24E-03	1.02E-04	3.31E-07
CR03G1	MCB Fire - 3VB1	0.5% / 84.2%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of the pressurizer heaters, and inability to establish high pressure recirculation. Failure of EFW and the SSF ASW leads to a loss of SSHR and mitigation fails at sump recirculation. Overcooling transients with a stuck open pressurizer relief valve make a small contribution.	3.73E-02	7.00E-06	2.61E-07
U3TB06B	Main FDW Pump 3A Fire - Severe	0.5% / 84.7%	This scenario is dominated by a fire induced loss of main feedwater initiating event, and EFW. Failure of the SSF and PSW leads to a loss of all SSHR. Failure to establish sump recirculation results in failure to mitigate the event.	8.85E-04	2.93E-04	2.59E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
AB101C5	Cabinet UCTC 5 Fire	0.5% / 85.2%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of 3TC and 3TD, and partial loss of instrument air. A stuck open pressurizer relief valve due to failure of EFW or an overcooling event with unthrottled HPI results. Mitigation fails at sump recirculation due to the power failures.	5.88E-03	4.05E-05	2.38E-07
U3TB38C1	T/G Hydrogen Severe Fire	0.4% / 85.6%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of EFW, and loss of instrument air. Failure of the EFW cross-connect, the SSF, and PSW result in a loss of SSHR. Failure to initiate high pressure recirculation fails mitigation. Stuck open pressurizer relief valves make a small contribution.	1.95E-04	1.10E-03	2.15E-07
AB101V	CRD V Reg / Xfmr Fire	0.4% / 86.1%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of the pressurizer heaters, RCP seal injection, and loss of the RCBUs. The sequences are split between overcooling events which result in unthrottled HPI and stuck open pressurizer relief valves and seal LOCAs. Mitigation fails due to failure to successfully establish sump recirculation.	3.34E-04	6.26E-04	2.09E-07
AB101C6	Cabinet UCTC 6 Fire	0.4% / 86.5%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of the RBCUs, and failure of the turbine to trip. The overcooling with failure to throttle HPI leads to a stuck open pressurizer relief valve. Mitigation fails due to failure to successfully establish sump recirculation.	5.10E-03	4.04E-05	2.06E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
AB106D	Closed Panel 1EPSLP2 Fire	0.3% / 86.8%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, and failure of both standby buses. Transformer CT3 is unavailable and a loss of all ac power results. Failure of EFW and the SSF ASW leads to an unmitigated loss of all SSHR. Failure of the SSF RCM leads to a seal LOCA.	3.47E-04	4.84E-04	1.68E-07
RB03E	RB Fire - East Half	0.3% / 87.1%	This scenario is dominated by a fire induced reactor trip initiating event, loss of the pressurizer heaters, loss of RCP thermal barrier cooling, spurious RCP start, and loss of the SSF RCM pump. Common cause failure of the N breakers or loss of seal injection with various sump recirculation failures lead to failure to mitigate a seal LOCA.	1.78E-04	8.99E-04	1.60E-07
AB101R1	EHC Cabinet - Severe Fire	0.3% / 87.5%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of the 4kV switchgear, loss of the pressurizer heaters. loss of TDEFW, and PSW. A stuck open PORV or failure of the SSF results in failure to mitigate the loss of power.	2.87E-01	5.54E-07	1.59E-07
AB105E	Closed Panel 2EPSLP2 Fire	0.3% / 87.8%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of instrument air, failure of both standby buses, and failure of the PSW power supply to HPI. Transformer CT3 is unavailable and a loss of all ac power results. Failure of EFW and the SSF ASW leads to an unmitigated loss of all SSHR. Failure of the SSF RCM leads to a seal LOCA.	3.47E-04	4.44E-04	1.54E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
U3TB07C	Hydrogen Seal Oil Skid Fire	0.3% / 88.1%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of the MDEFW pumps, and the EFW cross-connect. Failure of the TDEFW, SSF ASW, and PSW leads to loss of all SSHR. Failure to establish sump recirculation fails the mitigation.	5.27E-05	2.85E-03	1.50E-07
AB101X5	Transient Fire (Hotwork) near Covered Hatch	0.2% / 88.3%	This scenario is dominated by a spurious pressurizer PORV opening event, loss of 3TC and 3TD, HP-24, and BWST draining. Draining of the BWST or power failures lead to failure to mitigate the open PORV event.	4.45E-01	2.79E-07	1.24E-07
AB101M3	Cabinet 3MTC3 Fire	0.2% / 88.6%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of EFW, loss of the pressurizer PORV, PSW, instrument air and the RBCUs. Failure of the EFW cross-connect and SSF ASW leads to loss of all SSHR. Failure to establish sump recirculation fails the mitigation. Stuck open pressurizer safety valves also make a contribution.	2.96E-03	4.05E-05	1.20E-07
U3TB29C2	3TD Switchgear - Non Severe	0.2% / 88.8%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of 3TD, loss of LPSW, failure of the turbine to trip, and loss of instrument air. Failure to throttle HPI on the overcooling leads to a stuck open pressurizer relief valve. Various sump recirculation failures result in the mitigation failure. Loss of SSHR sequences make up a smaller but meaningful contribution.	2.17E-04	5.53E-04	1.20E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
CR12H	ES Cabinet Fire (LPSW)	0.2% / 89.0%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of CT4 and CT5, loss of the PSW, and loss of instrument air. Unavailability of CT3 leads to a loss of all ac power. Failure of EFW and the SSF ASW leads to a unmitigated loss of SSHR. Seal LOCA sequences make a smaller but important contribution.	3.40E-04	3.24E-04	1.10E-07
AB089C2	LC 3X9 Non-Severe Fire	0.2% / 89.2%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, draining of the BWST, loss of the PSW, and loss of instrument air. Failure of EFW and the SSF ASW leads to a unmitigated loss of SSHR.	4.26E-04	2.54E-04	1.08E-07
U3TB38B	T/G Exciter Fire	0.2% / 89.5%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of instrument air, and failure of the turbine to trip. The overcooling with failure to throttle HPI leads to a stuck open pressurizer relief valve. Mitigation fails due to failure to successfully establish sump recirculation.	2.72E-05	3.90E-03	1.06E-07
AB101B4	Aux BchBd 3AT4 Fire	0.2% / 89.7%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of EFW, loss of the pressurizer heaters, PSW, and the RBCUs. Failure of the EFW cross-connect and SSF ASW leads to loss of all SSHR. Failure to establish sump recirculation fails the mitigation. Stuck open pressurizer safety valves also make a contribution.	2.59E-03	4.05E-05	1.05E-07

ONS Significant Fire Initiating Events (Representing 90% of the Calculated Fire Risk)						
Scenario	**Description**	**Contribution[b]**	**Risk insights**	**CCDP**	**IF[a]**	**CDF**
AB089D	Control Battery Charger 3CA Fire	0.2% / 89.9%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, draining of the BWST, loss of the PSW, and loss of instrument air. Failure of EFW and the SSF ASW leads to a unmitigated loss of SSHR.	4.39E-04	2.35E-04	1.03E-07
U3TB29C1	3TD Switchgear - Severe Fire	0.2% / 90.1%	This scenario is dominated by a fire induced loss of condenser vacuum initiating event, loss of 3TD, loss of LPSW, failure of the turbine to trip, and loss of instrument air. Failure to throttle HPI on the overcooling leads to a stuck open pressurizer relief valve. Various sump recirculation failures result in the mitigation failure. Loss of SSHR sequences make up a smaller but meaningful contribution.	2.20E-04	4.64E-04	1.02E-07

a. Ignition Frequency (IF) includes severity factor and probability of non suppression, where applicable.

b. Individual contribution followed by cumulative contribution.

RAI 5-7:

Document and justify Fire PRA related analysis approaches and values that were not performed in accordance with NUREG/CR-6850. The LAR submittal letter states that the "PRA was developed in accordance with NUREG/CR-6850/EPRI TR-1011989 and is discussed in Enclosure 2 section 4.5." TR Section 4.5.1 further states that the "technical specifics for some elements of NUREG/CR-6850/EPRI TR-1011989 were altered or modified via the FAQ process." During the audit, reviewers found several undocumented variances from the methods prescribed in NUREG/CR 6850 and FAQs. It is unclear why the analysis deviated from the approach and it is unlikely the auditors found all the circumstances where the approach in NUREG/CR 6850 was not used. A few examples include:

- Fire Compartments as used in the ONS Fire PRA are different from those defined in NUREG/CR 6850.
- Incipient detection is credited with a factor of 100 decrease of the NUREG/CR 6850.
- The hot-work non-suppression probability was changed from the NUREG/CR 6850 value of 0.38 to 0.01.

RAI 5-7 RESPONSE:

This response summarizes the alterations of the methods in NUREG/CR-6850 that were applied in the development of the ONS FPRA in an attempt to obtain more realistic results. The analysis methods are described in the ONS FPRA documentation and stand alone justification is provided within the FPRA documentation file to support instances where NUREG/CR-6850 was not the sole method used. Reliance on guidance and/or clarification provided in approved FAQs is not considered a departure from NUREG/CR-6850. Accordingly, the NUREG/CR-6850 departures may be grouped into 2 categories:

1) Alterations to treatment methods where CR-6850 lacked sufficient guidance including apparent departures from CR-6850 definitions that were taken for convenience, but had no impact on the calculated risk.
2) Alterations to treatment methods to reduce conservatism in the CR-6850 guidance and therefore had a risk reduction impact. It should be noted that some departures from CR-6850 are defensible as risk reduction measures and may have been discussed in the FPRA documentation, but have not been credited in the ONS FPRA.

Each specific alteration is discussed below. A cross reference to the location in the PRA documentation where this method is discussed is also provided.

Alterations with No Impact on Calculated Risk

1. Fire Compartments: The ONS FPRA used fire zones to take advantage of the available cable resolution provided in ARTRAK. Oconee has relatively few fire areas with defined boundaries that meet the NUREG/CR-6850 definition of an enclosed compartment. In

fact, many of the Oconee fire zones have fewer than four walls. However, the scenario development approach did not limit the zone of influence or target damage set based on zone boundaries. Fire zones also provided a convenient grouping for multiple fire scenarios. There are over 500 fire scenarios in the BOP fire area. Grouping the BOP fire area into smaller fire compartments (based on fire zones) for analysis purposes introduced significant improvement in overall analysis efficiency. This is discussed in the FPRA scenario development documentation.

2. HEP Adjustments: NUREG/CR-6850 provided a screening approach for evaluating human failure events and indicated that more detailed best estimate analysis of human actions will be needed to obtain more realistic results. The ONS FPRA applied a simplified approach for adjusting HEP values that was presented to the NRC during the pilot meetings. This is discussed in the FPRA model development documentation.

Alterations to Reduce Conservatism

1. MFW Pumps: The MFW pump generic ignition frequency was refined with a zero events approximation technique. This refinement was needed to address the lack of data to support severity factors presented in CR-6850. This method is supported by preliminary FAQ 08-044 and EPRI Interim Report No. 1016735, *Fire PRA Methods Enhancements*. A scenario frequency reduction factor of 0.045 was applied to the MFW pump fire scenarios. This 0.045 factor is consistent with FAQ 08-044. The EPRI report suggested using a factor of 0.033. This is discussed in the FPRA scenario development documentation.

2. RCP Fires: The RCP generic ignition frequency was refined with a zero events approximation technique. This refinement was needed to account for the fact that none of the RCP fires in the fire events database involved any substantive volume of oil or significant damage beyond the ignition source. This method, which applies a 0.09 frequency reduction factor, utilizes the same approach applied in preliminary FAQ 08-044. This is discussed in the FPRA scenario development documentation.

3. Ventilated Cabinets: risk reduction factors developed for ventilated cabinets effectively reduce the frequency of fire propagation (tray damage) but do not reduce the cabinet fire frequency. Since the risk reduction factor does not impact the frequency of internal fire induced cabinet failures, the credit for these factors is highest when there are no credited internal cabinet failures. In other words, the risk reduction factor cannot simply be applied to the scenario result in determining the amount of risk reduction "credit" realized from the treatment.

    a. Low Energy: For severe fires, a zero event approximation technique was applied to ventilated cabinets in the Cable Room to account for the low likelihood of external fire damage. Most of the electrical cabinets in the Cable Room house low energy circuits (control and instrumentation). The lack of any highly energetic sources and the general behavior of electrical based fires all suggest that any realistic (credible) fire event would tend to be characterized by smoldering with relatively low heat release rates. These fire events would not be expected to involve rapid flame development. This characterization is consistent with insights gained by reviewing the EPRI Fire Events Database (FEDB). A review of the FEDB found that none of the events involved fire propagation beyond the boundary of an electrical cabinet of the type and voltage class that exists in the Cable Room. This approach, which involved application of a 0.00459 adjustment factor for the severe fire scenario, cannot be combined with credit for suppression.

b. Manual Suppression: Credit for detection and suppression for ventilated cabinet fire scenarios not subjected to the low energy cabinet treatment can also be justified. The onset of fire in a cabinet will likely be detected before it propagates beyond the cabinet. Therefore, for ventilated cabinets that are located in zones equipped with automatic detection, credit may be taken for suppression by the fire brigade prior to sustaining external target damage. An adjustment factor of 0.0365 would be applicable for the severe fire (tray impact) scenario. This approach was only partially applied to the Inverter and Isolating Transfer Diode Circuit cabinets in the Equipment Room. The applicable scenarios analyzed 3 cabinets together. The severe fire involved all 3 cabinets resulting in tray damage with a severity factor of 0.2. The non-severe fire was assumed to remain in the cabinet. This partial application of the full treatment constituted a deviation since the non-severe fire would also result in tray damage using CR-6850 assumptions.

Both of these items are discussed in the FPRA scenario development documentation. The FPRA scenario development documentation does not explicitly state that a severity factor of .2 was going to be used instead of the .0365; however, the FPRA summary report shows that scenarios for the inverter and isolating transfer diode circuit cabinets in the Equipment Room use the .2 factor.

4. Hot Work Transients: In the area of welding and cutting transients, a non-suppression probability factor of 0.01 was applied. This factor takes credit for prompt suppression and the procedural non-compliance required in order for target damage to occur during hot work activities; CR-6850 allows a 0.38 factor for prompt suppression for hotwork transient scenarios. This approach is more consistent with today's standards governing hot work activities. Accordingly, this treatment reduced the hot work transient frequency for each applicable scenario by a factor of 2.63E-02. Given that ONS utilizes armored cables, the likelihood of significant cable damage occurring from hotwork activities, which are conducted under fire watch, is much less than prescribed by the CR-6850 treatment. This is discussed in the FPRA Scenario Development documentation.

5. General Transients: New severity factors, taking into consideration plant location, were applied to the generic fire frequency for General Transients for better alignment with industry fire experience (no consequential fire damage). Factors of 8E-02, 5E-02, & 4E-02 were applied to the general transient frequency for transient scenarios in the Auxiliary

Building, Turbine Building, and Plant Wide fire compartments, respectively, to provide a more realistic accounting of the general transient fires that result in damage beyond the transient ignition source. This is discussed in the FPRA Scenario Development documentation.

6. Transient Heat Release Rate (HRR): The 75th percentile HRR was used for all transient fires. The HRR bounds all transient ignition source fire tests identified in CR-6850, Appendix G, Table G-7, with the exception of tests involving wood (untreated wood is typically prohibited at Nuclear Power Plants), airline trash bags with over 2 kg of paper products (such a large quantity of paper products is not likely to be present at one location near an ignition source) or over 4 kg of straw/grass/eucalyptus duff (such large quantities of combustible plant matter is unlikely to be present at Nuclear Power Plants). Additional refinement, which has not yet been applied at ONS, would further reduce the transient HRR to match the HRR for motor fires. Defense for the use of the 75th percentile is discussed in the FPRA Scenario Development documentation.

7. Installation of a highly sensitive incipient detection system within a cabinet may be credited to provide early warning before any loss of functions internal to the cabinet. While this approach was described in the FPRA Scenario Development documentation, Oconee has not yet committed to installation of incipient detection systems and no credit was taken for incipient detection in the FPRA results. Credit for incipient detection for risk reduction in the FPRA, including the non suppression probability of 1E-03 cited in the Fire Scenario Report, will require additional justification before it can be applied.

8. The reduced generic ignition source bin frequencies provided in EPRI Interim Report No. 1016735, Fire PRA Methods Enhancements have not been credited in the ONS FPRA. The ONS FPRA documentation does not explicitly state the fact that the reduced generic ignition source bin frequencies provided in the EPRI interim Report No. 1016735 are not used, but the generic frequencies that were used are provided in the FPRA Ignition Frequency calculation documentation.

RAI 5-8:

Provide adequate resolution of individual SR review findings and observations (F&O) for both the most recent review of the internal events PRA and the staff review of the Fire PRA. The discussions provided in the TR for many of the F&Os is not adequate to conclude if the issues have been satisfactorily resolved. ONS RAI 5-9 identifies specific internal events F&O's that require further explanation to justify that no further action is necessary, or may require modifications to the PRA. The response needs to include a detailed description of model changes made to resolve the findings and justification for how the resolutions result in the associated supporting requirement (SR) meeting the required Capability Category. Licensee needs to ensure adequate information is provided for each F&O for Staff to reach a conclusion on adequacy of the proposed/implemented resolution.

Appendices U and V to the TR identify review F&Os for the internal events and Fire PRAs. There are 70 F&O for the internal events PRA and 51 F&Os for the Fire PRA. The provided information in the appendices is inadequate to conclude if all issues have been satisfactorily resolved consistent with the guidance of RG 1.174 and RG 1.200.

RAI 5-8 RESPONSE:

This response is confined to the review findings associated with the staff review of the FPRA. The information relative to the internal events PRA is being compiled in response to RAI 5-9. With regard to the internal event F&Os, the response to ONS RAI 5-9 provides further explanation on why each corresponding ASME PRA Standard supporting requirement is adequately addressed for supporting the FPRA LAR, consistent with the guidance of RG 1.174 and RG 1.200. In addition, a meeting was held with Ray Gallucci (NRC) to provide additional information on the Appendix U items. The additional information provided to Mr. Gallucci along with the responses in RAI 5-9 should be adequate for NRC staff to conclude that the F&O resolutions are satisfactory.

The objective of Attachment V was to merely provide status of the finding. For closed findings, a reference to the supporting document was provided. The NRC was provided with the supporting documents in separate submittals. Notwithstanding, as part of the update to preliminary ONS calculation OSC-9518, NFPA 805 FPRA Application, an effort has been made to elaborate on the responses.

The following table was extracted from a preliminary revision to ONS calculation OSC-9518, NFPA 805 FPRA Application.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
CS-B1-1	Breaker coordination study incomplete (self identified)	Open	Breaker coordination issue not yet resolved; PIP O-08-2444 has been generated to track completion. The top 50% risk contributing scenarios involve loss of 4KV power and reliance on SSF mitigation which are not significantly impacted by additional failures due to improper breaker coordination.
CS-C4-1	Breaker coordination documentation	Open	This item is the documentation component of the breaker coordination issue discussed in CS-B1-1.
FQ-A2-1	Initiating events not defined for all fire scenarios	Closed	Loss of Condenser Vacuum (%T4) was our default initiator for the FPRA but some scenarios assumed a different initiator as discussed in the FPRA Model Development Report. The applied initiator has been added to information provided in quantification results summary table (FPRA Summary Report, Appendix A).
FQ-B1-1	Demonstrate convergence for selected truncation limit (see IEPRA-2 for related FPIE issue)	Closed	Reference IEPRA-2 below for closure of FPIE truncation issue. FPRA solves for CCDP (prior to application of ignition frequency) at one order of magnitude greater than FPIE. Since a typical scenario frequency is typically much less than 0.1, there is not a truncation (convergence) issue for FPRA.
FQ-C1-1	Use of nominal HEP values may result in loss of cutsets before application of recoveries/multipliers	Closed	To address the retention of additional cutsets, the HEP values were set to 0.1 for initial solve prior to application of recoveries (as described in FPRA Model Development Report).

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
FQ-E1-1	Identification of significant contributors	Closed	Risk insights from the risk significant fire initiating events with identification of significant contributors have been included in the FPRA Application Calc (see Section 3).
FQ-F1-1	Improve LERF documentation	Closed	Documentation concerns were largely confined to LERF. Accordingly, the insights section in the FPRA Summary Report was expanded to address LERF. Inconsistencies between LERF and CDF have been reconciled Additionally, the risk insights section of the FPRA Application Calc has been expanded (CDF and LERF) as described in response to RAI's 5-5 & 5-6. ;. Also, the applied initiator has been added to information provided in the quantification results summary as discussed in response to FQ-A2-1.
FSS-A5-1	Horizontal propagation for cables with PVC jackets (self identified)	Closed	PVC jacketing impact on horizontal fire spread is discussed in Fire Scenario Report. While armored cables are generally considered to be non-combustible (refer to NUREG/CR-6850 Section R.4.1.4), the armored cables at Oconee have a PVC jacketing. The justification concluded that the PVC coating will not sustain propagation of fire along the armored cable for a significant distance; any horizontal fire propagation along cable trays is adequately captured within the target set of each scenario involving overhead tray failures.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
FSS-C5-1	Potential for PVC pooling may impact assumed damage threshold	Closed	PVC pooling issue addressed in Fire Scenario Report. The justification centers on the expectation that the PVC jacket would melt and flow away creating voids for the flow of melting materials from other cables. In addition, the ridges that are characteristic of the armor jacketing provide additional free space for the flow of material. As such, it is not expected that pooling of PVC within cable trays is likely to occur even if multiple layers of armored cables exist.
FSS-D5-1	Justify use of 75% HRR for transient fires	Closed	Justification in Fire Scenario Report expanded; use of 75% HRR for transients is considered realistic treatment (more characteristic of actual transient fire scenarios identified in the fire events database) appropriate for PRA application. Use of a trash bag as the transient fuel package provides a bounding characterization of the behavior of observed transient fire ignition sources but ignores the ignition element. Consequently, while Admin controls are factored into the development of transient ignition frequencies for each compartment, the actual transient combustible loading that may be allowed or present does not directly impact the numerical results of the Fire PRA.
FSS-D6-1	Justify fire brigade response time with respect to formation of HGL	Closed	Developed support for fire brigade response time of 20 minutes from review of actual fire brigade drill performance; included in Fire Scenario Report.
FSS-G1-1	Multi-compartment analysis incomplete	Closed	Addressed via multi-compartment screening analysis added as Attachment D to Fire Scenario Report.
FSS-G2-1	Multi-compartment analysis screening criteria not defined	Closed	Screening criteria added to multi-compartment discussion in Fire Scenario Report.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
FSS-G3-1	Multi-compartment analysis screening incomplete; no MCA scenarios defined for quantification	Closed	Addressed via multi-compartment screening analysis added as Attachment D to Fire Scenario Report; screening criteria applied to all analyzed compartments with the end result being that all compartments (physical analysis units) screened.
FSS-G4-1	Multi-compartment analysis did not consider potential for barrier failure	Closed	Potential for fire barrier failure addressed via multi-compartment screening analysis added as Attachment D to Fire Scenario Report.
FSS-G5-1	Multi-compartment analysis did not assess active fire barriers	Closed	No active fire barriers have been credited in the FPRA for limiting the zone of influence; active fire barrier between BH12 and CT4 only credited for deterministic fire area boundary definition.
FSS-G6-1	Assessment of MCA scenarios relative to fire risk not performed	Closed	Addressed via multi-compartment screening analysis added as Attachment D to Fire Scenario Report; no additional scenarios identified for quantification.
FSS-H2-1	Document resolution of PVC pooling issue	Closed	Addressed in Fire Scenario report: while the PVC jacket is thermoplastic, the cable insulation within the armor is consistent with thermoset (flame retardant cross-linked polyethylene). See FSS-C5-1.
FSS-H8-1	Multi-compartment analysis documentation incomplete	Closed	Addressed via supplemental discussion and multi-compartment screening analysis added as Attachment D to Fire Scenario Report.
FSS-H9-1	Document justification for fire brigade response time	Closed	Support for fire brigade response time of 20 minutes based on review of actual fire brigade drill performance has been added to Fire Scenario Report.
HR-G7-1	Dependencies should be reviewed with respect to timing	Closed	Corrective Action 2 of PIP O-08-2915 dispositions issue as minimally conservative (compared to other sources of conservatism in FPRA) based on the degree of dependency between actions with different available response times.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
HRA-A1-1	SR considered met but documentation that safe shutdown actions carried over from FPIE model remain valid for FPRA was not provided	Closed	Discussion relative to reliance on EOP, AOP, and alarm response procedures given a fire has been added to FPRA Model Development report (see HRA-E1-1).
HRA-B2-1	HRA documentation for CASWHPIDHE and CEF0ASWDHE insufficient	Open	No impact on quantification; Corrective Action 3 of PIP O-08-2915 indicates that documentation deficiency will be addressed with issuance of future Revision 4 of the ONS PRA Model.
HRA-C1-1	Need to consider relative timing of HFE in fire scenario; time from cue versus time from fire	Closed	Risk significant HFE's revisited with respect to timing of cues; documentation provided in FPRA Model Development Report.
HRA-C1-2	Post-initiator HEP quantifications need to be checked for consistency	Closed	Top 6 risk significant operator actions in the FPRA were checked for consistent application of criteria (Corrective Action 4 of PIP O-08-2915). The following operator actions were reviewed: TTRHPITDHE was compared to CHPHPMUDHE, HHPHPR0DHE was compared to LLPLPR0DHE. And finally, FEFEFW2DHE was compared to FEFEFW1DHE. The basic events in each pair were similar in characteristics and they were accordingly mapped to the same HEP adjustment case (i.e. Case 3 or 4 etc.).
HRA-E1-1	Address how Alarm Response and EOP/AOPs are followed given a fire	Closed	Discussion relative to reliance on EOP, AOP, and alarm response procedures given a fire has been added to FPRA Model Development report to support the conclusion that credited FPRA actions are consistent with the expected plant response to a fire event including the decision to man the SSF.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
IEPRA-1	Resolve issues from gap assessment of the ONS PRA revision 3a	Closed	Appendix D to PRA Quality Self Assessment addresses ONS PRA technical adequacy for NFPA 805. Open S/Rs with potential impact on quantification of delta risk for change evaluations have been addressed in sensitivity analysis within NFPA 805 Fire PRA Application Calculation.
IEPRA-2	Demonstrate convergence for selected truncation limit (FPIE issue)	Closed	OSC-8863 demonstrates that the ONS PRA CDF results converge at 1E-09.
IGN-A5-1	Use of reactor year/critical year	Open	To be addressed when NUREG/CR-6850 is updated with the correct numbers based on reactor-year basis; impact is expected to be insignificant. Note that Interim EPRI Report 1019189, which was not used at ONS, would lower ignition frequencies for most bins. Also Bayesian update to 'lower' ignition frequencies was not applied at ONS (only 1 bin was increased). Both measures, if applied, would offset any expected increase. This open item is being tracked under PIP G-09-00698
MUD-B4-1	Procedure lacks reference to PRA combined standard (draft)	Closed	Per Corrective Action 5 of PIP O-08-2915, PRA Workplace Procedure XSAA-106 was revised to reference the Combined PRA standard.
MUD-E1-1	Qualify the FRANC computer code for use on FPRA	Closed	Per Corrective Action 6 of PIP O-08-2915, the FRANC computer code was qualified as documented in SDQA-30271-NGO.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
PP-B2-1	Justification for credit of non-rated partition boundaries insufficient	Closed	Partially addressed via multi-compartment analysis; failure to meet SR poses no adverse impact on the analysis quality or completeness. Deviation from "enclosed boundary" definition applied to compartment frequency calculation which has no impact on overall CDF/LERF results. Use of zone of influence for defining extent of fire scenario regardless of location of zone boundary ensures that scenario impacts are accurate.
PP-B3-1	Use of open fire zone boundaries implies credit for spatial separation	Closed	Partially addressed via multi-compartment analysis; failure to meet SR poses no adverse impact on the analysis quality or completeness (see disposition for PP-B2-1).
PP-C3-1	Improve general description and identification of unique FP features	Closed	Partially addressed via multi-compartment analysis (see PP-B2-1); failure to meet SR poses no adverse impact on the analysis quality or completeness.
PRM-B1-1	Impact of FPIE peer review open items on FPRA not addressed	Closed	See IEPRA-1.
PRM-D1-1	Circa 2005 fire structure in FPIE model not peer reviewed	Closed	Eliminated reliance on initiators %TBOFIRE and %CSFIRE from pre-existing fire structure in FPRA.
QLS-A3-1	Discussion in Partitioning & Ignition Frequency calc implied that actions pertaining to 4 structures that were excluded from ignition source counting had not been completed.	Closed	No impact on quantification; updated calculation to provide justification for exclusion for the 4 structures to reflect that no further action was necessary for these structures based on the multi-compartment analysis.

Table 4-1 ONS Fire PRA NRC Review – Findings

SR	Topic	Status	Response
SF-A2-1	Conduct assessment of the potential for diversion of suppression flow	Open	No impact on quantification of FPRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). See qualitative discussion of seismic fires in the Oconee Unit 3 Fire PRA Summary. PIP G-09-00698 will track the resolution of this open item.
SF-A4-1	Plant seismic response procedures do not cover seismically induced fire	Open	No impact on quantification of FPRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). See qualitative discussion of seismic fires in the Oconee Unit 3 Fire PRA Summary. PIP G-09-00698 will track the resolution of this open item.
SF-A5-1	Assessment of earthquake impact on fire brigade not documented	Open	No impact on quantification of FPRA or Change Evaluations (seismic-fire interaction is purely qualitative per NUREG/CR-6850). See qualitative discussion of seismic fires in the Oconee Unit 3 Fire PRA Summary. PIP G-09-00698 will track the resolution of this open item.
UNC-A1-1	Uncertainty and sensitivity analysis incomplete (not reviewed)	Closed	Uncertainty & Sensitivity Matrix added as Appendix D to FPRA Summary Report; sensitivity quantitatively addressed in NFPA 805 FPRA Application Calculation.

Most of the findings from the NRC review have either been addressed or deemed to have no impact on FPRA quantification. As required, a corrective action was generated to track completion of the open items. These items either relate to documentation deficiencies or final resolution of technical issues that are not expected to have a significant impact on the FPRA (i.e., breaker coordination).

RAI 5-9:

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
AS-A9	USE realistic, applicable (i.e., from similar plants) thermal hydraulic analyses to determine the accident progression parameters (e.g., timing, temperature, pressure, steam) that could potentially affect the operability of the mitigating systems. (See SC-B4)	Accident progression parameters have in some cases been determined with MAAP 3B. The next revision to the PRA is expected to use analyses performed with the most up-to-date MAAP version.	**Resolution**: None provided. **Impact:** Based on TH work performed for OPRA Revision 4 using MAAP 4, only the transient feed and bleed success criteria have a significant modification (1 HPI pump instead of the current 2).	**a)** Are the corrected success criteria included in the fire PRA? **b)** The resolution refers to the only "significant" modification. Any change to success criteria can have substantial impact of PRA results. Summarize the "TH work" in sufficient detail to describe the scope, methodology, and the results of the investigation
DA-B2	DO NOT INCLUDE outliers in the definition of a group (e.g., do not group valves that are never tested and unlikely to be operated with those that are tested or otherwise manipulated frequently)	The OSC-8796 calculation was reviewed to determine if any outlier components were inappropriately included in the established groupings. While it did not appear that outliers were included in any groups, there was no specific documentation to indicate this fact or to indicate that a conscious effort was made to ensure that outlier inclusion did not occur.	**Resolution:** Revise the data calc. to include a specific discussion of outlier treatment (i.e., do any outliers exist? If so, how are these events considered and grouped?) **Impact:** No impact is expected for documentation issues.	**c)** The Resolution indicates the evaluation should be performed which does not support the expected "no impact." Beyond assuming that this is a documentation issue, explain how the "no impact" conclusion was reached.

RAI 5-9 (continued)

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
DA-C13	DA-C13 EXAMINE coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) based on actual plant experience. CALCULATE coincident maintenance unavailabilities that reflect actual plant experience. Such coincident maintenance unavailability can arise, for example, for plant systems that have "installed spares," i.e., plant systems that have more redundancy than is addressed by tech specs. For example, the charging system in some plants has a third train that may be out of service for extended periods of time coincident with one of the other trains and yet is in compliance with tech specs.	Table 2 of OSC-8796 presents the plant-specific component specific unavailability data. Maintenance events are generally treated as independent within the PRA model. After the model is solved, cut sets involving coincident maintenance are deleted where such combinations are prohibited by the technical specifications, as documented in the model integration notebook. Cut sets involving coincident maintenance combinations prohibited by the online risk assessment tool are retained, but have their probability reduced. However, correlation among maintenance unavailabilities is not addressed.	**Resolution:** Put in place a mechanism for identifying and quantifying coincident unavailabilities. Incorporate in the system models those maintenance events allowed by technical specifications where 2 or more components have maintenance events that are correlated with each other. **Impact**: Low impact since risk significant combinations are not allowed by Tech. Specs or the Configuration Risk Management Program.	**d)** The "resolution" appears to confirm that such maintenance states exist while the "impact" simply states that they will not be risk significant. Beyond assuming that this is a documentation issue, explain which such combinations may exist and further explain how the conclusion that there is a low impact is reached.

RAI 5-9 (continued)

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
DA-D6	USE generic common cause failure probabilities consistent with available plant experience. EVALUATE the common cause failure probabilities consistent with the component boundaries.	Plant-specific CCF failure documentation (OSC-8797) was reviewed to ensure that the generic CCF estimates were consistent with plant operating experience. However, no evidence is provided to show that the component boundaries used in the CCF generic estimates are consistent with the component boundaries assumed for the PRA. (See also DA-A1a).	**Resolution:** Provide documentation in SAAG 637 of the comparison of the component boundaries assumed for the generic CCF estimates to those assumed in the ONS PRA to ensure that these boundaries are consistent. **Impact:** No impact is expected for documentation issues.	**e)** The "resolution" appears to confirm that a comparison of the boundaries needs to be performed while the "impact" simply states that no impact is expected. Beyond assuming that this is a documentation issue, explain how the conclusion that there is no impact is reached.
SC-A4	SPECIFY success criteria for each of the key safety functions identified per SR AS-A2 for each modeled initiating event [Note (2)].	F&O TH-2 remains open regarding basis and use of TH analyses for success criteria. The success criteria documentation (LPI with no LPSW SAAG 569, and HPI for Small and Medium LOCAs SAAG 213) is 7-11 years old and documents the success criteria for a few very specific functions, but does not provide an indication that such documentation is available for all safety functions and initiators. A review of samples of the AS documentation	**Resolution:** Improve the documentation on the TH bases for all safety function success criteria for all initiators. **Impact:** No impact is expected for documentation issues.	**f)** Success criteria can have a major impact on the PRA results, either through inappropriate success criteria or through improper modeling. Also, note similar discussions about weaknesses in the success criteria evaluations in SC-B5 and SC-C1. A lack of reasonable documentation may mean that incorrect evaluations or mistakes have been made Describe the success

RAI 5-9 (continued)

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
		(ATWS, Transients - SAAG 671, LOCAs - SAAG 241) determined that general statements of success criteria and references are included.		criteria for all safety functions and initiators and confirm that they have appropriately modeled in the PRA.
SY-A14	In meeting SY-A12 and SY-A13, contributors to system unavailability and unreliability (i.e., components and specific failure modes) may be excluded from the model if one of the following screening criteria is met: (a) A component may be excluded from the system model if the total failure probability of the component failure modes resulting in the same effect on system operation is at least two orders of magnitude lower than the highest failure probability of the other components in the same system train that results in the same effect on system operation. (b) One or more failure modes for a component may be excluded from the systems model if the contribution of them to the total failure rate or probability is less than 1% of the total failure rate or probability for that component, when their effects on system	Some failure modes are excluded in a qualitative fashion rather than by using quantitative criteria. Examples: the containment isolation system write-up notes that "electrical penetrations are not modeled due to their low probability of failure;" the High Pressure Service Water (HPSW) system write-up uses a redundancy argument for excluding inadvertent isolation of the main headers; the RCS write-up states that "transfer failure events for motor-operated valves, manual valves and check valves with 24 hr exposure times are not modeled unless probabilistically significant with respect to 'neighboring' basic events; the RPS write-up uses a diversity argument for	**Resolution:** Provide quantitative evaluations for screening. **Impact:** No impact is expected for documentation issues.	**g)** The "resolution" appears to confirm that quantitative evaluations are needed while the "impact" simply states that no impact is expected. Beyond assuming that this is simply a documentation issue, explain how the conclusion that there is no impact is reached.

RAI 5-9 (continued)

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
	operation are the same.	excluding common mode failure of sensors or instrument strings that generate a reactor scram signal.		
SY-A8	ESTABLISH the boundaries of the components required for system operation. MATCH the definitions used to establish the component failure data. For example, a control circuit for a pump does not need to be included as a separate basic event (or events) in the system model if the pump failure data used in quantifying the system model include control circuit failures. MODEL as separate basic events of the model, those subcomponents (e.g., a valve limit switch that is associated with a permissive signal for another component) that are shared by another component or affect another component, in order to account for the dependent failure mechanism.	Component boundaries are consistent with those identified in the data analysis (OSC-8796). For example, breakers that supply a single load are generally included within the boundary of the load. (An exception are the 4 kV circuit breakers.) However, boundaries are not discussed. Interlocks are explicitly modeled in the system models.	**Resolution:** Enhance systems analysis documentation to discuss component boundaries. **Impact:** No impact is expected for documentation issues.	**h)** Miss-modeled component boundaries can have an impact on the results, particularly if common parts are modeled as separate. Beyond assuming that this is simply a documentation issue, explain how the conclusion that there is no impact is reached.
SY-B15	IDENTIFY SSCs that may be required to operate in conditions beyond their environmental qualifications. INCLUDE dependent failures of multiple SSCs that result	SSCs that may be required to operate in conditions beyond their environmental qualifications are not identified. Examples include: • LOCA	**Resolution:** Cut set review during applications should address this. Suggest adding this guidance to workplace	**i)** The "resolution" appears to confirm that an evaluation of environmental conditions (beyond the fire zone) is needed while the

RAI 5-9 (continued)

	Category II Requirements	**Assessment Comment**	**Industry Resolution/ Expected Impact**	**NRC Staff RAIs**
	from operation in these adverse conditions. Examples of degraded environments include: (a) LOCA inside containment with failure of containment heat removal (b) safety relief valve operability (small LOCA, drywell spray, severe accident) (for BWRs) (c) steam line breaks outside containment (d) debris that could plug screens/filters (both internal and external to the plant) (e) heating of the water supply (e.g., BWR suppression pool, PWR containment sump) that could affect pump operability (f) loss of NPSH for pumps (g) steam binding of pumps (h) harsh environments induced by containment venting or failure that may occur prior to the onset of core damage	inside containment with failure of the Reactor Building Cooling system would expose steam generator instrumentation to a harsh environment • Steam line breaks in the Turbine Building could expose equipment other than just the 4 kV switchgear and EFW control panel to an adverse environment • Clogging of the RBES is not discussed and is not included in the system models.	procedure XSAA-103. **Impact:** No impact is expected for documentation issues.	"impact" simply states that no impact is expected. Beyond assuming that this is simply a documentation issue, explain how the conclusion that there is no impact is reached.
SY-C2	DOCUMENT the system functions and boundary, the associated success criteria, the modeled components and failure modes including human actions, and a description of modeled dependencies including support system and common cause failures, including the inputs, methods, and	The system notebooks contain much of the information listed in this SR. However, system model documentation should be enhanced to comply with all ASME PRA Standard requirements.	**Resolution:** Enhance system model documentation to comply with all ASME PRA Standard requirements. **Impact:** No impact is expected for documentation issues.	**j)** The "resolution" appears to confirm that there are multiple limitations in the documentation that supports the ONS PRA. Explain when the PRA will be enhanced to comply with the ASME PRA standards, and how ONS has

RAI 5-9 (continued)

	Category II Requirements	Assessment Comment	Industry Resolution/ Expected Impact	NRC Staff RAIs
	results. For example, this documentation typically includes: (a) system function and operation under normal and emergency operations (b) system model boundary (c) system schematic illustrating all equipment and components necessary for system operation (d) information and calculations to support equipment operability considerations and assumptions (e) actual operational history indicating any past problems in the system operation (f) system success criteria and relationship to accident sequence models (g) human actions necessary for operation of system (h) reference to system-related test and maintenance procedures (i) system dependencies and shared component interface (j) component spatial information (k) assumptions or simplifications made in development of the system models (l) the components and....			developed confidence in an apparently poorly documented model to the extent necessary to transition to a risk-informed fire protection program.

RAI 5-9 RESPONSE:

a) At the time that Supporting Requirement AS-A9 was assessed, the TH analysis using MAAP 4.0.6 had not yet been completed. The FPRA was based on the then-current TH analysis of record, which determined that two HPI pumps are required for feed and bleed cooling. Thus the fire model is conservative with respect to the latest thermal hydraulic analysis. However, the impact of this conservatism on the base case CDF and LERF results is negligible. (This is because unavailability of the HPI system is dominated by initiating events that fail motive power to the system and by common cause failures that render the supply or discharge paths unavailable, rather than availability of an individual HPI pump.) Similarly, the impact on fire change evaluations is negligible.

b) ONS Calculation OSC-9075, Oconee Thermal-Hydraulic Success Criteria for PRA Application (Ref. 1), documents the most recent ONS analysis. The analysis determines success criteria for small, medium and large LOCAs, feed and bleed sequences, stuck open pressurizer relief valve sequences and steam generator tube rupture sequences.

For sequences other than large LOCA, the MAAP code is employed in establishing success criteria for PRA application. MAAP has certain limitations that are especially important in the modeling of the short-term behavior of large LOCAs. The MAAP models are not sufficient for addressing flow reversal in the core and ECCS bypass. Thus for large LOCA, a review of the vendor analyses (References 2 and 3) was performed to assist in the determination of an appropriate set of success criteria.
The success criteria results are summarized in Table 5-9b below. As described above, only the feed and bleed success criteria have changed. However, this has a negligible impact on the FPRA.

Table 5-9b
PRA Success Criteria Summary

Initiating Event	Core Heat Removal, Early (Injection Phase)	Core Heat Removal, Late (Recirculation Phase)	Pressurizer PORV Required	Pressurizer Safety Valves Required	SSHR Required	Sump Recirculation Heat Removal
Large LOCA ($LL \geq 4.5$ inch dia.)[4]	1 LPI Pump AND 1 Core Flood Tank	1 LPI Pump	None Required	None Required	None Required	1 LPI Heat Exchanger OR 1 RBCU[4]
Core Flood Tank Line Break (Large LOCA)	1 HPI Pump into 4/4 Injection Paths AND 2 Core Flood Tanks AND RCP Trip in < 2 Minutes	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	None Required	None Required	1 LPI Heat Exchanger OR 1 RBCU[4]
Medium LOCA (1.5 inch dia. $\leq ML < 4.5$ inch dia.) with SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	None Required	1/2 Motor-driven EFW Pump OR 1/1 Turbine-driven EFW Pump	1 LPI Heat Exchanger OR 1 RBCU[4]
Medium LOCA (1.5 inch dia. $\leq ML \leq 3.6$ inch dia.) without SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	None Required	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]

Table 5-9b (continued)
PRA Success Criteria Summary

Initiating Event	Core Heat Removal, Early (Injection Phase)	Core Heat Removal, Late (Recirculation Phase)	Pressurizer PORV Required	Pressurizer Safety Valves Required	SSHR Required	Sump Recirculation Heat Removal
Medium LOCA (3.6 inch dia. < ML < 4.5 inch dia.) without SSHR	1 HPI Pump into 2/4 Injection Paths AND 1 Core Flood Tank	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	None Required	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]
Small LOCA (3/8 inch dia. ≤ SL < 1.5 inch dia.) with SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	None Required	1/2 Motor-driven EFW Pump OR 1/1 Turbine-driven EFW Pump	1 LPI Heat Exchanger OR 1 RBCU[4]
Small LOCA (3/8 inch dia. ≤ SL < 1.5 inch dia.) without SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	1/2 PZR Safety Valves	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]
Steam Generator Tube Rupture with SSHR (with injection)	1 HPI Pump into 2/4 Injection Paths	None Required[1]	None Required	None Required	1/2 Motor-driven EFW Pump OR 1/1 Turbine-driven EFW Pump	None Required

Table 5-9b (continued)
PRA Success Criteria Summary

Initiating Event	Core Heat Removal, Early (Injection Phase)	Core Heat Removal, Late (Recirculation Phase)	Pressurizer PORV Required	Pressurizer Safety Valves Required	SSHR Required	Sump Recirculation Heat Removal
Steam Generator Tube Rupture without SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	1/1 PORV	None Required	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]
Steam Generator Tube Rupture without SSHR (without injection)	None Required[2]	None Required[2]	None Required	None Required	1/2 Motor-driven EFW Pump OR 1/1 Turbine-driven EFW Pump into Unbroken SG	None Required
Stuck Open Pressurizer PORV Valve without SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	N/A	1/2 PZR Safety Valves	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]
Stuck Open Pressurizer PORV Valve with SSHR	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	N/A	None Required	1/2 Motor-driven EFW Pump OR 1/1 Turbine-driven EFW Pump	1 LPI Heat Exchanger OR 1 RBCU[4]

Table 5-9b (continued)
PRA Success Criteria Summary

Initiating Event	Core Heat Removal, Early (Injection Phase)	Core Heat Removal, Late (Recirculation Phase)	Pressurizer PORV Required	Pressurizer Safety Valves Required	SSHR Required	Sump Recirculation Heat Removal
Stuck Open 1 Pressurizer SRV	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	N/A	None Required	1 LPI Heat Exchanger OR 1 RBCU[4]
Stuck Open 2 Pressurizer SRVs	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	N/A	None Required	1 LPI Heat Exchanger OR 1 RBCU[4]
Transient Feed and Bleed (no operator action)	1 HPI Pump into 2/4 Injection Paths	1 HPI Pump into 2/4 Injection Paths AND 1 LPI Pump	None Required	1/2 PZR Safety Valves	N/A	1 LPI Heat Exchanger OR 1 RBCU[4]

Table 5-9b (continued)
PRA Success Criteria Summary

<table>
<tr><th>Initiating Event</th><th>Core Heat Removal, Early (Injection Phase)</th><th>Core Heat Removal, Late (Recirculation Phase)</th><th>Pressurizer PORV Required</th><th>Pressurizer Safety Valves Required</th><th>SSHR Required</th><th>Sump Recirculation Heat Removal</th></tr>
<tr><td rowspan="5">Transient Feed and Bleed (with operator action)</td><td>1 HPI Pump into 2/4 Injection Paths</td><td rowspan="2">1 HPI Pump into 2/4 Injection Paths</td><td rowspan="5">None Required</td><td rowspan="5">1/2 PZR Safety Valves</td><td rowspan="5">N/A</td><td rowspan="2">1 LPI Heat Exchanger</td></tr>
<tr><td>AND</td></tr>
<tr><td>Trip 3/4 RCPs[3]</td><td>AND</td><td>OR</td></tr>
<tr><td>AND</td><td rowspan="2">1 LPI Pump</td><td rowspan="2">1 RBCU[4]</td></tr>
<tr><td>Turn off PZR Heaters within 49 Minutes of RCS pressure > 2300 psig</td></tr>
</table>

[1] Core failure does not occur without recirc until well after the PRA mission time of 24 hours. For a longer mission time the operators either have to refill the BWST or start 1 LPI pump for recirculation.

[2] Core failure does not occur without inj. and recirc. Until well after the mission time of 24 hours. For a longer mission time the operators have to start RCS injection or start 1 LPI pump for recirculation.

[3] If RCPs not already tripped due to loss of subcooling

[4] RG 1.200 Capability Category I only

c) The "resolution" is Duke's long term plan to close the SR. However, for FPRA and other applications, it is concluded that this is a documentation issue only. The determination that the resolution to SR DA-B2 is a documentation issue was made based on a couple of considerations. First, the reviewer did not identify any technical issues; rather, it was noted that the documentation does not include a review for outliers. Second, the data calculations (Refs. 4 - 7) do not indicate differing failure rates within a component group. For example, the analysis documents two failures of MOVs to open on demand. Most of the MOVs in the PRA model are demanded four times a year (based on a quarterly test) or approximately once per year (based on a refueling outage test). A couple of the MOVs have a higher number of yearly demands. Approximately 70% of the demands are contributed from the group that is demanded from one to four times per year. Not surprisingly, both of the MOV failures came from the subgroup of MOVs that are demanded from one to four times per year. Based on the relatively small number of failures, there is no statistical basis for differentiating between infrequently demanded and frequently demanded MOVs. Therefore, there is no technical evidence that any outliers are included. Thus there is no impact on the FPRA.

d) The "resolution" is Duke's long term plan to close the SR. However, for FPRA and other applications, it is concluded that this is a documentation issue only. Further evaluation of this SR indicates that it is currently met. There are two mechanisms for coordinating maintenance activities: the Technical Specifications and the Maintenance Rule a(4) program. Restrictions imposed by both programs are accounted for in the ONS PRA model.

   ONS Work Control does coordinate work such that maintenance tasks that require a component to be out of service are performed under the same work window. Their phrase for this is 'bundling'. They use train rotation windows in the schedule to help with the bundling. Electrical work (i.e. supply breaker) may be bundled with a work on a motor-operated component. For example, a Low Pressure Injection pump lubrication PM could be bundled with a PM for the 4160 supply breaker. However, this type of maintenance coordination does not involve more than one train of equipment, and does not result in the plant taking on more risk (Ref. 8).

e) The "resolution" is Duke's long term plan to close the SR. However, for FPRA and other applications, it is concluded that this is a documentation issue only. The determination that the resolution to SR DA-D6 is a documentation issue was made based on various considerations. First, the reviewer did not identify any technical issues; rather, it was noted that the documentation does not address whether component boundaries used in the CCF generic estimates are consistent with those assumed for the PRA.

Second, the modeled common cause groups are consistent with generic experience. For example, for a generic DC power system, NUREG/CR-5497 (Ref. 9) develops common cause parameters for batteries and chargers. The ONS PRA includes common cause failure events for these same components (Ref. 10). Furthermore, the NUREG includes the AC input breaker and the DC output breaker within the battery charger component boundary. The battery output breaker and associated fuses were considered integral parts of the battery. In contrast, the breakers are explicitly modeled in the ONS PRA. However, should a CCF of the breakers occur, such a failure would be evaluated for impact on the CCF probability of the associated charger or battery.

Third, plant-specific operating experience is reviewed for impact on the PRA model (Refs. 6 and 7) in a manner consistent with component boundaries. For example, random and CCF of the instrument air dryer prefilters are both modeled in the PRA. When a CCF of these failures occurred, the failures were evaluated for both random and CCF rate impact (Ref. 7). Similarly, an issue involving potential CCF of 600 V breakers to close was evaluated for both random and CCF rate impact (Ref. 7).

Finally, the CCF analysis process has been reviewed internally by Duke's subject matter expert. No technical issues related to component boundaries were identified. Therefore, there is no evidence of a technical problem with common cause boundaries, just a need to improve the documentation. Thus there is no impact on the FPRA.

f) The success criteria analysis was updated in ONS calculation OSC-9075, Oconee Thermal-Hydraulic Success Criteria for PRA Application (Ref. 1), in accordance with this SR. This SR is now met.

g) The long term resolution acknowledges that the PRA Standard requires a quantitative evaluation for excluding components and specific failure modes. However, it is expected that conversion to a quantitative approach would not change the decision about whether or not to exclude a component or failure mode. A review of our qualitative screening process confirms this expectation. For example, the assessment comment notes that the PRA assumes that failure of High Pressure Service Water (HPSW) flow to various load branch lines is considered probabilistically insignificant due to the extent to which the various HPSW headers are interconnected. The HPSW pumps (and Elevated Water Storage Tank) are connected to a common header that forms a loop through-out the plant, providing a reliable source of fire protection, bearing lubrication, sealing, and cooling water to the three ONS units (Ref. 11). Thus if back-up cooling to the Unit 3 turbine-driven emergency feedwater pump cooling jacket became unavailable because 3HPSW-33 transferred closed, flow would be provided through 3HPSW-75 (Ref. 12). The probability that both manual valves would transfer closed over a 24-hour period is negligibly small (approximately 5E-13, Ref. 4).

Similarly, the assessment comment notes that "transfer failure events for motor-operated valves... with 24 hr exposure times are not modeled unless probabilistically significant with respect to 'neighboring' basic events." Consider the case of a flow path that is failed if an MOV fails to open or spuriously transfers closed. For failure of an MOV to open on demand, the ONS Rev. 3a PRA uses a failure probability of 2.6E-3 (Ref. 4). The probability that an MOV transfers closed over a 24-hour period is estimated to be 1.5E-6 (Ref. 4), or less than 0.1% of the demand failure probability. In cases like this, not including the relatively low probability failure mode in the PRA model does not have an appreciable impact on the model solution results. Therefore, there is no evidence of a technical problem associated with the screening of components or component failure modes, just a need to document a quantitative screening. Thus there is no impact on the FPRA.

h) The assessment comment notes that basic event component boundaries are consistent with the latest data analysis calculation (Ref. 13). The component boundaries are consistent with those defined in the source documents, such a NUREG/CR-6928. Dependencies among components, such as interlocks, are explicitly modeled, consistent with the PRA Modeling Guidelines workplace procedure (Ref. 14). No technical problems were identified during the assessment. Finally, the data analysis process has been reviewed internally by Duke's subject matter expert. No technical issues related to component boundaries were identified. Therefore, there is no evidence of a technical problem with component boundaries, just a need to improve the documentation. Thus there is no impact on the FPRA.

i) The "resolution" is Duke's long term plan to close the SR. However, for fire PRA and other applications, it is concluded that this is a documentation issue only. The examples cited in the assessment comment are not expected to have a material impact on the PRA results for the following reasons:

<u>LOCA inside containment with failure of the RB Cooling system</u>
Loss of all RB Cooling Units is a low probability event. SG level instrumentation is not required for all sequences leading to the harsh environment in containment. The combination of circumstances that lead to the harsh environment in containment and also would require the operators to have SG level instrumentation in order to prevent core damage is expected to be an insignificant contributor to the PRA results.

<u>Steam line breaks in the Turbine Building</u>
It is possible that other equipment could be affected by a steam line break in the Turbine Building; however, it would not be expected to contribute to CDF. If all 4 kV power is lost and the turbine-driven emergency feedwater pump is lost, the station would rely on the SSF. The loss of additional Turbine Building equipment should not be material to the PRA results.

<u>Clogging of the RB Emergency Sump</u>
Historically, all of the available design information has suggested that the probability of sump blockage was expected to be insignificant. It was neglected on that basis. In recent years this conclusion has been questioned but at the same time led to a requirement to modify sumps to again reduce the probability of clogging to an acceptably low level. Other failure modes that have the equivalent consequences to clogging exist in the PRA model. Unless clogging has a probability that is significant

when compared to these other failure modes, its omission has no material impact on the PRA results.

Therefore, there is no evidence of a technical problem associated with components that may be required to operate in conditions beyond their environmental qualification, just a need to improve the documentation. Thus there is no impact on the FPRA.

j) The long term resolution acknowledges that the ONS PRA documentation must be amended to meet the new standard set by ASME, as modified by RG 1.200. However, this does not imply that the current documentation is poor. The original ONS PRA model (NSAC/60) was a pilot for the industry and received many levels of internal and external review. More recent peer review of the ONS model (Ref. 15) concluded that the models are acceptable for applications. The model has been exercised for multiple applications and results reviewed for accuracy and completeness, has been compared to SPAR models, has been used for MSPI comparisons, etc. Thus prior to the advent of the ASME PRA Standard, the quality associated with the ONS PRA had been demonstrated to be adequate for supporting applications.

Revision 4 to the ONS PRA is currently in progress. One of the initiatives associated with this revision is to improve the documentation to meet the current ASME PRA Standard supporting requirements. One of the comments from the ONS Rev. 3 PRA gap analysis (Ref. 16) is that "the PRA Standard requires significantly more discussion of technical issues, key assumptions, and tests of reasonableness than has been typically developed for industry PRAs." The observation that ONS PRA documentation needs to be improved to fully comply with all of the ASME PRA Standard's requirements is neither unique to ONS nor is it surprising. Improving the PRA documentation to fully comply with all of the SRs is a process that has required and will continue to require a significant investment of time. Some of the guidance documents on how to meet standard requirements (e.g., treatment of parameter and model uncertainty, Refs. 17, 18) have only recently been made available.

As noted in the assessment comment, many of the requirements listed in documentation supporting requirement SY-C2 are addressed in the system notebooks. No technical issues are identified. Therefore, the model documentation is deemed sufficient for supporting the transition to a risk-informed fire protection program.

References

1. Oconee Thermal-Hydraulic Success Criteria for PRA Application, OSC-9075, March 2009.

2. BWNT LOCA Analysis, BAW-10192PA, Framatome ANP Inc., June 1998.

3. Summary of PSC 2-00 Analyses, Report 51-5009856-00, Rev. 1, Framatome ANP Inc., August 2001.

4. Oconee Failure Rates/Maintenance Unavailabilities, SAAG 522, Rev. 4, June 2004.

5. Oconee Component Failure Rate Denominator Estimates, SAAG 495, April 1998.

6. PSA Assessment of ONS Maintenance Rule Experience 1996 through 1997, SAAG 505, June 1998.

7. PSA Assessment of ONS Maintenance Rule Experience, Period Ending June 1999, SAAG 576, January 2000.

8. Wilkerson. L. C., E-mail to R. P. Boyer, Re: work control question, 6/4/09.

9. Common-Cause Failure Parameter Estimation, NUREG/CR-5497, Idaho National Engineering and Environmental Laboratory, October 1998.

10. Oconee PRA Revision 3 Common Cause Analysis, OSC-8797, Rev. 2, March 2007.

11. Design Basis Specification for the High Pressure Service Water System, OSS-0254.00-00-1002, Rev. 25, January 2007.

12. Flow Diagram of High Pressure Service Water System (Turbine Building), OFD-124C-3.2, Rev. 28, July 2006.

13. Oconee Failure Rates/Maintenance Unavailabilities, OSC-8796, Rev. 1, June 2007.

14. Workplace Procedure for PRA Modeling Guidelines, XSAA-115, Rev. 11, October 2008.

15. Oconee Nuclear Station Probabilistic Risk Assessment Peer Review Report, 38-1288512-00, B&W Owner's Group Risk-Informed Assessment Committee, October 2001.

16. An Independent Review of the Oconee PRA Against the Requirements of the ASME PRA Standard, Maracor Software & Engineering, Inc., August 2006.

17. Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments, EPRI Report 1016737, December 2008.

18. Guidelines on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making – Main Report, NUREG-1855, Vol. 1, March 2009.

RAI 5-10:

The table in attachment U is reported to contain all the F&Os from one of the reviews but there appear to be entries missing from the table. Explain why the entry for DA-D6 in table in attachment U refers the reader to see also DA-A1a but there is no DA-A1a in the table. HR-A3 refers to HR-A1, which is not in the Table.

RAI 5-10 RESPONSE:

The table in attachment U contains all open ASME PRA Standard Supporting Requirements (SRs). SRs DA-A1a and HR-A1 are met, thus they are not included in the table. DA-A1a is met since the data analysis calculation (Ref. 1) was revised to include a discussion of component boundaries. HR-A1 is met because a latent human error has been assigned to every train of standby equipment (Ref. 2). Attachment U will be revised to remove these references and submitted in November, 2009.

References

1. Oconee Failure Rates/Maintenance Unavailabilities, OSC-8796, Rev. 1, June 2007.

2. Oconee Human Reliability Analysis, OSC-8798, Rev. 1, February 2007.

RAI 5-11:

Provide clarification of PRA analysis methodology to be used for change evaluation and self-approval post-transition. ONS needs to clarify its bounding analysis method and confirm that only bounding non-compliant analyses as described in its submittal have been performed. Otherwise, ONS needs to describe any other analyses (e.g., where non-compliance is not assumed to cause complete failure) were used. ONS also needs to identify what analyses methods will be used to support future, self-approved changes. ONS needs to clearly describe the PRA analyses it has used to estimate changes in risk in support the transition to 10 CFR 50.48(c) and the analyses it intends to use for self-approval of future changes if such authority is granted by the NRC.

In the February 9, 2009, submittal, ONS stated that the Fire PRA analyses used to estimate the change in risk are described in Appendix C of its October 31, 2008, submittal. Appendix C in the October 31, 2008, submittal appears to include only PRA results and the Staff was unable to identify any analyses descriptions. Attachment 1 of the February 9, 2009, submittal does include a general description of the PRA analyses as the change in risk between the compliant and the non-compliant case (OSC-9518 R2, page 5). The compliant case is conservatively estimated by assuming that the non-compliant "failure" is removed from the PRA models and the CDF recalculated. Page 40 of OSC-9518 R2 appears to describe an alternative method where some fractional increase in CDF is estimated although it is unclear how the fractional increase can be calculated without using the method on page 5. This submittal further discusses the inclusion and treatment of future modifications in the base line PRA.

RAI 5-11 RESPONSE:

Please note that this RAI response is preliminary and will be finalized when ONS Calculation, OSC-9518, is completed.

The response to this RAI consists of two parts. The response to the RAI described in the first paragraph is similar to RAI 1-2. As such, the response to the first paragraph is combined with the response to RAI 1-2 and is not included here.

The method described in Appendix C of ONS calculation, OSC-9518, NFPA 805 FPRA Application, relied on calculating the difference in risk between the pre and post modification target set. Rather than removing a single non-compliance from the set of failed components, this method removed targets (trays) that would no longer be within the zone of influence. The presentation of this method was somewhat complicated by relying on a previously calculated CCDP for a non-severe case where critical trays were not impacted. Since the modifications associated with this method are no longer on TR, Attachment S, this discussion will be removed from Appendix C in an update to ONS calculation OSC-9518, NFPA 805 FPRA Application. However, the basic approach of calculating the change in risk based on target set differences is within the scope of the change process

RAI 5-12:

Describe how potential impact of induced steam generator tube ruptures has been evaluated as it relates to the possible impact on the change in LERF estimates from the Fire PRA.

ONS report, "Oconee Simplified LERF Methodology, SAAG 818." references NUREG-6595. NUREG-6595 discusses induced steam generator tube ruptures as a possible containment failure mode when core damage occurs at high pressure. The ONS SAAG 818 report did not appear to include this type of containment failure. High-pressure core damage sequences are defined in SAAG 818 as all non-LOCA sequences with loss of secondary side heat removal failure. However, fire core damage scenarios may well lead to non-LOCA sequences with loss of secondary side heat removal.

RAI 5-12 RESPONSE:

As documented in SAAG 818 (Ref. 1), it is assumed that induced tube ruptures would be of sufficiently low frequency that they would not contribute meaningfully to the evaluation of the LERF and their contribution is neglected. This assumption is predicated on the following considerations:

- Thermally induced Steam Generator tube ruptures (SGTR) are not probable if the steam side of the SGs remains pressurized (Ref. 2, p. 2-5). This is especially true for tubes with few flaws.

- Duke has explicitly evaluated the potential for induced SGTRs in the Level 3 PRA analysis (Ref. 3). The evaluation confirmed the low probability of SG tube failure so long as the RCPs are not restarted after core damage occurs. Since core damage occurs hours after the fire has been extinguished, spurious start of a RCP due to fire is not expected. In addition, there is no EOP guidance to restart the pumps. Thus the LERF contribution from induced SGTRs is relatively small.
- The "candy cane" region of the once through SG hot leg is expected to be especially effective at trapping hot gases generated during core damage. Any piping failure would be expected to occur at this location rather than at the steam generator tubes.

The following considerations based on work performed for EPRI by Creare Inc. (Ref. 4) also support the contention that the contribution to LERF from induced tube ruptures may be neglected:

- Unlike inverted u-tube SGTRs, the SG tubes are not the high point in the RCS.
- Natural circulation experiments run in vertical tubes suggest the following dynamics:
  - Fairly vigorous stratified flow in the horizontal part of the hot leg
  - Unorganized and weak flow in vertical section
  - No flow into the tubes from the candy cane region
- Natural circulation flow through the tubes is unlikely due to water trapped in the RCP suction portion of the cold legs, which provides an effective loop seal. This is a very deep pool in lowered loop plants like ONS.

The Creare evaluation concluded that no thermal challenge is expected unless whole-loop circulation begins. Whole-loop circulation is nearly impossible in lowered loop plants in the absence of pump bumping after core damage.

References

1. Oconee Simplified LERF Methodology, SAAG 818, Rev. 1, December 2004.
2. An Approach for Estimating the Frequencies of Various Containment Failure Modes and Bypass Events, NUREG/CR-6595, Rev. 1, October 2004.
3. Oconee PRA Level 2 & 3 Analysis, June 1995.
4. MAAP Calculations of High Pressure Severe Accident Sequences in B&W Plants, Presentation to SG Risk Assessment Advisory Committee, MTG-97-875, Marc Kenton, Creare Inc., December 1997.

RAI 5-13:

Describe reviews and resolutions of associated F&Os of aspects of the PRA and Fire PRA used to support the LAR but for which reviews are not documented in the TR (this is associated with ONS RAI 5-2).

For example, The NRC recommended that the Unit 3 FPRA receive a follow-up focused scope peer review of those aspects that were not ready for review, or for which findings were issued, during the Staff review. In addition, there is a need to review the FPRA for Units 1 and 2, at least to the extent where they may differ methodologically or plant-specifically from Unit 3 FPRA, against the supporting requirements (SRs) for Unit 3, especially where aspects were not ready or for which findings were issued. This is not documented in the LAR submittal. In addition, Table V-3 of the TR identifies SRs UNC-A1, UNC-A2, and UNC-A3, but these SRs were not reviewed by the NRC staff during their review nor is information provided in the LAR submittal describing the uncertainty and sensitivity analysis performed for the Fire PRA.

RAI 5-13 RESPONSE:

The response to RAI 5-2 addresses the "history" of the reviews performed on the internal events PRA. Review and resolution of associated F&Os of aspects of the PRA and FPRA were addressed via responses to RAIs 5-8 & 5-9.

As discussed during the meeting with the NRC on May 13 & 14, 2009, and documented in RG 1.205, the NRC review of the pilot plant FPRAs is fulfilling the role of the industry peer review for the pilot plants and specifically for the ONS FPRA developed for transition to NFPA 805. To facilitate that objective, updated documentation was submitted to the NRC subsequent to the NRC review of the ONS FPRA in March, 2008.

Specifically with regard to uncertainty (which was not reviewed by the NRC in March of 2008), an uncertainty and sensitivity matrix was developed and included with ONS calculation, OSC-9378, Oconee Fire PRA Summary Report as Appendix D. In addition, sensitivity to uncertainty associated with specific FPRA parameters was quantitatively addressed in ONS calculation, OSC-9518, NFPA 805 FPRA Application Calculation. Both of these calculations have been made available to the NRC. Additional discussions regarding NRC expectations for uncertainty/sensitivity have resulted in additional effort that will be incorporated into a future revision of ONS calculation, OSC-9518.

As discussed in Section 4.5.1.3 of the TR, an NRC assessment (or Industry peer review) of the Unit 2 FPRA is not required since the development of the Unit 2 FPRA applied the same process, methodology and treatment details as that applied for the Unit 3 analysis. In fact, much of the FPRA documentation developed for Unit 3 is directly applicable to the Unit 2 FPRA. A separate FPRA Summary Report (OSC-9519) was developed for Unit 2 to document the CDF and LERF results. The Unit 2 calculation was provided to the NRC along with ONS calculation, OSC-9622, which provides the Unit 1 Comparative Screening Analysis. As discussed in Section 4.5.1.2 of the TR, the comparative screening analysis indicated that a separate Unit 1 fault tree and FPRA quantification file were not necessary. For the limited number of cases where the Unit 2 results were not considered to be bounding, a method for adjustment of the Unit 2 results for application to Unit 1 was provided.

RAI 5-14:

Describe the process for and results from assessing the interactions of fires that impact more than one unit.

The fires described in the LAR submittal may affect safe shutdown structures, systems, and components in more than one unit (e.g., it is expected that a fire in one Unit could impact equipment from another Unit). The analysis and results from these types of fires is not clearly discussed in the LAR submittal.

RAI 5-14 RESPONSE:

All of the fire compartments (fire zones) within the 3-unit plant were included within the scope of both the Unit 2 and 3 FPRA. All of the fire compartments were also considered within the scope of the comparative screening analysis performed for Unit 1.

ARTRAK contains cable routing information for all components within the scope of the Units 1, 2, & 3 safe shutdown analyses as well as some additional FPRA credited components. The credited equipment and cable location information is included in the ZoneTag table within the FRANC database. The same ZoneTag table is utilized during quantification of each FPRA model since the component footprint does not change with the unit analyzed. While the units are quantified individually, some of the scenarios developed for Unit 3 do, in fact, include failure of Unit 1 and/or Unit 2 components and some of the Unit 2 scenarios involve failure of Unit 1 and/or Unit 3 components. Accordingly, a single scenario can have multi-unit impacts. For example, the same fire scenario may have a significant impact on Unit 2 and a less significant impact on Unit 3. The impact on Unit 2 is counted in the CDF/LERF total for Unit 2 while the impact on Unit 3 is counted in the CDF/LERF total for Unit 3. In other words, while the target damage set is the same for a given scenario regardless of the unit, the impact on fire CDF/LERF is dependent on the unit being quantified.

Important shared equipment is directly modeled in the PRA and credit for the associated function is dependent upon the postulated equipment or power supply failures and/or cable damage for a given scenario. Likewise, credit for cross-tied support from another unit is also factored into the analysis. For example, the operator action to provide emergency feedwater from another unit may be credited provided the fire has not rendered the action too difficult or impossible to implement or disabled key equipment needed for success of the action.

A final element to consider is the impact, if any, on the HEP value for an operator action if more than one unit is impacted. An example of where this might come into play was discussed during the NRC visit to Oconee for the LAR Audit in February of 2009. The question pertained to the SSF action to throttle ASW flow assuming more than one unit was affected by the fire. This is the only SSF action that would not be considered simple. The NRC suggested that the HEP value for this action may vary depending on how many units are impacted by the fire. However, the number of licensed reactor operators dispatched to the SSF is commensurate with the number of units impacted. The action already contains a feedback step to throttle flow; consequently, each unit's reactor operator will need to adjust the ASW flow to that unit's SGs as necessary.

Therefore, regardless of whether the SSF ASW pump is used to feed the SGs for a single unit or for multiple units, there is no additional impact on the HEP value for throttling ASW flow.

RAI 5-15:

Provide information concerning which and how operator actions were evaluated for risk. Include in this discussion which actions are specifically modeled in the PRA and/or Fire PRA, which were evaluated using a quantitative assessment, which were evaluated using a qualitative bounding assessment, and which, if any, were not evaluated for their impact on risk. In addition, provide the results of the risk increase (CDF) evaluation for each operator action. Table G-2 of the TR should include some of this information (e.g., whether modeled, how evaluated) for each OMA.

Table G-2 of the TR provides a listing of OMAs and their binning, but is not clear which were evaluated using which methods. In addition, while the bounding analysis is briefly discussed, sufficient detail to judge the methodology is not provided. For example, Section G.5.2.2 notes in the first paragraph that "Qualitative methods were used to assess the additional risk presented by the use of Bin D OMAs," yet in the third paragraph states, "Bin D actions not directly associated with the SSF operation were qualitatively addressed for additional risk."

The LAR does not provide sufficient information for the staff to review the additional risk presented by the use of recovery actions. Section 4.2.4 of NFPA 805 requires that the additional risk of relying on recovery actions be evaluated.

RAI 5-15 RESPONSE:

Before RAI 5-15 can be addressed, issues related to FAQ 07-30 (OMA Transition to Recovery Actions) need to be resolved. These issues were discussed and feedback requested from the NRC in a conference call on July 10, 2009.

RAI 5-16:

Provide an evaluation of OMAs that are transitioning to recovery actions that is consistent with the definition of a recovery action in the Draft Regulatory Guide, DG-1218. In the response, identify the OMAs that changed classification from defense-in-depth to recovery action and provide an evaluation of the additional risk presented by these recovery actions.

In Attachment G of the TR, ONS classified OMAs as recovery actions based on the assumption that the definition of an "emergency control station" in RG 1.189, rev. 1, is the same as "primary control station" in Section 1.6.52 of NFPA 805. However, DG-1218 provides additional guidance on the NRC's definition of "primary control station" and it is different then an "emergency control station" as defined by RG 1.1.89. Because of this difference, there is insufficient information in the LAR for the staff to complete its review. Section 4.2.4 of NFPA 805 requires that the additional risk of recovery actions needs to be evaluated.

RAI 5-16 RESPONSE:

Before RAI 5-16 can be addressed, issues related to FAQ 07-30 (OMA Transition to Recovery Actions) need to be resolved. These issues were discussed and feedback requested from the NRC in a conference call on July 10, 2009.

RAI 5-17:

ONS document "Oconee Fire PRA Scenario Development," OSC-9375, summarizes the scenario development process and the individual scenario details. NFPA 805, Section 2.4.1.3, Fire Scenarios, states that,

> *...The fire scenarios shall establish the conditions under which a proposed solution is expected to meet the performance criteria. These scenarios shall become the fire protection design basis associated with performance objective for that area. The set of fire scenarios for each plant area shall include the following:*
> *(1) Maximum expected fire scenarios*
> *(2) Limiting fire scenarios."*

Provide the methods, assumptions, and results for the detailed fire modeling analysis that provides the basis for OSC-9375. Identify any specific fire modeling analyses which use maximum expected fire scenarios and limiting fire scenarios.

RAI 5-17 RESPONSE:

ONS used an integrated assessment approach as delineated in 4.2.4.2 of NFPA 805 and did not use the MEFS / LFS approach. The response to RAI 7-1 addresses the generic fire modeling treatments that provided the basis for ONS calculation OSC-9375, Oconee Fire PRA Scenario Development calculation.

RAI 6-1:

Provide a detailed overview of the monitoring program that will be used to assess acceptable levels of availability, reliability, and performance of fire protection systems and features. In the response, 1) identify the fire protection systems and features, and the attributes of those systems and features that will be monitored, 2) identify the criteria used to assess the acceptable level of availability, reliability, and performance for each attribute, 3) identify the methods used to monitor availability, reliability, and performance, and 4) describe how unacceptable levels of availability, reliability, and performance will be managed.

Section 2.6 of NFPA 805 requires licensees to establish and monitor acceptable levels of availability, reliability, and performance of fire protection systems and features. While a brief description of how the monitoring program will be developed is provided in Section 4.6 of the TR, insufficient detail is provided for the NRC to make an assessment of the adequacy of the program.

RAI 6-1 RESPONSE:

Duke will provide a detailed overview of the monitoring program by September 30, 2009.

RAI 7-1:

Section 2.4.1.2 of NFPA 805 requires the verification and validation (V&V) of all fire models used in the analysis. Additionally, Sections 2.7.3.2, 2.7.3.3, and 2.7.3.4 of NFPA 805 provide detailed requirements for the V&V, limitations of use, and user qualifications for all numerical methods, including fire models. Regulatory Position 4.2 of DG-1218 states, "Licensees may also propose the use of other fire models; however, licensees are responsible for providing evidence of acceptable V&V of these fire models using ASTM1355-05a standard. The V&V documents for licensee-proposed fire models should be submitted with the license amendment request for NRC review."

Section 4.5.2 of the TR states: "The approach taken at ONS to simplify the analysis process incorporates features of several fire model tools covered by NUREG 1824 as well as additional features. The approach is collectively referred to as the Fire Modeling Generic Treatments. The analysis basis and V&V documentation was provided in a proprietary Hughes Associates, Inc. report provided to the NRC on January 24, 2008." The "Fire Modeling Generic Treatments" document states: "The calculation methods are not bound or limited to those contained in any one document such as the Nuclear Regulatory Commission's (NRC) Nuclear Regulatory Guide (NUREG) 1805 [2004] or NUREG 6850 [2005]. Rather, the tools available were assessed and those that were most appropriate in terms of application, validation, range, and degree of conservatism were selected and used."

Identify the specific fire models and correlations (such as the Beyler/Shokri detailed target heat flux correlation) used. Provide assurance that the identified fire models and correlations were applied within their appropriate scopes and limitations. Provide specific version information (for CFAST for example), as well as a more detailed description of the V&V status of the applied fire models and correlations. One approach for the latter would be to note the consistencies and inconsistencies of the V&V efforts with ASTM E1355-05a "Standard Guide for Evaluating the Predictive Capability of Deterministic Fire Models" and justify the inconsistencies. Other approaches for judging the V&V status of the models and correlations may also be acceptable. Also, describe the qualifications of the Oconee engineers to use the methods and tools used by the analysis.

Of specific concern are fire location corner and wall proximity effects, which can affect entrainment and flame height as well as ZOI and target impacts. The ONS response should address how the effects of fires located near corners and walls were accounted for in the analysis (specifically in "Fire Modeling Generic Techniques" and in CFAST). In addition, for specific ignition sources identified in close proximity to walls and/or corners, how did the data collection account for such cases?

RAI 7-1 RESPONSE:

Only the 'Generic Treatments' discussed above were used for the ONS FPRA. No additional fire models (e.g., CFAST) or detailed fire modeling was performed. Table RAI 7–1(a) provides a summary of the correlations that have been used to develop the zone of influences about ignition sources in the 'Generic Treatments' document.

Table RAI 7–1(b) provides a cross reference between major sections of ASTM E 1355–05a and the correlations in terms of their applicability and their validation.

Qualified contractor personnel were utilized to develop and apply the 'Generic Treatments' for the FPRA. Personnel qualification requirements consistent with Section 2.7.3.4 of NFPA 805 will be developed for Duke personnel prior to application of the generic fire modeling treatments in subsequent fire scenario development activities.

Wall and corner effects as well as the impact of room size on the ZOI were addressed in ONS calculation OSC-9375, Fire Scenario Report. The scenario walkdowns did not identify any fixed ignition sources as being located in a corner, but a few fixed ignition sources were confirmed to be located against a wall and the impact on ZOI was dispositioned in ONS calculation, OSC-9375. Similarly, the impact on ZOI due to a small enclosure volume relative to the limiting radiant heat source was addressed during the hot gas layer evaluation. The hot gas layer evaluation includes assessment of the impact on the ZOI as well as the potential for room burnout. See answer to RAI 5-17 for further details of the application of fire modeling in scenario development

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Table RAI 7–1(a) – Summary of Correlations used in the Generic Treatments Document

Correlation	Location in Treatments	Original Reference	Application	Original Correlation Range	Subsequent Validation and Verification	Limits in Treatments
Flame Height	Page 18	Heskestad [1981]	Provides a limit on the use of the Zone of Influence (ZOI)	$-4 < \log_{10}\left(\frac{c_p^3 T_o \Delta T_f}{g^2 \rho_o^2}\right)\frac{\dot{m}r^3}{\alpha \Delta H_c D^5} < 1$ In practice, wood and hydrocarbon fuels, momentum or buoyancy dominated, with diameters between 0.05 – 10 m.	Directly NUREG 1824 [2007] Volume 3 Indirectly NUREG 1824 [2007] Volume 5 (Correlation used in CFAST)	$\frac{4\dot{m}\Delta H_c}{\pi D^2} < 3000$
Point Source Model	Page 19	Modak [1976]	Lateral extent of ZOI – comparison to other methods	Isotropic flame radiation. Compared with data for 0.37 m diameter PMMA pool fire and a target located at a Ro/R of 10.	NUREG 1824 [2007]; SFPE [1999]	Predicted heat flux at target is less than 5 kW/m^2 per SFPE [1999]
Method of Shokri and Beyler	Page 19	Shokri et al. [1989]	Lateral extent of ZOI – comparison to other methods	Pool aspect ratio < 2.5 Hydrocarbon fuel 1 – 30 m diameter Vertical target, ground level	SFPE [1999]	Ground based vertical target
Method of Mudan (and Croce)	Page 20	Mudan [1984]	Lateral extent of ZOI – comparison to other methods	Round pools; Hydrocarbon fuel 0.5 – 80 m diameter.	SFPE [1999]	Total energy emitted by thermal radiation less than total heat released
Method of Shokri and Beyler	Page 20	Shokri et al. [1989]	Lateral extent of ZOI	Round pools; Hydrocarbon fuel 1 – 50 m diameter	SFPE [1999] NUREG 1824 [2007] Volume 3	Predicted heat flux at target is greater than 5 kW/m^2 per SFPE [1999] Shown to produce most conservative heat flux over range of scenarios considered among all methods considered.

Table RAI 7–1(a) – Summary of Correlations used in the Generic Treatments Document (continued)

Correlation	Location in Treatments	Original Reference	Application	Original Correlation Range	Subsequent Validation and Verification	Limits in Treatments
Plume heat fluxes	Page 22	Wakamutsu et al. [1996]	Vertical extent of ZOI	Fires with an aspect ratio of about 1 and having a plan area less than 1 m^2.	Wakamatsu et al. [2003] (larger fires) Lattimer [2008]	Area source fires with aspect ratio ~ 1. Used with plume centerline temperature correlation; most severe of the two is used as basis for the ZOI dimension.
Plume centerline temperature	Page 23	Yokoi [1960]; Beyler [1986]	Vertical extent of ZOI	Alcohol lamp assumed to effectively be a fire with a diameter ~0.1 m.	NUREG 1824 [2007] Volume 3; Heskestad [2008]	Area source fires with aspect ratio ~ 1. Used with plume flux correlation; most severe of the two is used as basis for the ZOI dimension.
Hydrocarbon spill fire size	Page 51	Gottuk et al. [2002]	Determine heat release rate for unconfined hydrocarbon spill fires.	Hydrocarbon spill fires on concrete surfaces ranging from ~1 to ~10 m in diameter.	None. Based on limited number of observations.	None. Transition from unconfined spill fire to deep pool burning assumed to be abrupt.
Flame extension	Page 100	Lattimer [2002]	Determine the fire offset for open panel fires.	Corner fires ranging from ~10 to ~1,000 kW. Fires included gas burners and hydrocarbon pans.	None. Based on limited number of observations.	None. Offset is assumed equal to the depth of the ceiling jet from the experiments.
Line source flame height	Page 101	Delichatsios [1984]	Determine the vertical extent of the ZOI	Theoretical development.	Lattimer [2008]	None. Transition to area source assumed for aspect plan ratios less than 4. Maximum of area and line source predictions used in this region.

Table RAI 7–1(a) – Summary of Correlations used in the Generic Treatments Document (continued)

Correlation	Location in Treatments	Original Reference	Application	Original Correlation Range	Subsequent Validation and Verification	Limits in Treatments
Corner flame height	Page 108	Lattimer [2002]	Determine the vertical extent of the ZOI	Corner fires ranging from ~10 to ~1,000 kW. Fires included gas burners and hydrocarbon pans.	None. Correlation form is consistent with other methods; comparison to dataset from Lattimer [2002] provides basis.	None.
Air mass flow through opening	Page 140	Kawagoe [1958]	Compare mechanical ventilation and natural ventilation	Small scale, 1/3 scale, and full scale single rooms with concrete and steel boundaries. Vent sizes and thus opening factor varied. Wood crib fuels.	Drysdale [1999]; SFPE [2004]	None. SFPE [2004] spaces with a wide range of opening factors.
Line fire flame height	Page 210	Yuan et al. [1996]	Provides a limit on the use of the Zone of Influence (ZOI); Extent of ZOI for cable tray fires.	$0.002 < \frac{Z}{\dot{Q}'} < 0.6$ In practice, from the base to several times the flame height from 0.15 – 0.5 m wide gas burners.	None. Correlation form is consistent with other methods; comparison to dataset from Yuan et al. [1996] provides basis	None.
Cable heat release rate per unit area	Page 210	Lee [1985]	Provides assurance that the method used is bounding	Cables with heat release rates per unit area ranging from about 100 – 1000 kW/m^2.	None.	Correlation predicts a lower heat release rate than assumed in the Treatments and is based on test data.

Table RAI 7–1(a) – Summary of Correlations used in the Generic Treatments Document (continued)

Correlation	Location in Treatments	Original Reference	Application	Original Correlation Range	Subsequent Validation and Verification	Limits in Treatments
Line fire plume centerline temperature	Page 212	Yuan et al. [1996]	Provides a limit on the use of the Zone of Influence (ZOI); Extent of ZOI for cable tray fires.	$0.002 < \frac{Z}{\dot{Q}'} < 0.6$ In practice, from the base to several times the flame height from 0.15 – 0.5 m wide gas burners.	None. Correlation form is consistent with other methods; comparison to dataset from Yuan et al. [1996] provides basis	None.
Ventilation limited fire size	Page 283	Babrauskas [1980]	Assessing the significance of vent position on the hot gas layer temperature	Ventilation factors between 0.06 – 7.51 Fire sizes between 11 – 2800 kW Wood, plastic, and natural gas fuels	SFPE [2004]	None. Provides depth in the analysis of the selected vent positions.

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations

Correlation	ASTM E 1355–05a Section					
	<u>Section 7.1</u> Model (Correlation) Documentation	<u>Section 7.2</u> Scenarios for which Model has been validated	<u>Section 8</u> Theoretical basis for model (correlation)	<u>Section 9</u> Mathematical and Numerical Robustness	<u>Section 10</u> Model (Correlation) Sensitivity	<u>Section 11</u> Model Evaluation and <u>Section 12</u> Evaluation Report
Flame Height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$; Flame heights restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Point Source Model	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or $\dot{q}''_c$; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Method of Shokri and Beyler	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{q}''_c$; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Method of Mudan (and Croce)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative D_c; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Method of Shokri and Beyler	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative D_c; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Plume heat fluxes	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{q}''_c$; R_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Plume centerline temperature	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or ΔT_{CL} ; R_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Hydrocarbon spill fire size	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; fire size restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs.	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Flame extension	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; flame extensions restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs.	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Line source flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Corner flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or D_e; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Air mass flow through opening	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative vent height; $\dot{m}_{air}$ restricted to real, positive values.	One-to-one non-linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Line fire flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Cable heat release rate per unit area	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; fire size restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Line fire plume centerline temperature	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$ or Z; T_p restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Ventilation limited fire size	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative vent height; $\dot{Q}_v$ restricted to real, positive values.	One-to-one non-linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Flame Height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$; Flame heights restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Point Source Model	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or $\dot{q}_c''$; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Method of Shokri and Beyler	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{q}_c''$; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Method of Mudan (and Croce)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative D_e; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Method of Shokri and Beyler	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative D_e; R restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Plume heat fluxes	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{q}''_c$; R_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Plume centerline temperature	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or ΔT_{CL} ; R_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Hydrocarbon spill fire size	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; fire size restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs.	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Flame extension	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; flame extensions restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs.	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Line source flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Corner flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}$ or D_e; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Air mass flow through opening	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative vent height; $\dot{m}_{air}$ restricted to real, positive values.	One-to-one non-linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Line fire flame height	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$; F_H restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Cable heat release rate per unit area	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution with no singularities; fire size restricted to real, positive values.	One-to-one linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

Table RAI 7–1(b) – ASTM E 1355–05a Compliance Matrix Cross Reference for Generic Treatments Correlations (continued)

Correlation	ASTM E 1355–05a Section					
	Section 7.1 Model (Correlation) Documentation	Section 7.2 Scenarios for which Model has been validated	Section 8 Theoretical basis for model (correlation)	Section 9 Mathematical and Numerical Robustness	Section 10 Model (Correlation) Sensitivity	Section 11 Model Evaluation and Section 12 Evaluation Report
Line fire plume centerline temperature	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative $\dot{Q}'$ or Z; T_P restricted to real, positive values.	Sensitivity over range of parameters considered in Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008
Ventilation limited fire size	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008	Table RAI 7–1(a)	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008 Table RAI 7–1(a)	Closed form solution; singularity at negative vent height; $\dot{Q}_v$ restricted to real, positive values.	One-to-one non-linear input mapping; not sensitive to inputs	Generic Fire Modeling Treatments, Rev. 0, 1/15/2008

RAI 7-2:

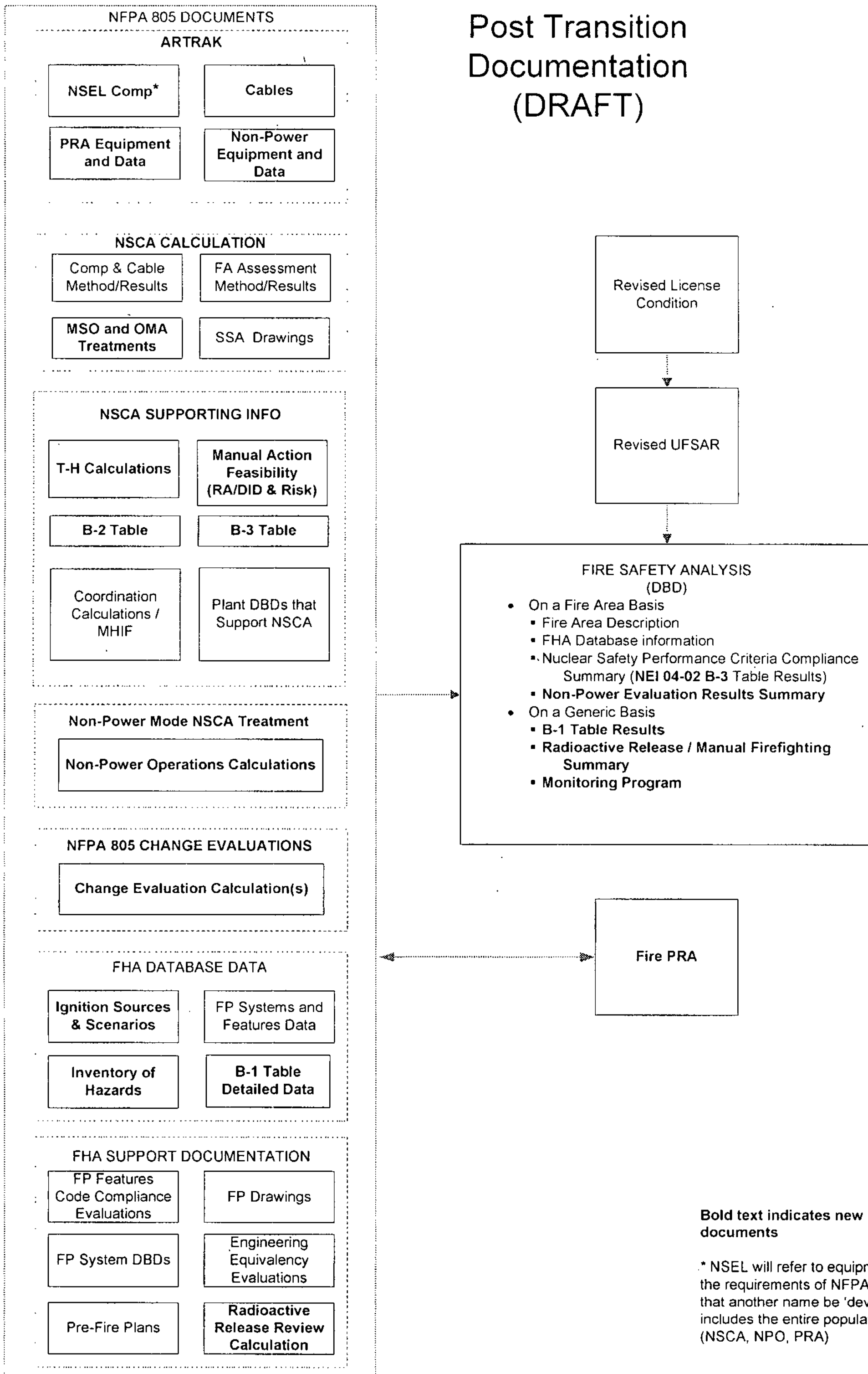
Provide a detailed overview of the configuration control and quality assurance programs that will be implemented on the new fire protection program. For the configuration control program, identify the design basis documents and supporting documents that will be maintained under configuration control, describe the process/procedure for maintaining these documents up-to-date, and describe the process/procedure for assuring that plant changes are reviewed for potential impact on the fire protection program. For the quality assurance program, describe the process/procedure for performing independent reviews of calculations and evaluations and describe the uncertainty analyses performed in support of the LAR.

Sections 2.7.2 and 2.7.3 of NFPA 805 require licensees to implement configuration control and QA programs, respectively.

RAI 7-2 RESPONSE:

The Oconee Fire Protection Program configuration is defined by the program documentation. This is depicted in the draft figure below showing the NFPA 805 post-transition fire protection program documents.

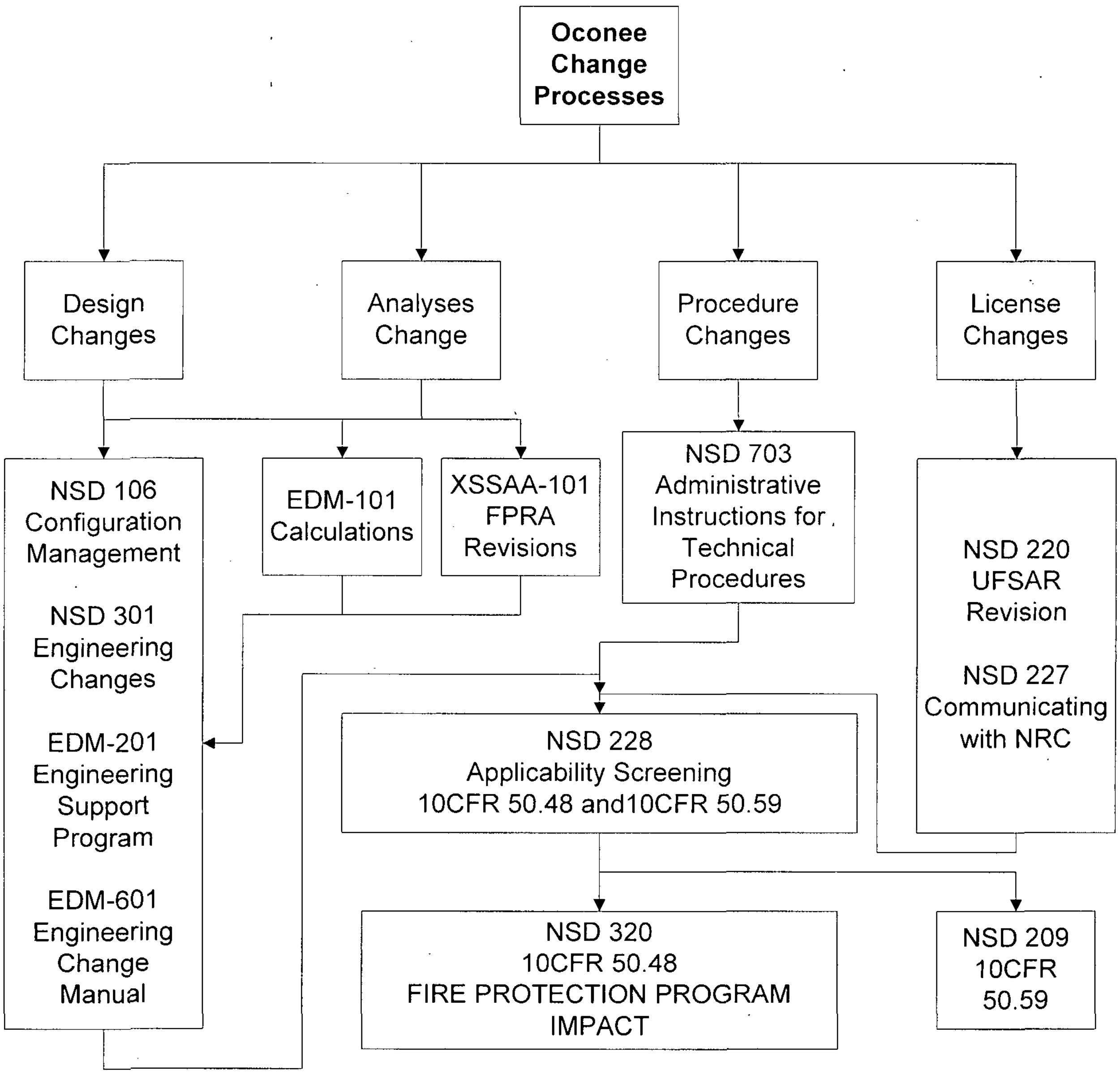
Post Transition Documentation (DRAFT)



Bold text indicates new NFPA 805 documents

* NSEL will refer to equipment / cables that support the requirements of NFPA Chapter 4 (at power) and that another name be 'developed' for the list that includes the entire population of equipment / cables (NSCA, NPO, PRA)

To the greatest extent possible, the existing configuration control processes for modifications, calculations and analyses, and Fire Protection Program License Basis Reviews will be utilized to maintain configuration control of the Fire Protection program documents. The configuration control procedures which govern the various Duke documents and databases that currently exist will be revised to reflect the new NFPA 805 licensing bases requirements. The main procedures and the change process they control are shown in the table below:



The items shown in bold text on the first figure will generally require new control procedures and processes to be developed since they are new documents and databases created as a result of the transition to NFPA 805. The new procedures will be modeled after the existing processes for similar types of documents and databases. System level design basis documents will have to be revised to reflect the NFPA 805 role that the system components now play.

The process for capturing the impact of proposed changes to the plant on the Fire Protection Program will continue to be a multiple step review. The first step of the review is an initial screening for process users to determine if there is a potential to impact the Fire Protection program as defined under NFPA 805 through a series of screening questions/checklists contained in one or more different procedures depending upon the change process being used. Reviews that identify potential Fire Protection program impacts will be sent to a select group of qualified individuals (classical Fire Protection, safe shutdown, FPRA) to ascertain the exact program impacts, if any. If Fire Protection program impacts are determined to exist as a result of the proposed change, the issue would either be resolved deterministically (i.e. maintain the current configuration of the program or feature, or maintain conformance with the licensing bases), or utilize the change evaluation process developed in accordance with RG 1.205 for the new NFPA 805 based standard fire protection license condition to assess the risk impact of the proposed change to determine if the proposed change could be implemented "as-is" or whether prior NRC approval of the proposed change is required. This process follows the guidance outlined in RG 1.174 which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered.

The following specifically describes the configuration control and quality assurance program of the FPRA portion of the new Fire Protection Program. Configuration control of the FPRA model will be maintained by integrating the FPRA model into the existing processes used to ensure configuration control of the internal events PRA model. This process complies with Section 5 of the ASME Standard for PRA Quality and ensures that Duke maintains an as-built, as-operated PRA model of the plant. The process has been peer reviewed and identified as a 'Strength' in the peer review report. Quality assurance of the FPRA is assured via the same processes applied to the internal events model. This process follows the guidance outlined in RG 1.174 which requires the use of qualified individuals, procedures that require calculations be subject to independent review and verification, record retention, peer review, and a corrective action program that ensures appropriate actions are taken when errors are discovered. Although the entire scope of the formal 10CFR50 Appendix B program is not applied to the PRA models or processes in general, often parts of the program are applied as a convenient method of complying with the requirements of RG 1.174. For instance, the procedure which addresses independent review of calculations for 10CFR50 Appendix B is applied to the PRA model calculations, as well.

With respect to Quality Assurance Program requirements for independent reviews of calculations and evaluations, those existing requirements for Fire Protection Program documents will remain unchanged. Duke specifically requires that the calculations and evaluations in support of the NFPA 805 LAR, exclusive of the FPRA, be performed within the scope of the QA program which requires independent review as defined by Duke Engineering Department procedures.

As recommended by NUREG/CR-6850, the sources of uncertainty in the FPRA were identified and specific parameters were analyzed for sensitivity in support of the NFPA 805 change evaluation process. Specifically with regard to uncertainty (which was not reviewed by the NRC in March of 2008), an uncertainty and sensitivity matrix was developed and included with ONS calculation, OSC-9378, Oconee Fire PRA Summary Report as Appendix D. In addition, sensitivity to uncertainty associated with specific FPRA parameters was quantitatively addressed in ONS calculations, OSC-9518, NFPA 805 FPRA Application Calculation. Both of these calculations have been made available to the NRC. Additional discussions regarding NRC expectations for uncertainty/sensitivity have resulted in additional effort that will be incorporated into an in-process revision of ONS calculation, OSC-9518. While the removal of conservatism inherent in the FPRA is a long-term goal, the FPRA results were deemed sufficient for evaluating the risk associated with this application. While Duke continues to strive toward a more "realistic" estimate of fire risk, use of mean values continues to be the best estimate of fire risk. During the change evaluation process, the uncertainty and sensitivity associated with specific FPRA parameters were considerations in the evaluation of the change in risk relative to the applicable acceptance thresholds. Additionally, some informal uncertainty analyses have been performed subsequent to the LAR submittal and are still subject to change.

ATTACHMENT 1

Arreva Change Number A0000650-03, Rev. 2
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

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Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

BACKGROUND:

The following is from Section 2.1.4.6 of the AREVA NP Inc. (AREVA) Oconee Nuclear Station Appendix R Fire Safe Shutdown Analysis Proposal EDA07-617.0, Revision 0, and dated May 11, 2007.

"In performing the cable selection, it will be assumed that sufficient electrical coordination exists for common power supplies. Associated circuits (downstream) of the coordinated protective devices will not be identified as safe shutdown cables nor included in the cable selection unless the associated device is required to be operable for safe shutdown also. All circuits/cables that are electrically connected to the circuit under analysis will be identified up to a credited isolation device."

This assumption was acknowledged by Duke and thought to be valid at that time so no future action was necessary on the part of AREVA or Duke with respect to breaker/fuse coordination while performing the ONS Fire Safe Shutdown Analysis scope of work. It was subsequently determined that this assumption may not be valid based on experience with similar ongoing Fire Safe Shutdown related work at Duke's McGuire Nuclear Station. As a result, this assumption now needs to be validated.

SCOPE:

Duke has requested the assistance of AREVA to perform a breaker/fuse coordination study, as discussed above, as part of the ongoing Oconee Fire Safe Shutdown Analysis (FSSA) Project and NFPA-805 Transition process in order to validate an assumption that was made as part of the ONS Appendix R deterministic Fire Safe Shutdown Analysis previously performed by AREVA.

In performing cable selection for safe shutdown (SSD) related components, it was assumed during the FSSA that sufficient electrical coordination existed for all associated power supplies. If coordination does not actually exist on a safe shutdown related power supply then further analyses must be performed for those feeders (safe shutdown and non-safe shutdown) which are shown to be uncoordinated. The cables associated with these uncoordinated feeders need to be identified and then routed by Fire Area in order to see if the downstream cabling is routed through a Fire Area where the subject SSD related power supply is credited to achieve safe shutdown. If the power supply is credited in the subject Fire Area this would then result in what is being defined as a "fire area conflict".

The voltage levels included in this scope of work are 6.9KV, 4160V, 600V, 208/120VAC, 250VDC and 125VDC. This proposal will provide pricing for AREVA to perform the "Pass 1" analysis (as defined below) for the safe shutdown related power supplies associated with these defined voltage levels.

The power supplies included in the scope of work at each voltage level will be those safe shutdown related power supplies contained within AREVA Engineering Information Record (EIR) 51-5044354-003 and/or the associated Appendix R Database Management System (ARTRAK). Coordination will be re-validated for any power supplies already addressed in existing Oconee Calculations (e.g. OSC-3120, OSC-1366, etc.) by inclusion in the new AREVA calculation or by referencing the existing Duke calculations.

Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

METHODOLOGY:

In order to determine if coordination exists under fire related circumstances (bolted fault conditions); multiple levels of analysis or "Passes" may be required. As each "Pass" progresses the level of detail will increase until coordination is proven to be successful (when it is required) or ultimately ruled out. The requirement here is not to protect the cabling or the load, but to ensure that the first in-line isolation device clears a fire induced bolted fault without tripping the upstream breaker (isolation device) whereby power would be lost to the entire SSD bus and therefore all safe shutdown related loads.

All of the "Passes" will be based on a comparison of the Time Current Characteristic (TCC) curve of the subject compartment/branch isolation device (e.g. breaker, fuse or protective relay) and the TCC curve of the isolation device immediately upstream.

"Pass 1" will determine if the protective device is coordinated in general, for all credible levels of fault downstream taking into account the maximum available short circuit current adjacent to the bus (e.g. switchgear, load center, motor control center, panel board). Therefore, this "pass" does not take into account any specific cable length. This "pass" is done for all circuits connected to safe shutdown related power supplies under normal plant operating conditions. If "Pass 1" is successful then no further action is necessary (coordination is validated). If 'Pass 1" is not successful additional "Passes" may be required to justify coordination in a given plant Fire Area. These additional "Passes" would be addressed under Phase 2 of this study and are therefore not included in this proposed scope and cost estimate at this time. Phase 2 estimates will be provided under a separate scope as needed, once the "Pass 1" analysis is complete.

These additional 'Passes" would take into account the associated cabling attributes (size, length, type, impedance, etc.) in order to determine the maximum available Short Circuit Current (SSC) at a specific cable point downstream of the subject isolation device to see if coordination exists. Phase 2 of the coordination study would address any analysis required beyond "Pass 1" as needed to support the fire safe shutdown analysis at Oconee.

The Star Module of ETAP will be utilized to produce the Time Current Characteristic curves to support the analysis discussed above. The ETAP Star Module contains over 100,000 verified and validated device libraries. Duke is already a licensed user of this Verified and Validated Software. For each of the TCCs generated, a "single line" will be created to show each of the plotted protective devices and applicable fault current being evaluated to support the "Passes" described above. The fault currents and supporting data used to determine coordination will be provided in an Excel spreadsheet format for each power source, and will allow ease of maintenance for future changes. The fault current for each power source can be manually 'inserted' into the Star Module and the spreadsheet. Fault currents for applicable downstream loads will be generated via existing spreadsheet formulae.

An ETAP load flow model will not be created for the support of this project, including interconnection of the low voltage electrical system with the existing medium voltage distribution system buses currently in Duke's ETAP model. Additionally, cable protection and respective load information will not be included/addressed since it is not required for this specific scope of work.

Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

The methodology presented above is the same as that being used at McGuire for similar fire safe shutdown related work. If requested by Duke at the conclusion of the project, a one day training session will be provided by AREVA to Duke Personnel on the proper method of maintaining the project related deliverables.

ASSUMPTIONS:

- Duke will supply the available (maximum) short circuit fault current available at each safe shutdown bus/power supply as required to support the analysis down to the 600V Motor Control Center level and the DC system distribution panels. For the DC system, the available fault currents at the buses directly connected to the batteries shall be provided in order to provide a starting point basis for downstream calculations, if required. If the short circuit data needed for this project is beyond what is available in existing Duke calculations, AREVA will identify this to Duke and then provide a subsequent change order for the added scope and associated additional costs to generate the required data.

- Based on deterministic Appendix R requirements, each "pass" will be done for circuits connected to safe shutdown related power supplies under "Plant Normal Operating Conditions" at 100% power. Deign Basis Events and single failure criteria do not need to be considered. Normal Plant Operations which would include any auto-transfers and also any manual transfers that occur when swapping to the SSF for fire related scenarios will be included. Other alignments that are done manually for other purposes will not be included based on this criterion.

- Duke will provide a "Key" for the ETAP software (including Star Module) to be utilized for preparing the Time Current Characteristic curves for this analysis throughout the project.

- An Oconee ETAP Model exists for the AC Distribution System and a copy will be provided to AREVA as "Information Only" for use during this project. This data will only be used to help expedite the development of the buses & components within in the STAR Module.

- All Duke Databases and controlled documents are considered complete, current (as-built) and accurate. Any valid input from Duke controlled design verified inputs (e.g. calculations) will be utilized when available for direct input into this project.

- No plant walkdowns by AREVA are included in the project scope.

- Breaker model types are assumed to be available in the Duke Databases or one-line diagrams. Breaker/relay settings that are not available in the Duke Databases or documents (Bill of Materials or Plant Procedures) will be assumed to be set at their respective maximum (or minimum) setting for conservatism. However, AREVA will identify to Duke where settings can not be obtained from documents in order to allow Duke the opportunity to capture this data in the field to be provided to AREVA formally as design input into the project.

Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

- Component Time Current Characteristic Curves (TCCs) will be utilized from the ETAP Star Module and do not require verification. When multiple versions of a TCC are available in ETAP the most conservative TCC will be used. Bill of Materials and other Duke documents will not be searched in order to identify specific *vintages* of breaker or fuse TCCs unless the specific curve type is unavailable in ETAP.

- There will not be any adjustments to the TCCs for temperature correction factors. This will be included in the calculation as an assumption with adequate basis provided as required by Duke Procedure EDM-101. If for any reason Duke feels that the basis is not adequate enough, then AREVA will provide a subsequent change order for the added scope and associated additional costs to satisfy the procedural requirements.

- Thermal overload protection of motor (or other) loads will not be plotted or considered.

- Inverter panelboards are not included in this scope. The load breakers on the 120VAC inverter panelboards are assumed to not require review because the inverter limits output current to a value lower than that required to trip one of the source breakers.

- This proposal does not included resolution (modifications, evaluations, etc.) of any unacceptable/undesired results identified during this project.

- Any subsequent changes required to the ARTRAK Database or ONS FSSA Engineering Information Record (EIR) 51-5044354 previously prepared by AREVA as a result of new breaker/fuse coordination study results are not included in this scope and cost estimate.

- Any revisions required to existing Duke Electrical Calculations (e.g. OSC-3120, OSC-1366, etc.) identified during or resulting from the results of this study are not included in this scope and cost estimate.

- Multiple High Impedance Fault (MHIF) Analyses are not included in this scope. This remains as discussed in original AREVA Proposal EDA07-617.0, Revision 0, dated May 11, 2007.

- All other terms and conditions of the original AREVA Proposal EDA07-617.0, Revision 0, dated May 11, 2007 still apply, with the exception of the T&M labor rates which are defined below.

- Any design inputs (not contained within a controlled document) supplied by Duke for input into this project must be provided by formal correspondence (i.e. company letterhead, not by email) and signed by an authorized representative and be in accordance with Duke Procedure EDM-101.

- The cost estimate for this scope assumes one consolidated set of comments from the Duke Energy responsible engineer. Turn around time is assumed to be 4 weeks for each deliverable submitted.

Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

- Pricing for these tasks assumes that no deliverable will require a 10CFR50.59 Safety Evaluation.

- AREVA NP assumes that ONS acceptance of the final deliverable and turnover of project records establishes the basis for the completion of services and project closure.

Attachment 1 (AREVA Change No. A0000650-03, Revision 2)
Oconee Appendix R Fire Safe Shutdown Analysis (FSSA) Update Project
Breaker/Fuse Coordination Study Phase 1 (Pass 1 only)

DELIVERABLES:

The analysis and associated results for "Pass 1" along with any subsequent "Passes" required in Phase 2 (as discussed above), will be documented in a single AREVA Calculation. The calculation will describe the methodology, assumptions, results and conclusions for all aspects associated with this scope of work and include applicable attachments. A project kickoff meeting will be scheduled for stakeholders prior to the start of project; in addition, a pre-job brief will be held to review methodology. A review of the calculation methodology and overall progress/status will be conducted with Duke Personnel at approximately 25%, 50% and again at approximately the 75% completion points of the preparation phase of the project, to ensure that the Duke team is aligned with the details of the analysis and that the final deliverable will meet Duke's requirements.

This project is estimated to be completed on or before December 31 2009. Based on the results of Phase 1 and subsequent work required in Phase 2, this date may need to be extended. A project kickoff meeting will be held near the start of the project between Plant Personnel (Electrical & Safe Shutdown), Duke Corporate Group Personnel (PSA & NFPA-805) and AREVA.

This task will be performed as safety-related in accordance with the AREVA Quality Assurance Program. The deliverables will be formatted in accordance with Duke Power Procedure EDM-101, Engineering Calculations/Analyses, in order to facilitate a seamless transition into the ONS Document Management System.

ATTACHMENT 2

MODIFICATION SCOPING

ATTACHMENT 2
Modification Scoping

Attachment S, Plant Modifications of the Transition Report (TR) was submitted as Enclosure 2 in the NFPA 805 License Amendment Request (LAR) dated October 31, 2008. In a supplement to the LAR dated May 31, 2009, Duke Energy Carolinas, LLC (Duke) provided scoping information for the following Attachment S modifications:

- Protected Service Water
- Main Steam Isolation Valves
- Turbine Building/Auxiliary Building Wall upgrade to a 3-hour barrier
- Purge Inlet Room/Auxiliary Building Wall upgrade to a 3-hour barrier
- U1/2 Blockhouse Tornado Vent Dampers

Duke also committed in the May 31, 2009 supplement to provide the following in this supplement:

- Scoping for RCP spurious start and ability to ensure pump trip
- Scoping for HPI Pressure pump spurious start and ability to ensure pump trip
- Scoping for spurious operation of the RCS vent valves
- Revision to Attachment S, Plant Modifications

Scoping for spurious operation of the RCS vent valves is provided below. Other modifications and revision to Attachment S, Plant Modifications are dependent upon resolution of issues with FAQ 07-30, (OMA Transition to Recovery Actions). These issues were discussed and feedback requested from the NRC in a conference call on July 10, 2009.

Spurious Operation of RC Head and Loop Vent Valves:

A hot short in Unit control boards could cause the spurious operation (opening) of the RC Head and/or Loop Vent valves due to fire induced failure. A hot short could occur on the conductor between the valve control switch and the solenoid valve. This would spuriously energize the solenoid valve causing the valve to open (vent). This can occur even though these valves are de-energized (breaker open) during normal operation.

A "short to ground contact" will be added on the existing "power on" control switch in the solenoid valve circuit to ensure the solenoid remains de-energized in a hot short scenario. The circuit will provide a shorting contact which when closed creates a ground path to the neutral leg of the solenoid and thereby will allow the solenoid to remain de-energized (closed). In accordance with the guidance in NUREG/CR-6850 an open circuit is not considered as a primary cable failure therefore the coil would remain shorted out during a hot short scenario. The "short to ground contact" will be closed during normal power operations. It will open when the "power on" control switch is manually turned to "on" and thereby will allow the solenoid to be energized (open) when when the valve is required open.

Attachment 2
Modification Scoping

This modification is modeled in the FPRA and will provide risk reduction. TR, Attachment C and S will be revised to reflect the risk reduction accordingly.

The RC Head and Loop Vent Valve modifications will be completed during refueling outages on Unit 1 in the fall of 2012, Unit 2 in the fall of 2011, and Unit 3 in the spring of 2012.

ATTACHMENT 3

REGULATORY COMMITMENTS

Attachment 3
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
Revise the Transition Report submitted October 31, 2008 to reflect the following (RAI 3-1): Rewrite of Attachment A, Table B-1 to include the following: • Enhanced wording for clarity where prior approval is cited and modifications are referenced. Clarification will be provided where modifications for compliance are referenced in order to clearly indicate what modifications will occur. (RAI 2-2) • Include the code years evaluated. (RAI 2-4) • Clearly indicate ONS compliance status. (RAI 2-5) • A review of all areas with suspended ceilings will be performed and wiring, if present, dispositioned. (RAI 2-5) • The reference to the propane tank and its orientation will be removed. (RAI 2-5) • All 'Further Actions Required' statements will be reviewed to ensure that those actions that are required to demonstrate compliance are retained as include in LAR 'Yes' and include the commitment in the LAR. Where necessary the compliance statement and compliance basis sections will be updated. (RAI 2-5) Attachment C, B-3 Table will be reviewed and updated accordingly to clarify the lack of suppression in a III.G.3 area. (RAI 2-3) Attachment G - Operator Manual Actions will be revised as follows: • The term challenging active fire was used incorrectly. It should have stated ... "Embedded in the compliance strategy is the assumption that the 10 minute time frame does not start until confirmation of an active fire." (RAI 3-1) • A revised list of recovery actions for each fire area will also be included. The completion date for this document is also contingent upon the pending outcome of FAQ 07-30 and its impact on change evaluations, recovery actions, and required plant modifications. (RAI 3-10) Attachment J – EEEE Transition will be revised to remove all engineering evaluations currently listed. The engineering	November 30, 2009

Commitment	Due Date
evaluations currently listed in Attachment J are allowed and do not need to be submitted in the LAR for NRC approval. (RAI 2-6) Attachment L will be revised to reflect completion of the NFPA code compliance reviews. (RAI 2-4) Attachment P will be deleted, along with reference to P-2. (RAI 2-4) Attachment S will be revised to reflect correct modifications. Attachment T will be revised as follows: • Prior Approval Clarification Request 2 will be revised to eliminate BOP fire area and discuss the new fire areas (AB and TB. (RAI 2-3) • To request that the NRC formally document as a "prior approval" recognition that within the first 10 minutes following the identification of a confirmed active fire, fire growth will not reach a point where fire damage will (RAI 3-1): • Result in spurious equipment operation • Result in a loss of offsite power condition • Preclude operation of plant equipment from the control room Attachment U will be revised to remove references to DA-A1a and HR-A1. (RAI 5-10	
Implement modifications to address spurious operation of the RCS vent valves (Attachment 2)	U2EOC25 - fall 2011 U3EOC26 - spring 2012 U1EOC27 - fall 2012
Complete breaker coordination study (RAI 3-3)	June 30, 2010
Implement modifications to address RCPs spurious start and ability to ensure pump trip – Scoping is not complete. (Attachment 2)	Pending resolution of FAQ 07-30
Implement modifications to address HPI Pressure pump spurious start and ability to ensure pump trip – Scoping is not complete. (Attachment 2)	Pending resolution of FAQ 07-30
Response to RAIs 5-15 and 5-16	Pending resolution of FAQ 07-30
Complete code compliance reviews and identify required modifications in Attachment S. (RAI 2-4)	November 30, 2009
IN 92-18, "Potential for Loss of Remote Shutdown Capability During a Control Room Fire," Study results (RAI 1-1)	TBD

Attachment 3
Regulatory Commitments

Commitment	Due Date
Finalize preliminary answers to questions based on completion of OSC-9518, Oconee Fire PRA Application Calculation (RAI 1-2, 5-5, and 5-11 responses)	September 30, 2009
Duke will provide a detailed overview of the monitoring program (RAI 6-1 Response).	September 30, 2009