FORM 805.3 (R01-08)

CERTIFICATION OF ENGINEERING CALCULATION - REVISION LOG

Station And Unit Number: Oconee Units 1, 2, and 3

Title Of Evaluation: NFPA 805 Transition Risk-Informed, Performance-Based Change Evaluation Methodology

Evaluation Number: OSC-9314

Active Evaluation / Analysis: Yes ☐ No ☐

	Evaluation Pages (Vol)			Supporting Documentation (Vol)			Volumes		Orig	Chkd	Verif. Meth.	Appr[1]	Issue Date
Rev. No.	Revised	Deleted	Added	Revised	Deleted	Added	Deleted	Added	Date	Date	1, 2, 3, "Other"	Date	Rec'd Date

Note 1: When approving an Evaluation revision with multiple Originators or Checkers, the Approver need sign only one block.

CALCULATION IMPACT ASSESSMENT (CIA)

ation / Unit	Oconee Units 1, 2, & 3	Calculation No.	OSC-9314	Rev.	0 Page
P No. (if applicable)	O-08-02520	By	Ira M. Heatherly	Date	10/28/08
Prob. No. (stress & s/r use only)	N/A	Checked By		Date	10/28/08

NEDL reviewed to identify calculation? (formally SAROS) ☐ YES ☒ NO

Note: A NEDL search is NOT required for calculation originals (i.e. Rev. 0's)

Identify in the blocks below, the groups consulted for an Impact Assessment of this calculation origination/revision.

	Indiv. Contacted/Date		Indiv. Contacted/Date
☐ RES [Power, I & C, ERRT, Reactor]		☒ NGO [QA Tech. Services (ISI), Severe Accident Analysis, Elect. Sys. & Equip., Design & Reactor Supp., Civil Structural, Core Mech. & T/H Analysis, Mech. Sys. & Equip., Nuclear Design, Safety Analysis, and Matls/Metallurgy/Piping]	
☒ MCE [Primary Systems, Balance of Plant, Rotating Equipment, Valves & Heat Exchangers, Civil]	Steve Jarrett* Harold Lefkowitz*	PRA	Brandi Weaver*
☐ MOD [Mechanical Engr., Electrical Engr., Civil Engr.]		☐ Training	
☒ Operations - OPS Support	Cam Eflin*	☐ Local IT	
☐ Maintenance - Tech. Support		☐ Regulatory Compliance	
☐ Work Control - Program. Supp.		☐ Chemistry	
☐ Other Group		☐ Radiation Protection	
		☐ **No Group required to be consulted**	

**** - These consultations occurred as part of the development of the NFPA 805 Transition LAR.***

Listed below are the identified documents (ex: TECHNICAL SPECIFICATION SECTIONS, UFSAR SECTIONS, DESIGN BASIS DOCUMENTS, STATION PROCEDURES*, DRAWINGS, OTHER CALCULATIONS, ETC.) that may require revision as a result of the calculation origination or revision, the document owner/group and the change

DOCUMENT	GROUP	CHANGE REQUIRED
OSC-9292	MCE	No change required
OSC-9518	NGO	No change required
OSC-9375	NGO	No change required
PIP O-08-02520	MCE	No change required

(Attach Additional Sheets As Required)

1.0 PROBLEM STATEMENT

1.1 DESCRIPTION OF THE PROPOSED CHANGE

During the transition to NFPA 805 variances from the deterministic requirements of NFPA 805 or with the current licensing basis will be evaluated using the 'Change Process' outlined in NEI 04-02. These variances are based on the regulations that were applicable to Oconee Nuclear Station (ONS) prior to the transition to a 10 CFR 50.48(c) FPP and would be limited to cases where the nuclear safety performance criteria, also referred to as fire protection program variances, of NFPA 805 are not met

During the transition to NFPA 805, a change evaluation calculation will be performed for each fire area containing variances from the deterministic requirements of the current licensing basis.

Change evaluations will result in a demonstration that the variances are compliant with the acceptance criteria of NFPA 805 and therefore are in compliance with the NFPA 805 performance based approach.

The purpose of this calculation is to document the methodology to be used in the performance of the change evaluations created for the NFPA 805 Transition.

1.2 CALCULATION CLASSIFICATION

This calculation provides a methodology to be used in support of the transition of the Fire Protection Safe Shutdown Program in accordance with 10CFR50.48(c). Since the transition documentation generated using this methodology is considered QA-1 (see Enclosure, Section 1.2 for additional information) this calculation is classified as QA-1.

2.0 DESIGN METHOD/ANALYTICAL METHOD

2.1 BACKGROUND

The methodology for the change evaluations is based on the guidance in NFPA 805 and Section 5.3 of NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 1, as modified by Frequently Asked Questions (FAQs) 06-0002 and 06-0003, and Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants.

Section 5.3 of NEI 04-02 provides requirements and guidance for performing risk-informed, performance-based change evaluations. Frequently Asked Questions (FAQs)[1] 06-002 and 06-003 provided clarifications and minor modifications on the process. Nuclear Regulatory Commission (NRC) clarification of the change evaluation process is provided in Section C.3.2 of Regulatory Guide 1.205. The change evaluation process consists of the following subtasks (Figure 1 depicts the NEI 04-02 process as envisioned for post transition. Figure 2 reflects the process to be used during transition.):

[1] Both the NRC and the industry recognized the need for additional clarifications to the guidance provided in RG 1.205, NEI 04-02, Revision 1, and NFPA 805. The NFPA 805 FAQ process was jointly developed by NEI and NRC to facilitate timely clarifications of NRC positions.

- Defining the Change
- Preliminary Risk Screening
- Risk Evaluation
- Acceptance Criteria Assessment

In addition, the fire protection features and systems relied upon in the change evaluations will be assessed for inclusion into the monitoring program.

The following subsections describe the methodology used to evaluate the acceptability of this risk-informed, performance-based change evaluation.

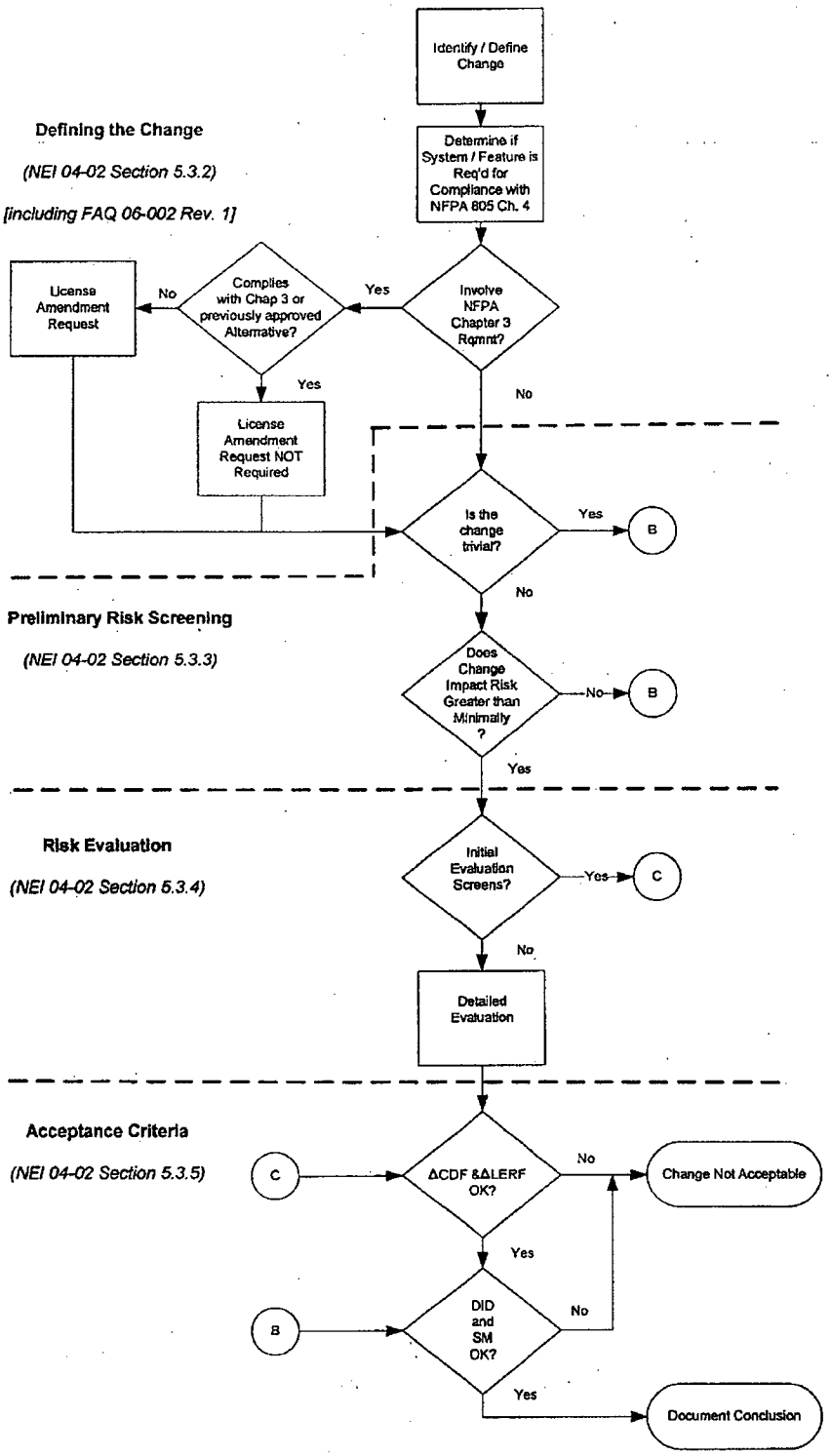


Figure 1 – Change Evaluation Process (based on NEI 04-02 Figure 5-1)

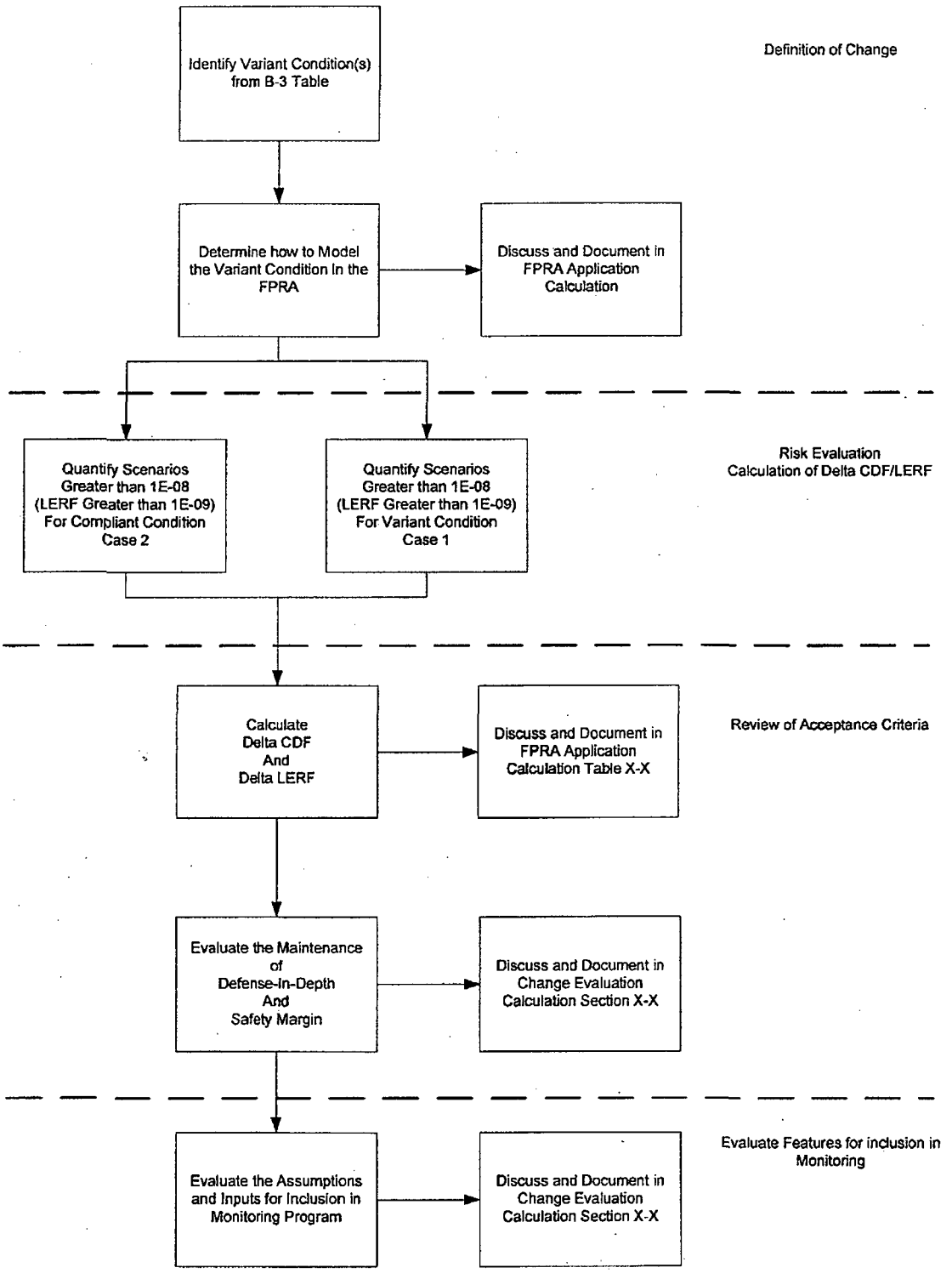
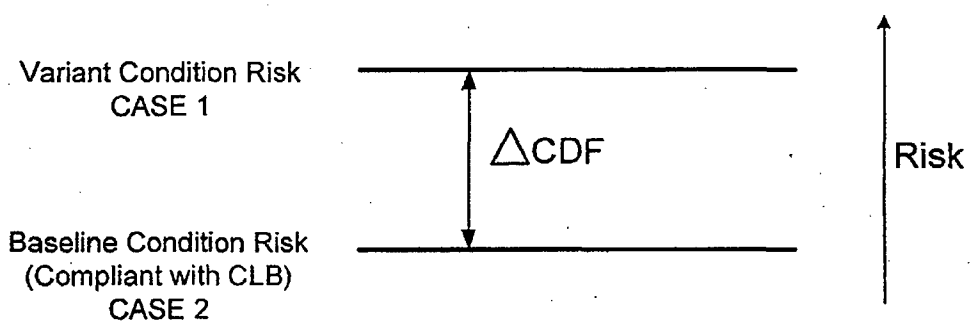


Figure 2 – Change Evaluation Process Modified for Transition

2.2 DEFINING THE CHANGE

The Change Evaluation process begins by defining the variant condition to be examined and the baseline (compliant) configuration as defined by the Licensing Basis (current pre-transition licensing basis). Refer to NEI 04-02 Sections 5.3.2 and J.2 for more detail.

The Baseline Condition is defined as that plant condition or configuration that is consistent with the Licensing Basis (pre-transition licensing basis, shown as Case 2 in Figure 1). The variant condition or configuration, either 'as found' or proposed by a plant change, that is not consistent with the Licensing Basis is defined as the Variant Condition (shown as Case 1 in Figure 3).



$$\Delta CDF = CDF_{CASE\ 1} - CDF_{CASE\ 2}$$

Figure 3 – Baseline versus Variant Conditions

The 'changes' associated with NFPA 805 transition are those fire protection program variances that will be addressed using the NFPA 805 performance-based approach (via the change evaluation process). These changes are identified in Calculation OSC-9292, NFPA-805 Transition B-3 Table/Report, as Open Items requiring 'change evaluations'.

Due to the nature and complexity of individual changes that will be addressed as part of the transition change evaluation, it is necessary to organize and group individual changes. To the maximum extent possible, variances being addressed by change evaluations should be organized by plant location (i.e., fire area). The rationale for this grouping is:

- Key analytical tools for measuring compliance (e.g., safe shutdown analyses for 10 CFR 50) are organized in this manner. This will facilitate the clear documentation of a 'compliant case' for use in the change evaluation.
- Analytical tools for measuring fire risk (i.e., Fire PRA) are primarily spatially oriented and can focus analyses on specific targets and scenarios.
- This grouping supports reporting requirements for the NFPA 805 transition License Amendment Request (LAR) in Regulatory Guide 1.205.

The definition of the change should include the following information:

- Background – A short discussion of the performance goal affected by the change evaluation.
- Description of the Variant Condition – describe the component/cabling or components/cabling that cause the variant condition.

- Description of how the variant condition will be modeled in the Fire PRA (FPRA) – describe how and why the variant condition will be modeled in the FPRA, including the basic events that will be manipulated. If the variant condition is not modeled in the FPRA, the rationale for exclusion should be noted.

2.3 PRELIMINARY RISK SCREENING

Once the definition of the change is established and groupings/organizations are completed, a preliminary risk screening may be performed to identify and resolve minor changes to the fire protection program. Refer to NEI 04-02 Sections 5.3.3 and J.3 for additional detail.

It is not expected that the reviews associated with change evaluations during the transition to NFPA 805 will identify trivial or editorial changes or those that have minimal risk impact (since the focus of the transition change evaluation is to address variances from the current licensing basis). However, a record of changes that were screened from detailed evaluation should be developed for traceability.

2.3.1 SCREENING OF TRIVIAL OR EDITORIAL CHANGES

The Evaluator shall determine if the change is trivial, based upon guidance in NEI 04-02. Examples include:

- Changes to titles in procedures or program documents
- Change to Fire Brigade Training facility that has no impact on established training scenarios
- Changes to the Combustible Control Form that does not affect content.
- Changes to document layout.
- Changes to document numbers.

During the initial transition to NFPA 805, no open items were screened using the trivial or editorial criteria summarized above.

2.3.2 SCREENING OF MINIMAL RISK IMPACT CHANGES

If the change is determined not to be trivial, the Evaluator shall perform a preliminary risk review, using the guidance in NEI 04-02, Appendix I, Attachment 3. If any of the preliminary risk review questions have "potentially greater than minimal" impact, a detailed quantitative risk evaluation is required.

During the initial transition to NFPA 805, no open items were screened using the minimal risk criteria summarized in the referenced NEI 04-02 guidance.

2.4 RISK EVALUATION

The FPRA is adequate to support the NFPA 805 change evaluation process. In lieu of a peer review, the NRC conducted of review of the ONS FPRA March 17-21, 2008. The NRC review, which was required by Regulatory Guide 1.205, noted that there were a number of incomplete supporting requirements communicated as 'findings'. Most of the findings from the NRC review have either been addressed or deemed to have no impact on FPRA quantification. Additionally, the FPRA has been found to meet Capability Category II in most but not all cases; those areas where the Capability Category II requirement was not met were evaluated and found to have no

impact on the application. Finally, the unresolved open items and any identified uncertainties with potential impact on the FPRA results have been addressed in the sensitivity analysis. Based on the evaluations performed and the sensitivity analyses (Refer to OSC-9518, ONS NFPA 805 Application Calculation), it was concluded that the FPRA was adequate for use in the change evaluation process.

It is noted that only the Unit 3 FPRA was reviewed by the NRC. That review is considered sufficient for Units 1 and 2 since the approach applied for development of the Unit 3 FPRA was not significantly altered. For Unit 2, a separate fault tree model was developed and a quantification file was built using the methodology contained in the Unit 3 FPRA documentation file. For Unit 1, a comparative screening analysis was performed which demonstrated that a separate Unit 1 fault tree and FPRA quantification file are not necessary.

The Evaluator shall coordinate as necessary with the Safe Shutdown Engineer, Fire Protection Engineer and Fire PRA Engineer to perform/revise the calculations to assess the change using risk-informed, performance-based techniques (including, but not limited to fire modeling and PRA). The risk evaluation may be in the form of a limiting or bounding fire modeling/fire risk analysis or a detailed integrated analysis. Refer to NEI 04-02 Sections 5.3.4, J.4, and J.5 for more detail.

The risk evaluation, as discussed in NEI 04-02 may consist of a number of steps

- Initial Evaluation (Sections 5.3.4.1, 5.3.4.2, J.4.1, and J of NEI 04-02)
  - Fire Modeling
  - Bounding Risk Assessment
- Detailed Risk Evaluation (Sections 5.3.4.3 and J.5 of NEI 04-02)

This step as outlined in Section 5.3.4.1 and 5.3.4.2 of NEI 04-02 can be used to perform bounding fire modeling and bounding risk assessments. This step can be performed in order to optimize the effort necessary to successfully document the change evaluation.

If the Initial Evaluation cannot demonstrate the acceptability of the change, a detailed Risk Evaluation shall be performed. This is discussed further in Appendix J of NEI 04-02.

If the change involves equipment/cables required only for cold shutdown or whose function is not modeled in the PRA, then a qualitative risk assessment will be performed. This qualitative assessment should include the following:

- The desired safe end state for the traditional treatment of post fire safe shutdown under the provisions of 10 CFR 50.48(b) is cold shutdown. The transition to invoke the provisions of 10 CFR 50.48(c) includes the use of a Fire PRA. The safe end state evaluated in a PRA is not cold shutdown, but is instead a condition characterized as "safe and stable".. This is typically hot standby/shutdown conditions. The PRA treatment of the plant response to a fire event does not necessarily require or credit the use of plant systems exclusive to cold shutdown. As such, the treatment of any such systems and functions in the context of a change evaluation would generally result in no measurable impact on the calculated plant risk.
- There are however, some possible exceptions.

- If the fire induced plant transient is of such a nature that in order to achieve safe and stable conditions, cold shutdown related systems and/or functions are required, the PRA would inherently require those functions to be successful. In these cases, the calculated risk metrics for the postulated fire event includes the consideration of failures that would disable there systems and/or functions, or
- If the variance would affect achievement of a key safety function during a non-power higher risk-evolution, then deterministic options should be considered in accordance with the non-power operations methodology.

The following programmatic level items have the potential to impact the results of the FPRA and are included here due to their potential for impacting the results contained in OSC-9518, ONS NFPA 805 Application Calculation.

- The breaker coordination analysis for Oconee is not complete, and therefore the extent to which this may increase conditional core damage probability (CCDP) is currently unknown. The current FPRA does not assume any breaker coordination issues.
- For risk reduction, the baseline Fire PRA includes the Protected Service Water and the 3TC/3TD cable reroute/protection modifications.
- The Unit 3 PRA model has known issues that have been identified and are being tracked in the PRA Tracker item database. These model discrepancies have been assessed and do not affect the final delta CDF or delta LERF results.
- The review of the Unit 3 PRA model also included items issues related to application of the Regulatory Guide 1.205. These items have been assessed and do not affect the final delta CDF or delta LERF results.
- As part of the ongoing pilot plant effort, various topics needing clarification or additional guidance are being addressed through the Frequently Asked Questions (FAQ) process. Several of these items have the potential to directly or indirectly affect the FPRA and/or the change evaluation process. FAQs have been incorporated into the change process to the extent practical. A discussion of methods that may be considered beyond those contained in NUREG/CR-6850 is included in the Oconee Fire PRA Scenario Development calculation, OSC-9375, Revision 0.

2.5 ACCEPTANCE CRITERIA ASSESSMENT

For the transition to NFPA 805, the overall acceptance of the transition change evaluation will be in the form of a license amendment per 10 CFR 50.90, as required by 10 CFR 50.48(c)(3)(i). Refer to NEI 04-02 Sections 5.3.5 and J.6 for more detail. Acceptance criteria for fire area change evaluations are based on ensuring:

- Quantitative Risk Acceptance Criteria
- The change in core damage frequency (ΔCDF) is acceptable, and
- The change in large early release frequency (ΔLERF) is acceptable, and
- Defense-in-depth is maintained
- Safety margins are maintained

The acceptance criteria for the change evaluation consist of two parts. One is quantitatively based and the other is qualitatively based. The quantitative figures of merit are ΔCDF and ΔLERF. The qualitative factors are defense-in-depth and safety margin. If a change meets the acceptance criteria described below, this is confirmation that a success path effectively remains free of fire damage (Ref. NEI 04-02, Sections 5.3.4.2 and 5.3.5).

QUANTITATIVE RISK ACCEPTANCE CRITERIA

The transition risk evaluation shall be measured quantitatively for acceptability using the ΔCDF and ΔLERF criteria from Regulatory Guide 1.174, as clarified in Section 5.3.5 of NEI 04-02 and Regulatory Guide 1.205. The results of the acceptability determination shall be clearly documented in the calculations/analyses.

From an overall transition perspective, the acceptance criteria are provided in Regulatory Guide 1.205, Section C.2.2:

> *"The total change in risk associated with the transition to NFPA 805 should be consistent with the acceptance guidelines in Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis."*

The acceptance criteria of Regulatory Guide 1.174 are summarized in the following table.

Region	ΔCDF/yr	ΔLERF/yr	Status	Comments/Conditions
I	≥ 1.0E-05	≥ 1.0E-06	Unacceptable	Proposed changes in this region are not acceptable.
II	< 1.0E-05 and ≥ 1.0E-06	< 1.0E-06 and ≥ 1.0E-07	Acceptable w/ conditions	Proposed changes in this region are acceptable provided the cumulative total CDF from all CDF initiators is less than 1.0E-04/yr and from all LERF initiators is <1E-5/yr. Cumulative effect of changes must be tracked and included in subsequent changes.
III	< 1.0E-06	< 1.0E-07	Acceptable w/ conditions	Proposed changes in this region are acceptable provided the cumulative total CDF from all initiators is less than 1.0E-03/yr and from all LERF initiators is <1E-4/yr. Cumulative effect of changes must be tracked and included in subsequent changes.

If the risk evaluation determines that ΔCDF and ΔLERF are acceptable and that defense-in-depth and safety margins are maintained, then the Evaluator shall document the results. This is confirmation that a success path effectively remains free of fire damage.

If the risk evaluation determines that either ΔCDF or ΔLERF are not acceptable, then the Evaluator shall document that the results are not acceptable and alternatives should be pursued until the quantitative acceptance criteria are met.

OSC-9518, ONS NFPA 805 FPRA Application Calculation, contains the methodology associated with calculating these values.

DEFENSE-IN-DEPTH

A review of the impact of the change on defense-in-depth shall be performed, using the guidance below from NEI 04-02. NFPA 805 defines defense-in-depth as:

- Preventing fires from starting
- Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting damage
- Providing adequate level of fire protection for structures, systems and components important to safety; so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

In general, the defense-in-depth requirement is satisfied if the proposed change does not result in a substantial imbalance among these elements.

The review of defense-in-depth is typically qualitative and should address each of the elements with respect to the proposed change. Defense-in-depth may be assessed at a compartment, fire scenario, or fire area basis if applicable to multiple changes.

Fire protection features and systems relied upon to ensure defense-in-depth should be clearly identified in the assessment (e.g., detection, suppression system).

Reliance on OMAs (recovery actions) in lieu of protection is considered part of the third echelon of defense-in-depth.

> **NOTE:** Per NFPA 805, recovery actions are defined as: "Activities to achieve the nuclear safety performance criteria that take place outside of the main control room or outside of the primary control(s) station for the equipment being operated, including the replacement or modification of components."

Consistency with the defense-in-depth philosophy is maintained if the following acceptance guidelines, or their equivalent, are met:

1. A reasonable balance is preserved among 10 CFR 50.48(c) defense-in-depth elements.
2. Over-reliance and increased length of time or risk in performing programmatic activities to compensate for weaknesses in plant design is avoided.
3. Pre-fire nuclear safety system redundancy, independence, and diversity are preserved commensurate with the expected frequency and consequences of challenges to the system and uncertainties (e.g., no risk outliers). (This should not be construed to mean that more than one safe shutdown train must be maintained free of fire damage.)
4. Independence of defense-in-depth elements is not degraded.
5. Defenses against human errors are preserved.
6. The intent of the General Design Criteria in Appendix A to 10 CFR Part 50 is maintained. For ONS the applicable GDC are contained in UFSAR, Chapter 3.

It should be noted that all elements of fire protection defense-in-depth may not exist for beyond design basis fire scenarios. For example, a conditional core damage probability of 1.0 is possible if enough fire barriers are breached. Such beyond design basis scenarios, however, should be demonstrated to be of less risk significance, with certainty. A scenario that does not lead to core damage and a CDF of 9E-08/year would be treated differently than a scenario that leads to core damage and a CDF of 9E-08/year. In the end, the balance results in consideration of all aspects of the scenario, including the risk, defense-in-depth, safety margins, uncertainty, and other relevant issues.

SAFETY MARGIN

A review of the impact of the change on safety margin shall be performed. An acceptable set of guidelines for making that assessment is summarized below. Other equivalent acceptance guidelines may also be used.

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

The requirements related to safety margins for the change analysis is described for each of the specific analysis types used in support of the fire risk assessment.

These analyses can be grouped into three categories. These categories are:

- Fire Modeling
- Plant System Performance
- PRA Logic Model

The following guidance on these topics is provided. Additional information is contained in NEI 04-02 Section 5.3.5.3.

Fire Modeling

A combined analysis approach will be used during the transition change evaluations and therefore maximum expected fire scenario (MEFS)/limiting fire scenario (LFS) will not be analyzed separately. From a safety margin perspective the evaluation of the fire modeling performed should address the following topics:

- Was the fire modeling performed using codes/methods acceptable to the NRC?
- Were the heat release rates used conservative?
- If the treatment in the FPRA, resulted in no damage to the targets of concern, and an LFS is not calculated, ensure that over reliance is not placed on the third element of defense-in-depth (i.e., operator manual actions).

Plant System Performance

The development of the fire risk assessment may involve the re-examination of plant system performance given the specific demands associated with the postulated fire event. The methods, input parameters, and acceptance criteria used in these analyses need to be reviewed against that used for the plant design basis events. This review would serve to establish that the Safety

Margin inherent in the analyses for the plant design basis events have been preserved in the analysis for the fire event and therefore satisfy the requirements of this section.

From a safety margin perspective the evaluation of the plant system performance should address the following topics:

- Were input parameters for plant performance analyses (e.g., heat transfer coefficients, pump performance curves) altered from those used for plant design basis events such that margin was lessened?
- Were codes and standards used to determine plant system performance acceptable to the NRC?

PRA Logic Model

The quantification for fire related CDF/LERF relies upon the FPRA model. It is recognized that use of a FPRA often requires model modifications to be performed to the internal events PRA. These modifications may include altering basic event failure probabilities, adding basic events, and logic structure changes. These changes should be evaluated against the methods and criteria for the overall FPRA model development for consistency, or confirmation of bounding treatment, to confirm that the Safety Margin inherent in the PRA model is preserved.

From a safety margin perspective the evaluation of the PRA logic model should address the following topic:

- Were the risk-informed, performance based processes utilized based upon NFPA 805, 2001 edition, endorsed by the NRC in 10 CFR 50.48(c)?
- Was the change evaluation process in accordance with NEI 04-02, Revision 1, which is endorsed by the NRC in Regulatory Guide 1.205, Revision 0?
- Was the FPRA developed in accordance with NUREG/CR-6850, which was developed jointly between the NRC and EPRI?

2.6 MONITORING OF CREDITED SYSTEMS, FEATURES AND PROGRAM ELEMENTS

Section 2.6 of NFPA 805 states:

> *"A monitoring program shall be established to ensure that the availability and reliability of the fire protection systems and features are maintained and to assess the performance of the fire protection program in meeting the performance criteria. Monitoring shall ensure that the assumptions in the engineering analysis remain valid"*

The intent of the review will be to confirm (or modify as necessary) the adequacy of the existing surveillance, testing, maintenance, compensatory measures, and oversight processes for transition to NFPA 805. This review will consider the following:

1) The adequacy of the scope of systems and equipment within existing plant programs, i.e., the necessary fire protection systems and features and nuclear safety capability equipment (NFPA 805 Section 1.5.1) are included to ensure the assumptions and results of the change evaluation remain valid.

2) The performance criteria for the availability, reliability or condition of fire protection systems and features relied on for the change evaluation.

3.0 DESIGN BASIS AND REFERENCE INFORMATION

3.1 National Fire Protection Association (NFPA) Standard 805-2001, Performance Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants

3.2 Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants, May 2006

3.3 NEI 04-02, Guidance For Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 1

3.4 FAQ 06-0002, NEI 04-02 Section 5.3.3 and App. I, Order of Questions for Change Analysis Screening, Revision 1c, Closure Memo ML070030276

3.5 FAQ 06-0003, Change Analysis Screening, Revision 1b, Closure Memo ML070030242

3.6 Regulatory Guide 1.174, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis, Revision 1

3.7 Engineering Directives Manual EDM-510, PRA Technical Adequacy, Rev. 0

4.0 SUMMARY INFORMATION

See Enclosure 1 for an outline of format and content.

4.1 CONCLUSION

The conclusion should determine that the variances identified in Section 1.1, Problem Statement are acceptable (with or without modifications) based upon:

- The measured change in CDF and LERF
- Adequate defense-in-depth and safety margins are maintained

The conclusion should also state that the results of this evaluation meet the requirements of NFPA 805 change evaluation and the guidance of Regulatory Guide 1.205; therefore, the fire area is compliant using the risk-informed, performance-based approach and that a success path effectively remains free of fire damage and therefore the nuclear safety performance criteria of NFPA 805 are met.

4.2 DISPOSITION OF IDENTIFIED VARIANCES

The disposition of the final compliance strategies for the identified variances should be documented in the 'Conclusion Section'. For example:

- Operator manual actions will be dispositioned as a required action, a defense-in-depth action, or the action is no longer required.
- The variance requires a modification to demonstrate compliance with either the deterministic or performance-based requirements of NFPA 805.

4.3 DEFENSE-IN-DEPTH ENHANCEMENTS

In Section 4.3 the Evaluator will discuss any enhancements to defense-in-depth elements made as a result of this change evaluation. This discussion is a summary of the information presented in Section 6.

4.4 MONITORING OF REQUIRED SYSTEMS, FEATURES AND PROGRAM ELEMENTS

For each fire area the fire protection systems, features, and program elements that were relied upon should be listed for input into the monitoring program (note a comprehensive list of systems, features, and elements to be considered for monitoring are listed in the defense-in-depth table).

5.0 ASSUMPTIONS

The following assumptions were used in this calculation:

- No specific assumptions.

6.0 TECHNICAL PRESENTATION

Enclosure 1 contains an outline of the format and content of the change evaluations.

ENCLOSURE 1

1.0 PROBLEM STATEMENT

1.1 DESCRIPTION OF THE PROPOSED CHANGE

In Section 1.1 of each change evaluation the following information should be presented:

- Open Item Identifier from Calculation OSC-9292, NFPA 805 Transition B-3 Table/Report.
- For each open item a brief description of the variant condition

For fire areas that impact multiple fire areas, such as SSF, sub-divide the discussion by Unit. For example:

"1.1.1 UNIT 1 VARIANCES

A fire in this fire area does not result in an impact to Unit 1.

1.1.2 UNIT 2 VARIANCES

A fire in this fire area does not result in an impact to Unit 2.

1.1.3 UNIT 3 VARIANCES

The changes identified for evaluation for the Unit 3 Reactor Building (Fire Area RB3) are grouped as follows:

- Variances from Deterministic Separation Requirements - Based on the deterministic pre-transition safe shutdown analysis, Calculation OSC-9292, NFPA-805 Transition B-3 Table/Report, Revision 0, the following variances are evaluated in this calculation. See Attachment A for additional information associated with the variances.
  - *Open Item RB3-07-0 – Potential spurious opening of power operated relief valve (PORV) 3RC-66 with failure of the corresponding PORV block valve (3RC-4) to close results in breach in the Reactor Coolant System (RCS) integrity that requires crediting an unallowed manual action to de-energize PORV block valve (3RC-4) control circuitry to fail the PORV block valve closed. This action is necessary to demonstrate compliance with the nuclear performance criteria for Inventory and Pressure Control. (Reference: PIP No.: O-08-02520, CA Sequence No. 23)*
  - *Open Item RB3-11-0E – Potential Failure of steam generator (SG) Crosstie valve 3CCW-269, results in runout of 3A motor driven emergency feedwater pump (MDEFDWP) and uncontrollable diversion of emergency feedwater (EFDW) to the non-credited SG. This results in an unallowed manual action to remove power from 3CCW-269 to preclude potential fire effects. This action is necessary to demonstrate compliance with the nuclear performance criteria for Decay Heat Removal. (Reference: PIP No.: O-08-02520, CA Sequence No. 25)."*
  - *Open Item RB3-33-O – Potential spurious closing of 3FDW-347 results in the isolation of the credited feedwater flowpath to 3B steam generator and the inability to*

cool down the RCS. This results in an unallowed manual action to remove power from 3FDW-347 to preclude potential fire effects. This action is necessary to demonstrate compliance with the nuclear performance criteria for Decay Heat Removal and Inventory and Pressure Control. (Reference: PIP No. O-08-02520, CA Sequence No. 88)

1.2 CALCULATION CLASSIFICATION

In section 1.2 of each change evaluation, the calculation classification should be addressed. For example,:

"This calculation is being performed in support of the transition of the Fire Protection Safe Shutdown Program in accordance with 10CFR50.48(c). This calculation is considered QA-1 because it impacts systems, structures, or components (SSCs) that are QA-1 (for purposes other than fire mitigation). Note that per a 4/12/1995 letter from J.W. Hampton to the NRC, the NRC has specifically approved the use of non safety-related SSCs to mitigate fires.

The PRA insights associated with this calculation are not QA-1 based on the guidance provided in Regulatory Guide 1.174, Revision 1, dated November 2002. Regulatory Guide 1.174 (An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis) section 2.5 states:

"...To the extent that a licensee elects to use PRA information to enhance or modify activities affecting the safety-related functions of SSCs, the following, in conjunction with the other guidance contained in this guide, describes methods acceptable to the NRC staff to ensure that the pertinent quality assurance requirements of Appendix B to 10 CFR Part 50 are met and that the PRA is sufficient to be used for regulatory decisions..."

See EDM-510 for how Duke Energy PRA meets the requirements of Regulatory Guide 1.174."

2.0 DESIGN METHOD/ANALYTICAL METHOD

Section 2.0 of each fire area specific change evaluation should include the following information:

"The methodology for the change evaluations is based on the guidance in NFPA 805 and Section 5.3 of NEI 04-02, Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c), Revision 1, as modified by Frequently Asked Questions (FAQs) 06-0002 and 06-0003, and Regulatory Guide 1.205, Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants.

NEI 04-02 provides requirements and guidance for performing risk-informed, performance-based change evaluations. FAQs 06-0002 and 06-0003 modified the process slightly. NRC clarification of the change evaluation process is provided in Section C.3.2 of Regulatory Guide 1.205. The Plant Change Process consists of the following subtasks:

- Defining the Change
- Preliminary Risk Screening
- Risk Evaluation

- Acceptance Criteria Assessment
  - *The change in core damage frequency (ΔCDF) is acceptable, and*
  - *The change in large early release frequency (ΔLERF) is acceptable, and*
  - *Defense-in-depth and safety margins are maintained.*

The Acceptance Criteria Assessment is evaluated utilizing the risk guidance outlined in Regulatory Guide 1.174 and the plant specific risk as determined in OSC-9518, ONS NFPA 805 Fire PRA Application Calculation. In addition, the fire protection features and systems relied upon in the change evaluations are assessed for inclusion into the monitoring program.

Background and methodology for performing change evaluations are documented in Calculation OSC-9314, NFPA 805 Transition Risk-Informed, Performance-Based Change Evaluation Methodology."

3.0 DESIGN BASIS AND REFERENCE INFORMATION

3.1 DESCRIPTION OF THE FIRE AREA

This section provides an overall discussion of the fire protection systems and features available in the area.

In Section 3.1 of each fire area specific change evaluation the following information should be presented:

- Section 3.1 – Description of the Fire Area

  An overall description of the area should be provided. This description should include zones in the area, elevations, and a general description of the interfacing boundaries. For example, "*the south wall separates the Fire Area X Fire Area from the Unit 3 Reactor Building*"

- Section 3.1.1 - Construction

  Provide a general description of the Fire Area boundary construction and its adequacy for the hazard. For example "*The fire barrier separating the East and West Penetration Rooms erected near the west corner of the Spent Fuel Pools is a 3 hour rated fire barrier constructed of a tube steel/pyrocrete. The fire barriers for Fire Area X are 3 hour fire rated construction or have been evaluated to be adequate for the hazards. Applicable evaluations are documented in the following:*

  - OSC-7350, Att. 31, ONS Penetration Seal Database and 86-10 Evaluations, Rev. 6"

- Section 3.1.2 - Ventilation

  Provide a general discussion of the smoke removal capability provided for the area if necessary for an OMA and any specific assumptions/limitations associated with the fire modeling performed for the area. For example, *"Fire Area X has dedicated smoke removal system which can be aligned to exhaust smoke to the exterior of the building."*

- Section 3.1.3 - Detection

A general description of the detection provided in the area. For example, *"An ionization-type smoke detection system is provided for Fire Area X. Alarms for the detection system are supervised in the Unit 3 Control Room."*

- Section 3.1.4 – Fixed Fire Suppression

  A general description of the automatic fire suppression provided in the area. For example, *"Fire Area X has an automatic fire suppression providing partial area coverage."*

- Section 3.1.5 – Manual Suppression

  A general description of the manual fire suppression available to fight a fire in the area will be described. For example, *"A hose station is located on elevation 809' adjacent to Fire Area X. The hose station is supplied from interior station loop fire main and looped yard fire main system supplied from high pressure service water fire pumps located in lower level of the Turbine Building. Portable fire extinguishers are located throughout Fire Area X."*

- Section 3.1.6 – Fire Scenario

  Include a discussion of the fire scenario. For example, *"Fire Scenarios evaluated in the Fire PRA include a transient hot work fire, a transient fire, a fire associated with Electrical Cabinet-6."*

The information included in Section 3.0 of each fire area specific change evaluation should follow the requirements of EDM-101.

3.2 REFERENCES

This section provides references used as design input during the development of the change evaluation.

4.0 SUMMARY INFORMATION

Section 4.0 of the fire area specific change evaluation calculation shall contain the conclusions of the evaluation. These conclusions shall include:

- A statement that the change(s) met the acceptance criteria
- A disposition of the individual open items, and
- Suggested input to the monitoring program.

4.1 CONCLUSION

In Section 4.1, the Evaluator shall state how the acceptance criteria are met. Examples of conclusion statement are provided below:

"The change evaluation determined that [upon completion of modifications and/or upon completion of programmatic changes] the variances identified in Section 1.1 are acceptable based upon:

- The measured change in CDF and LERF
- Adequate defense-in-depth and safety margins are maintained

The results of this evaluation meet the requirements of NFPA 805 Change Evaluation and the guidance of Regulatory Guide 1.205; therefore, the fire area is compliant with the performance-based approach."

For fire areas that affect more than one unit, ensure that the conclusions are specific to the unit as necessary.

4.2 DISPOSITION OF IDENTIFIED VARIANCES

In Section 4.2 the Evaluator will disposition each of the variances identified in Section 1. An example of this treatment is provided below:

Based on the change evaluation results, the variances identified in section 1.1 have been dispositioned as follows:

Table 4.2-1

Final Variance Disposition

Variance Description	*Disposition*
Open Item Fire Area X-10-O – Potential spurious PORV opening / failure of block valve, resulting in crediting an unallowed manual action to remove powet to the de-energize PORV block valve (3RC VA0004) control circuitry to fail the PORV block valve closed.	*No recovery action required. Necessity of a defense-in-depth action will be evaluated in OSC-9535.*

For fire areas that affect more than one unit, ensure that the variances are specific to the unit as necessary.

4.3 DEFENSE-IN-DEPTH ENHANCEMENTS

In Section 4.3 the Evaluator will discuss any enhancements to defense-in-depth elements made as a result of this change evaluation. This discussion is a summary of the information presented in Section 6.

The following table represents those elements of defense-in-depth that were enhanced as a result of the change:

Table 4.3-1

Defense-in-Depth Enhancements

Defense-in-Depth Element	*Disposition*
Prevent fires from starting	*Acceptable as is.*
Rapidly detect control and extinguish promptly those fires that do occur thereby limiting fire damage	*Acceptable as is.*

Table 4.3-1

Defense-in-Depth Enhancements

Defense-in-Depth Element	*Disposition*
Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed	*No recovery action required. Necessity of a defense-in-depth action will be evaluated in OSC-9535. (See Section 4.2 for additional information)*

For fire areas that affect more than one unit, ensure that the defense-in-depth enhancements are specific to the unit as necessary.

4.4 MONITORING OF REQUIRED SYSTEMS, FEATURES AND PROGRAM ELEMENTS

Upon completion of the change evaluations the Evaluator will review the assumptions and input in the analysis to verify / identify those systems, features and/or programmatic controls that need to be included in the monitoring program. Examples include:

- Combustible Free Zones established to ensure fire damage is limited.
- Suppression Systems credited for defense in depth or credited in the FPRA to limit damage
- Detection systems credited for defense in depth or credited in the FPRA to limit damage

The results of this review shall be documented in Section 4.4 of the change evaluation calculation. An example of this treatment is provided below (note a comprehensive list of systems, features, and elements to be considered for monitoring are listed in the defense-in-depth table):

The following fire protection systems, features, and program elements are required by the FPRA and will be included in the monitoring program:

- Passive boundaries
  - *External Fire Area Boundaries*
- Programmatic/Administrative Controls
  - *Transient Control Program (if 'combustible free zones' are established – they should be discussed here)*
  - *Hot Work Control Program*

5.0 ASSUMPTIONS

5.1 METHODOLOGY ASSUMPTIONS

Calculation OSC-9314, NFPA 805 Transition Risk-Informed, Performance-Based Change Evaluation Methodology contains the methodology, including items which have been addressed

through the methodology. These items address topics which are programmatic in nature and which otherwise would be treated as assumptions in the area specific change evaluations. Refer to OSC-9314, Section 2.4 for a list of these items.

5.2 AREA SPECIFIC ASSUMPTIONS

5.2.1 Transients and/or hotwork activities are not normally allowed in the Reactor Building during power operations.

5.2.2 Sufficient intervening combustibles do not exist to allow propagation of fire between the east and west side of the Reactor Building.

6.0 TECHNICAL PRESENTATION

6.1 PRELIMINARY RISK SCREENING RESULT

Section 6.1 of the change evaluation calculation shall document the results of the Preliminary Risk Screening

It is not expected that the Transition Change Evaluations will screen at this stage. The following statements can be utilized: (1) For trivial or editorial changes, "*The open item(s) contained in section 1.1 do not include any changes evaluated to meet the trivial or editorial change criteria and therefore detailed change evaluations were performed.*" or (2) For minimal risk impact changes, *"The open item(s) contained in section 1.1 do not include any changes meeting the minimal risk impact change criteria and therefore detailed change evaluations were performed."*

6.2 RISK EVALUATION RESULTS

Section 6.2 of each change evaluation will document the results of the Risk Evaluation.

6.2.1 FIRE MODELING

Enter the following statement in Section 6.2.1, Fire Modeling, "*At this time, no specific fire modeling has been performed to support the change evaluation process. The combined analysis approach is used during transition (Ref. NEI 04-02, 5.3.4.3); therefore, MEFS/LFS is not analyzed separately from the Fire PRA results.*"

6.2.2 RISK EVALUTION

For each variance identified in Section 1.1 provide the results of the Risk Evaluation in Section 6.2.2 of the change evaluation calculation. An example of the results table is provided below.

"For each variance identified in Section 1.1, the detailed risk evaluation determined the change in CDF and LERF: Of the scenarios identified in Calculation OSC-9375, Oconee Fire PRA Scenario Development, Rev. 0, no scenarios had a calculated CDF; above 1E-08/yr.

These values are summarized in the table below. Details regarding the change evaluation are included in Attachment A."

Table 6.2-1

Quantitative Change Evaluation Results

Variance Description	*Delta CDF*	*Delta LERF*
Enter each open item from B-3 Table requiring a change evaluation	*Enter results of calculation*	*Enter results of calculation*
Total	*Enter Total*	*Enter Total*

Details regarding the change evaluation are included in Attachment A."

For fire areas that affect more than one unit, provide a Table for each Unit.

6.2.3 DEFENSE-IN-DEPTH AND SAFETY MARGIN

In Section 6.2.3, of the change evaluation calculation the Evaluator shall provide the results of the Defense-in-Depth and Safety Margin Review

DEFENSE-IN-DEPTH

In this Section provide the results of the Defense-in-Depth review in the following tabular format. Care should be taken to only include those fire protection features, operator manual actions that are necessary to ensure that defense-in-depth is maintained. For example:

A review of the impact of the change on defense-in-depth was performed using the guidance from NEI 04-02 (See OSC-9314, Transition Risk-Informed, Performance-Based Change Evaluation Methodology). In general, the defense-in-depth requirement is satisfied if the proposed change does not result in a substantial imbalance among these elements. The table below summarizes how adequate defense-in-depth is maintained.

NOTE: Unless specifically noted, the information below is applicable for all 3 units.

Table 6.2-2

Defense-in-Depth Impact Review

Method of Providing DID	*Impact on Plant Configuration and Control*	*Changes and Net Effect*
Prevent fires from starting		
Combustible Control is implemented in accordance with NSD-313, Control of Flammable and Combustible Materials	*Required*	*This element of defense-in-depth remains unchanged*
Hot Work Control is implemented in accordance with NSD-314, Hot Work Authorization	*Required*	*This element of defense-in-depth remains unchanged*

Table 6.2-2

Defense-in-Depth Impact Review

Method of Providing DID	*Impact on Plant Configuration and Control*	*Changes and Net Effect*
Rapidly detect, control and extinguish promptly those fires that do occur thereby limiting fire damage		
Ionization-type smoke detection system is provided	*Credited*	*This element of defense-in-depth remains unchanged*
Automatic fire suppression is not provided in the area.	*Not Credited*	*Not Applicable*
Portable fire extinguishers are provided in the area.	*Credited*	*This element of defense-in-depth remains unchanged*
Hose stations and hydrants located in the area(s)	*Credited*	*This element of defense-in-depth remains unchanged*
Fire Pre-Fire Plan	*Not Credited*	*Not Applicable*
Provide adequate level of fire protection for systems and structures so that a fire will not prevent essential safety functions from being performed		
Walls, floors ceilings and structural elements are rated or have been evaluated as adequate for the hazard.	*Required*	*This element of defense-in-depth remains unchanged*
Penetrations in the fire area barrier are rated or have been evaluated as adequate for the hazard.	*Required*	*This element of defense-in-depth remains unchanged*
Discuss any supplemental barriers (e.g., ERFBS, cable tray covers, etc.)	*Not Credited*	*Not Applicable*
Discuss fire rated cable	*Not Credited*	*Not Applicable*
Reactor coolant pump oil collection system (as applicable)	*Not Credited*	*Not Applicable*
Guidance provided to operations personnel detailing the credited success path(s) including recovery actions to achieve nuclear safety performance criteria.	*Credited*	*No recovery action required. Necessity of a defense-in-depth action will be evaluated in OSC-9535.*

LEGEND:

- *Required – DID feature is required for inclusion in the Monitoring Program to maintain FPRA inputs and assumptions.*
- *Credited – DID feature is not required per FPRA, however it provides additional levels of nuclear safety, and will be included in a plant tracking program.*
- *Not Credited – Not required for DID.*

SAFETY MARGIN

In this Section the Evaluator shall document the results of the **qualitative** safety margin review. The treatment should evaluate each issue presented in Section 2.5.3 and provide an answer/discussion of these issues. An example of this treatment is provided below:

"In accordance with NEI 04-02, 5.3.5.3 guidance, the maintenance of adequate Safety Margin is assessed by the consideration of categories of analyses utilized by this change evaluation. Safety margins are considered to be maintained if

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

The maintenance of adequate safety margin for the change analysis is described for each of the specific analysis types used in support of the fire risk assessment.

FIRE MODELING

Fire modeling performed in support of the transition has been performed within the FPRA utilizing codes and standards developed by industry and NRC staff to provide realistic yet conservative results. Specifically, the heat release rates utilized in the transition analysis are based upon NUREG 6850, Task 8, Scoping Fire Modeling. These heat release rates are conservative and represent values used to screen out fixed ignition sources that do not pose a threat to the targets within specific fire compartments and to assign severity factors to unscreened fixed ignition sources.

The combined analysis approach is used during transition; therefore, maximum expected fire scenario/limiting fire scenario have not been analyzed separately. The bases for the application of these fire modeling codes and standards were not altered in support of this change evaluation.

PLANT SYSTEM PERFORMANCE

Plant system performance parameters were not modified as a result of this change evaluation. These performance parameters were originally established to support nuclear performance criteria contained in the plant specific accident analyses. These analyses established component and system performance criteria necessary to establish safe and stable plant operation, as well as, safe shut down of the unit in the event of a fire. These performance parameters were not modified as a result of this change evaluation.

PRA LOGIC MODEL

Adequate safety margin is maintained because the codes and standards used have been accepted for use by the NRC and the acceptance criteria in NFPA 805. The specific codes and standards used in the FPRA application and development were 10CFR50.48(c), NFPA 805, 2001 edition, NRC Regulatory Guide 1.205, Revision 0, and NUREG 6850. These codes and standards were applied in a manner which would provide FPRA results which contain and complement safety

margin. The bases for the application of these FPRA codes and standards were not altered in support of this change evaluation.

ATTACHMENT A – PRA CALCULATION RESULTS

In the Attachments for to the change evaluations the PRA calculation results should be summarized for each unit (Attachment A for Unit 1, Attachment B for Unit 2 and Attachment C for Unit 3). The following discusses the concept/purpose of each section/sub-section and provides an example of content and level of detail.

A KEY PRA INSIGHTS UNIT 1

Section A.1 should provide an overview of the fire area as follows:

- Fire zones in area
- Key PRA Components
- General layout of targets and combustibles
- Discussion of the fire scenarios evaluated
- Discussion of the fire scenarios that will be evaluated as part of the change evaluation

See example below:

A.1 FIRE AREA RB3 – UNIT 3 REACTOR BUILDING

This fire area is comprised of the Unit 3 Reactor Building (RB3). Key PRA components/functions potentially impacted include:

- *Motor driven EFW Pumps (automatic SGLC operation only)*
- *CC Pumps 3A & 3B*
- *Miscellaneous HPI and LPI valves*
- *RCP's*
- *Pressurizer Heaters*
- *Turbine Bypass Control*
- *PORV & PORV Block*
- *SSF RCM Pump*

Failure of Compressed Air systems is excluded in the Reactor Building (no credited AOVs in RB). Also, since power is removed from 3RC VA0155, 156, 157, 158, 159, & 160 during power operations (reference OP 1103/002), spurious opening of the vent valves due to a fire is precluded. Development of west side and east side fire scenarios on either side of the shield wall within the Reactor Building relied on cable separation, location of key ignition sources and their associated zones of influence, and lack of intervening combustibles for the assumed target damage. There are relatively few ignition sources in the Reactor Building that are capable of damaging multiple target sets, primarily the reactor coolant pumps.

Of the scenarios identified in Calculation OSC-9375, Oconee Fire PRA Scenario Development, Rev. 0, only RB03 Scenarios B1, D, and E had a calculated CDF above 1E-08/yr and a calculated LERF above 1E-09/yr. It is noted that containment bypass scenarios involving failure of the HP20, & HP21 inboard and outboard isolation valves for a fire in RB3 are possible.

Refinements to the Altered Events table in FRANC were identified to support the delta risk calculations. The changes included the addition of circuit failure likelihood values for valves in containment bypass pathways and elimination of circuit failure likelihood values for spurious RCP operation. None of these changes had a significant impact on the overall fire CDF/LERF results. It is recommended that these changes be captured in a future revision to the FPRA.

The following table provides summary level information on the RB03 scenarios above the screening thresholds.

Table A.1-1

FC	Scenario	Scenario Description	Targets	CDF	LERF
RB03	B1	RCP 3B1/3B2 Severe Fire	Damage to trays within shield wall result in loss of PORV and PORV Block and other failures; Containment bypass scenario (ISLOCA pathway via seal return).	1.05E-08*	7.80E-09*
RB03	D	RB Fire - West Half	Damage to trays between containment wall and shield wall in west half of Reactor Building result in loss of SSF RCM Pump, MD EFDW pumps, Turbine Bypass Control, pressurizer heaters.	7.96E-07	7.99E-07

Table A.1-1

FC	*Scenario*	*Scenario Description*	*Targets*	*CDF*	*LERF*
RB03	*E*	*RB Fire - East Half*	*Damage to trays between containment wall and shield wall in east half of Reactor Building result in loss of SSF RCM Pump, MD EFDW pumps, Turbine Bypass Control, pressurizer heaters, 3LP VA0001, and PORV.*	*1.60E-07*	*6.97E-09*

*It was discovered that the Altered Events table in FRANC included circuit failure likelihood values for various power operated valves for RB03 scenarios 'B' and 'C'; these scenarios should have been identified as 'B1' and 'C1'. A change was made to the Altered Events table to support the delta risk calculations; however, this change has a minimal impact on the overall CDF / LERF results. It is recommended that this change be captured in a future revision to the FPRA.

LEGEND:

- FC – Fire Compartment (generally corresponds to Fire Zone; AB098 is FZ 98 in the Aux Building)
- Scenario – unique alpha-numeric designator
- Scenario Description – ignition source
- Targets – External damage based on ignition source zone of influence
- CDF – Core damage frequency per reactor year
- LERF – Large early release frequency per reactor year

Section A.2 should provide an overview of each open item that requires a change evaluation as follows:

- Discuss the variance and the nuclear safety performance criteria that are not being met
- Discuss the variance and how it will be modeled in the Fire PRA
- In tabular format provide 'Basic Event(s)' that will be 'toggled' to manifest the variant condition and the results of the FPRA manipulations

See example below:

A.2 – OPEN ITEM CHANGE EVALUATIONS

Open items from calculation OSC-9292, NFPA-805 Transition B-3 Table/Report are identified as candidates for change evaluation. The following discussions provide a definition of the problem statement (i.e., the variant condition) followed by a table that provides a summary of results. The following information is provided for the Open Items addressed in this calculation:

A.2.1 OPEN ITEM RB3-07-O – POTENTIAL SPURIOUS PORV OPENING / FAILURE OF BLOCK VALVE

An unallowed operator manual action to isolate pressurizer PORV block valve control circuitry for HSB is credited in the safe shutdown analysis due to the postulated fire-induced failure of 3RC-66 and 3RC-4 in Fire Area RB3. In order to ensure RCS pressure and inventory control for achieving and maintaining hot standby (HSB), the safe shutdown analysis requires either that the pressurizer PORV (3RC-66) remain closed or that the PORV Block valve (3RC-4) remains available to be closed from the control room. Cables for both 3RC-66 and 3RC-4 are routed in close proximity in Fire Area RB3. Therefore, it is assumed in the safe shutdown analysis that RC-66 could spuriously open and 3RC-4 may not be able to be closed (Case 1) from the control room.

From the FPRA [Calculation OSC-9375, Oconee Fire PRA Scenario Development, Rev. 0, also see supporting fire PRA database file], the PORV and PORV Block Valve are included in RB3 Scenarios B1 and D. The basic event for spurious PORV opening (G00PORVDEX) is more representative of the non-compliant case (Case 1). The compliant case (Case 2) will assume G00PORVDEX is not failed for Scenarios B1 and D.

Table A.2-1

Component	Manual Action	Basic Event	Basic Event Description/Discussion	Case 1 Variant CDF/LERF	Case 2 Compliant CDF/LERF	ΔCDF/ ΔLERF
3RC VA0066 (Pzr PORV)	Close	G00PORVDEX	PORV 3RC-V66 Spuriously Opens due to Fire	9.66E-07/ 8.13E-07	8.95E-07/ 7.42E-07	7.17E-08/ 7.16E-08

These values fall within the acceptable limits of delta CDF less than 1E-06 and a delta LERF less than 1E-07. The delta risk values are within the acceptable range, including consideration of uncertainties, as described in Calculation OSC-9518, ONS NFPA 805 Fire PRA Application Calculation, Revision 0. (NOTE: If the delta CDF and delta LERF fall within a decade of the acceptance criteria, additional discussion should be provided to support the conclusion and it's susceptibility to impacts from uncertainties.)

LEGEND:

- *Component – The component of concern that is associated with the variant condition/change evaluation.*
- *Manual Action – For variant conditions/change evaluations involving unallowed operator manual actions, a brief statement of the operator manual action function.*
- *Basic Event – Fire PRA Basic Event ID(s) corresponding to the component(s) and function of concern.*
- *Basic Event Description/Discussion - Text description associated with the Basic Event IDs.*
- *Case 1 - Variant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition exists in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be unprotected).*
- *Case 2 - Compliant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition was assumed NOT to exist in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be protected or routed outside of the fire area).*
- *ΔCDF/ΔLERF – The change in CDF and LERF between the variant condition and the compliant condition (Case 1 – Case 2).*

A.2.2 OPEN ITEM RB3-11-OE – POTENTIAL FAILURE OF SG CROSSTIE VALVE 3CCW-269

An unallowed operator manual action to remove power to the SG crosstie valve 3CCW-269 is credited in the safe shutdown analysis for hot standby (HSB). Flow to either SG A or B is necessary to ensure HSB decay heat removal. For a fire in the west side Fire Area RB3 emergency feedwater is isolated to the 3B SG via 3FDW-316 due to the potential loss of all 3B SG instrumentation. It is conservatively assumed that spurious opening of SG Crosstie Valve 3CCW-269 will divert flow from SG A to isolated SG B and fail motor driven EFW Pump A due to a potential pump runout (based on lack of documented analysis). Therefore, based on the conservative assumptions in the safe shutdown analysis, a complete loss of EFW to both SGs could occur due to the interactions described above.

Spurious opening of 3CCW-269 is not explicitly modeled in the FPRA. In order to address the potential for pump runout the spurious opening of 3CCW-269 was modeled in the FPRA as failure of both motor driven EFW pumps and the turbine driven EFW pump. 3CCW-269 is only impacted in RB03 Scenarios B1 and D.

From the FPRA [Calculation OSC-9375, Oconee Fire PRA Scenario Development, Rev. 0 (also see supporting fire PRA database file, archived with this calculation)], CCW-269 are included in RB03 Scenarios B1 and D. The spurious opening of 269 was conservatively modeled and are removed failure of EFW events, where they are linked to 269, in order to create the compliant case. The non-compliant case (Case 1) will assume failure of both MD EDFW pumps and the TD EFDW pump.

Table A.2-2

Component	Manual Action	Basic Event	Basic Event Description/Discussion	Case 1 Variant CDF/LERF	Case 2 Compliant CDF/LERF	ΔCDF/ ΔLERF
3CCWVA0269	Close	FEFTDFPTPS FEFMDPAMPR FEFMDPBMPR NCW0269MVT NCW0269MVO	While only FEFMDPAMPR is relevant; all 5 events (including failure to open, transfers closed, and failure of the remaining EFDW pumps) are toggled.	9.66E-07/ 8.14E-07	9.29E-07/ 7.77E-07	3.73E-08/ 3.70E-08

These values fall within the acceptable limits of delta CDF less than 1E-06 and a delta LERF less than 1E-07. The delta risk values are within the acceptable range, including consideration of uncertainties, as described in calculation OSC-9518, ONS NFPA 805 Fire PRA Application Calculation, Revision 0.. (NOTE: If the delta CDF and delta LERF fall within a decade of the acceptance criteria, additional discussion should be provided to support the conclusion and it's susceptibility to impacts from uncertainties.)

LEGEND:

- *Component – The component of concern that is associated with the variant condition/change evaluation.*
- *Manual Action – For variant conditions/change evaluations involving unallowed operator manual actions, a brief statement of the operator manual action function.*

- *Basic Event – Fire PRA Basic Event ID(s) corresponding to the component(s) and function of concern.*
- *Basic Event Description/Discussion - Text description associated with the Basic Event IDs.*
- *Case 1 - Variant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition exists in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be unprotected).*
- *Case 2 - Compliant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition was assumed NOT to exist in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be protected or routed outside of the fire area).*
- *ΔCDF/ΔLERF – The change in CDF and LERF between the variant condition and the compliant condition (Case 1 – Case 2).*

A.2.3 OPEN ITEM RB3-33-O – POTENTIAL FAILURE OF FEEDWATER VALVE 3FDW-347

The safe shutdown analysis requires that emergency feedwater flow be established to both steam generators to ensure that the requirements of the inventory and pressure control nuclear safety performance criterion are maintained during the transition from hot standby to cold shutdown. The spurious closing of 3FDW-347 results in the isolation of the credited emergency feedwater flowpath to the B steam generator and the inability to cool down the RCS. The inability to cool down the RCS results in water solid conditions in the reactor coolant system, and lifting of the Pressurizer PORV (if available) or the Pressurizer code safety relief valves. The lifting of the Pressurizer relief valves challenges containment habitability and the ability to perform required cold shutdown manual actions within containment. An unallowed operator manual action to remove power to 3FDW-347 to preclude fire effects is credited in the safe shutdown analysis.

Flow to the steam generator B is required to allow transition from hot standby conditions to cold shutdown. The FPRA does not model the risk impacts for this transition period. Hot and stable conditions can be achieved utilizing the credited A steam generator. Based on the FPRA success criteria, this assessment concludes that the impact of the manual action is not risk significant and no quantification of delta risk is required. Therefore, the delta risk associated with the operator manual action is negligible and is characterized as epsilon (ε).

Table A.2-3

Component	*Manual Action*	*Basic Event*	*Basic Event Description/Discussion*	*Case 1 Variant CDF/LERF*	*Case 2 Compliant CDF/LERF*	*ΔCDF/ ΔLERF*
3FDW-347	Open	N/A	N/A	*ε/ε*	*ε/ε*	*ε/ε*

These values fall within the acceptable limits of delta CDF less than 1E-06 and a delta LERF less than 1E-07. The delta risk values are within the acceptable range, including consideration of uncertainties, as described in calculation OSC-9518, ONS NFPA 805 Fire PRA Application Calculation, Revision 0. (NOTE: If the delta CDF and delta LERF fall within a decade of the acceptance criteria, additional discussion should be provided to support the conclusion and it's susceptibility to impacts from uncertainties.)

LEGEND:

- *Component – The component of concern that is associated with the variant condition/change evaluation.*
- *Manual Action – For variant conditions/change evaluations involving unallowed operator manual actions, a brief statement of the operator manual action function.*
- *Basic Event – Fire PRA Basic Event ID(s) corresponding to the component(s) and function of concern.*
- *Basic Event Description/Discussion - Text description associated with the Basic Event IDs.*
- *Case 1 - Variant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition exists in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be unprotected).*
- *Case 2 - Compliant CDF/LERF – For the specific open item, these values represent the core damage frequency (CDF) and large early release frequency (LERF) if the discrepant condition was assumed NOT to exist in the fire area (e.g., the cable routed in the fire area that prompted the need for the operator manual action is assumed to be protected or routed outside of the fire area).*
- *ΔCDF/ΔLERF – The change in CDF and LERF between the variant condition and the compliant condition (Case 1 – Case 2).*