ENCLOSURE 8

Point Beach Units 1 and 2
License Amendment Request to Revise Technical Specifications
to Adopt Risk Informed Completion Times TSTF-505, Revision 2,
"Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b"

Attributes of the Configuration Risk Management Model

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1.0 Purpose

Section 4.0, Item 9 of the Nuclear Regulatory Commission's (NRC) Final Safety Evaluation (Reference 1) for NEI 06-09, Revision 0A, "Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines," (Reference 2) requires that the license amendment request (LAR) provide a description of PRA models and tools, including identification of how the baseline probabilistic risk assessment (PRA) model is modified for use in the Configuration Risk Management Program (CRMP) tools, quality requirements applied to the PRA models and Configuration Risk Management (CRM) tools, consistency of calculated results from the PRA model and the CRM tools, and training and qualification programs applicable to personnel responsible for development and use of the CRM tools. The scope of structures, systems, and components (SSCs) within the CRM will be provided. This item should also confirm that the CRMP tools can be readily applied for each Technical Specification (TS) limiting condition for operation (LCO) within the scope of the plant-specific submittal.

This enclosure describes the necessary changes to the peer-reviewed baseline PRA models for use in the CRM software to support the Risk-Informed Completion Time (RICT) Program. The process employed to adapt the baseline models for CRM use is demonstrated:

(a) to preserve the core damage frequency (CDF) and large early release frequency (LERF) quantitative results;

(b) to maintain the quality of the peer-reviewed PRA models; and

(c) to correctly accommodate changes in risk due to configuration-specific considerations.

Quality controls and training programs applicable for the CRM are also discussed in this enclosure.

2.0 Revision Summary

Initial Issuance.

3.0 Evaluation

3.1.1 Translation of Baseline PRA Model for Use in Configuration Risk Management

The baseline PRA models for internal events, internal flood, and internal fire models, are the peer-reviewed models, updated when necessary to incorporate plant changes to reflect the as-built/as-operated plant as discussed in Enclosure 7. Prior to implementation of the RICT Program, the internal events model will be integrated with internal flood model and internal fire model to develop a one-top integrated baseline model applicable to both units. This baseline model will be translated to develop the CRM model to be used for the RICT Program by allowing user-specified alignments to match the actual plant configuration.

It is intended to use EPRI's Phoenix Risk Monitor software to facilitate all configuration-specific risk calculations, and in particular, to support the RICT Program implementation. This software will also include the seismic risk contributions for RICT calculations. The integrated model may include additional changes that are currently logged in the database for periodic model maintenance and update that are considered pending for the upcoming cycles of model update in accordance with fleet procedures.

3.1.2 Quality Requirements and Consistency of PRA Model and Configuration Risk Management Tools

The approach for establishing and maintaining the quality of the PRA models, including the CRM model, includes both a PRA maintenance and update process (described in Enclosure 7), and the use of self-assessments and independent peer reviews (described in Enclosure 2).

The information provided in Enclosure 2 demonstrates that the PBN internal events, internal flood, and internal fire PRA models conform to the associated industry standards endorsed by Regulatory Guide 1.200 (Reference 3). This information provides a robust basis for concluding that the PRA models are of sufficient quality for use in risk-informed licensing actions.

For maintenance of an existing CRM model, changes made to the baseline PRA model in translation to the CRM model will be controlled and documented in accordance with NEE fleet procedures. These procedures address the process for identification and corrective actions to evaluate and disposition model errors and changes to ensure models are accurate, as described in Enclosure 7. Acceptance testing is performed after every configuration risk model update to ensure that the software works as intended and that quantification results are reasonable. The CRM model is nominally updated to reflect the as-built, as-operated plant once every two refueling cycles, of both units in a two unit site, but may be sooner depending on plant needs and management discretion.

These actions satisfy NEI 06-09-A, Section 2.3.5, Item 9.

3.1.3 Training and Qualification

PRA staff is responsible for development and maintenance of the CRM model. The PRA staff is trained in accordance with the Engineering Support Personnel (ESP) training program. Operations and Work Control staff will use the CRM tool under the RICT Program and staff are trained in accordance with a program using National Academy for Nuclear Training (ACAD) documents, which is also accredited by INPO.

3.1.4 Application of the Configuration Risk Management Tool to the RICT Program Scope

The plant will use the EPRI Phoenix Risk Monitor software or equivalent as its CRM platform. This program is specifically designed by EPRI to support implementation of RICT, and is currently used at site. Phoenix Risk Monitor will permit the user to evaluate all configurations within the scope of the RICT Program using appropriate mapping of equipment to PRA model elements.

5.0 References

1. ML071200238, Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, "Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines (TAC No. MD4995)," Letter from the NRC to NEI, May 17, 2007.

2. NEI 06-09, Risk-Informed Technical Specifications Initiative 4B: Risk-Managed Technical Specifications (RMTS) Guidelines, Nuclear Energy Institute, Revision 0-A, November 2006..

3. Regulatory Guide 1.200, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Revision 2, March 2009 .

ENCLOSURE 9

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Key Assumptions and Sources of Uncertainty

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1.0 Introduction

The purpose of this enclosure is to disposition the impact of Probabilistic Risk Assessment (PRA) modeling epistemic uncertainty for the Risk Informed Completion Time (RICT) Program. Nuclear Energy Institute (NEI) topical report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 3), Section 2.3.4, item 10 requires an evaluation to determine insights that will be used to develop risk management actions (RMAs) to address these uncertainties. The baseline internal events PRA (including internal flood) and fire PRA models document assumptions and sources of uncertainty and these were reviewed during the model peer reviews. The approach taken is, therefore, to review these documents to identify the items which may be directly relevant to the RICT Program calculations, to perform sensitivity analyses where appropriate, to discuss the results and to provide dispositions for the RICT Program. The PB internal events (including internal flooding) and fire PRA models described within this LAR are based on those described within NEE submittals regarding adoption of 10 CFR 50.69, “Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors” (Reference 2).

The epistemic uncertainty analysis approach described below applies to the internal events PRA and any epistemic uncertainty impacts that are unique to fire PRA are also addressed. In addition, NEI 06-09-A requires that the uncertainty be addressed in RICT Program Real Time Risk tools by consideration of the translation from the PRA model. The Real Time Risk model, also referred to as the Configuration Risk Management (CRM) model, discussed in Enclosure 8 of this license amendment request (LAR), includes internal events, flooding events and fire events. The model translation uncertainties evaluation and impact assessment are limited to new uncertainties that could be introduced by application of the Real Time Risk tool during RICT Program calculations.

2.0 Revision Summary

Initial Issuance.

3.0 Evaluation

3.1 Assessment of Internal Events PRA Epistemic Uncertainty Impacts

In order to identify key sources of uncertainty for RICT Program application, an evaluation of internal events baseline PRA model uncertainty was performed, based on the guidance in NUREG-1855, “Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making”, Revision 1 (Reference 5) and Electric Power Research Institute (EPRI) Technical Report (TR)-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments” (Reference 6). As described in NUREG-1855, Revision 1, sources of uncertainty include "parametric" uncertainties, "modeling" uncertainties, and "completeness" (or scope and level of detail) uncertainties.

Parametric uncertainty was addressed as part of the PB baseline PRA model quantification (Reference 9). Modeling uncertainties are considered in both the base PRA and in specific risk-informed applications. Assumptions are made during the PRA development as a way to address a particular modeling uncertainty because there is not a single definitive approach. Plant-specific assumptions made for each of the PB internal events PRA technical elements are noted in the individual notebooks. The internal events PRA model uncertainties evaluation is documented in (Reference 9) and considers the modeling uncertainties for the base PRA by identifying assumptions, determining if those assumptions are related to a source of modeling uncertainty and characterizing that uncertainty, as necessary. EPRI compiled a listing of generic sources of modeling uncertainty to be considered for each PRA technical element (Reference 6), and the evaluation performed for PB (Reference 9) considered each of the generic sources of modeling uncertainty as well as the plant-specific sources.

Completeness uncertainty addresses scope and level of detail. Uncertainties associated with scope and level of detail are documented in the PRA. No specific issues of PRA completeness have been identified relative to this LAR, based on the results of the internal events PRA (including internal flood) review.

3.2 Assessment of Fire PRA Epistemic Uncertainty Impacts

The purpose of the following discussion is to address the epistemic uncertainty in the PB fire PRA. The fire PRA model includes various sources of uncertainty that exist because there is both inherent randomness in elements that comprise the fire PRA and because the state of knowledge in these elements continues to evolve. The development of the PB fire PRA was guided by NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology” (Reference 7), and the fire PRA model used consensus models described in NUREG/CR-6850. Enclosure 2 provides a detailed discussion of the peer review Facts and Observations (F&Os) and the resolutions.

The PB fire PRA was developed using consensus methods outlined in NUREG/CR-6850 and interpretations of technical approaches as required by NRC. Further, appropriate fire impacts were identified for the systems modeled in the internal events PRA and were addressed in the fire PRA. Fire PRA methods were based on NUREG/CR-6850, as well as other more recent NUREGs and published “frequently asked questions” (FAQs) for the fire PRA.

PB used guidance provided in NUREG-1855 and EPRI TR-1026511, “Practical Guidance on the Use of PRA in Risk-Informed Applications with a Focus on the Treatment of Uncertainty”, to review plant-specific and generic uncertainties associated with the fire PRA for the RICT Program application. The potential sources of model uncertainty in the PB fire PRA model were evaluated for their potential impacts on the RICT calculations. The review identified no specific uncertainty issues that would impact the RICT application.

3.3 Assessment of Online Risk Monitor Uncertainty Impacts

Incorporation of the baseline PRA models into the Online Risk Model used for RICT Program calculations may introduce new sources of model uncertainty. Table E9-1 provides a description of the relevant model changes and dispositions of whether any of the changes made represent possible new sources of model uncertainty that must be addressed.

Table E9-1			
Online Risk Monitor Assumption	**Aspect Impacted**	**Potential Impact on PRA**	**Disposition for RICT Program**
Incorporation of seismic risk bias to support RICT Program risk calculations. A conservative value for the seismic delta CDF is applicable	Calculation of RICT and risk management action time (RMAT) within the OLRM.	The addition of bounding impacts for seismic events has no impact on baseline PRA or OLRM. Impact is reflected in calculation of all RICTs and RMATs.	Since this is a bounding approach for addressing seismic risk in the RICT Program, it is not a source of translation uncertainty, and RICT Program calculations are not impacted. Therefore, no mandatory RMAs are required.

4.0 References

Reference 1. "Letter from the Technical Specification Task Force (TSTF) to the NRC, “TSTF Comments on Draft Safety Evaluation for Traveler TSTF-505, ‘Provide Risk-Informed Extended Completion Times’ and Submittal of TSTF-505, Revision 2”." *ML18183A493.* July 2, 2018.

Reference 2. "NEE License Amendment Request 287, “Application to adopt 10 CFR 50.69, ‘Risk-informed Categorization and Treatment of Structures, System, and Components (SSCs) for Nuclear Power Plants’”." *ML17243A201.* August 31, 2017.

Reference 3. "NEI 06-09-A Risk-Informed Technical Specifications Initiative 4b." *ML12286A322.* October 2012.

Reference 4. "Letter from the NRC to NEI, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, ‘Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines." *ML071200238.* May 17, 2007.

Reference 5. "NUREG-1855 'Guidance on the Treatment of Uncertainties Associated with PRAs'." *ML17062A466.* March 2017.

Reference 6 . "EPRI Technical Report TR-1016737." *Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments.* December 2008.

Reference 7. "NRC NUREG/CR-6850 Volume 2, 'EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Volume 2: Detailed Methodology'." *ML15167A411.* September 2005.

Reference 8. "NRC NUREG/CR-6850, Supplement 1, “Fire Probabilistic Risk Assessment Methods Enhancements'." *ML103090242.* September 2010.

Reference 9. "PRA 11.0." *Internal Events Quantification Notebook.* PBNP, December 2021.

Table E9-2 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption	**Potential Impact on PRA**	**Disposition for RICT Program**
The extension of power to H-02 from H-03 by closing of breakers H52-21 and H52-31 and vice versa via H-01 has not been modeled.	**Minimal Impact** - H-02 can be powered from H-03 through a jumper for simultaneous closing of breakers H52-21 and H52-31. The extension of power to H-02 from H-03 and vice versa via H-01 has not been modeled because of the involvement of this jumpering activity. The Impact of this conservatism is minimal and offset by the likelihood of properly completing the jumpering.	This assumption has a potentially conservative impact on the calculated RICT since power supplies which may actually provide support for mitigating capability are not reflected in the calculations. The impact is judged to be small since the H-01, H-02 and H-03 buses and associated breakers are not risk significant based on Fussell-Vesely (FV) and Risk Achievement Worth (RAW). The primary mode is modeled. The secondary mode identified in this assumption would have a smaller risk contribution. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
A flow diversion created by transferring open or misalignment of the RHR heat exchanger cross-connect valves, RH-713A, RH-713B, RH-716C, and RH-716D was not modeled in the fault tree.	**Minimal Impact** - A flow diversion through the RHR to CS system pump suction was not modeled in the fault tree due to its low probability. If a pump failed to run and these valves provided an open path between trains, flow through the minimum flow line would be insignificant. For significant flow to occur, the pump and its discharge and suction check valves would both have to fail open. These valves are verified shut monthly via 1 (2)-TS-ECCS-001. This event is considered to have a low frequency of occurrence and would have minimal Impact to the fault tree model.	The assumption has a potentially non-conservative impact on the calculated RICT for the ECCS systems, since there are failure modes which have not been included in the PRA. The failure modes which have not been included have a very small probability of occurrence (failure of 2 check valves to close) and are therefore judged not to be significant. Since the impact is conservative, no additional considerations are required to address this source of uncertainty in the RICT Program.
Although a significant portion of the small break spectrum does not require secondary heat removal, a means of makeup to the steam generators is assumed necessary in all cases.	**Minimal Impact** - This model simplification is conservative and adds a negligible amount of uncertainty to the PRA model.	This conservative assumption may cause shorter RICTs for equipment used for secondary heat removal functions. Since the impact is conservative, no additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-2 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption	**Potential Impact on PRA**	**Disposition for RICT Program**
Only the relay failures associated with the under-voltage (Relay-27) and bus fault (Relay-86) on bus 1A-03 were modeled in the CAFTA fault tree. The modeling assumption was repeated for other three EDG loading bus tie breakers too.	**Minimal Impact** - There are multiple instrumentation paths to trip the bus tie breaker 1A52-57 between 1A-03 and 1A-05. The bus tie breaker tripping and successful load shedding are the criteria for closing of the EDG breaker onto the bus. The logics leading to the tripping of tie breaker is as following: a. Under-voltage on 1A-05 (2/3 logic) or b. Degraded voltage on 1A-05 (2/3 logic) and SI signal (Unit-1 or Unit-2) or c. Degraded voltage on 1A-05 with 40 seconds time delay (2/3 logic) or d. Bus fault on 1A-05 or e. Bus fault on normal supply bus 1A-03. The logic is the same for other three EDG loading bus tie breakers. The failure rate of all the logical sequences except under-voltage (Relay-27) and bus fault (Relay-86) on bus 1A-03 was less that 1e-13. Thus only the respective relay failures were modeled in the CAFTA fault tree.	The assumption has a potentially non-conservative impact on the calculated RICT for the 4160 VAC system since there are additional relays which would provide protection and are not included in the model. Since the impact is on the order of 1E-13, it is judged to be small. No additional considerations are required to address this source of uncertainty in the RICT Program.

Table E9-2 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations		
Assumption	**Potential Impact on PRA**	**Disposition for RICT Program**
The RHR min flow line is required for successful system operation during injection.	**Minimal Impact -** Since the RHR pumps are high capacity and low head pumps, they need a minimum flow to prevent pump overheating. During an accident in which an "S" signal is generated, the RCS pressure may be above the shutoff head of the RHR pumps requiring the pumps to be on min flow for a period of time until they are procedurally stopped. The isolation valve in each RHR min flow line RH-733A and RH-733B have their positions indirectly verified quarterly during the performance of IT 03 (IT 04) and are verified locked open monthly via 1 (2)-TS-ECCS-001. These manual valves are normally open during plant operation and are likely to be open when the RHR is required to operate.	This assumption is conservative with respect to RICT calculations. The requirement to have min-flow for all LOCA break sizes will increase the potential failure modes of each RHR pump increasing risk importance systems.

Table E9-2 Disposition of Key Assumptions/Sources of Uncertainty Impacting Configuration Risk Calculations

Assumption	Potential Impact on PRA	Disposition for RICT Program
Two alternate sources of water to the Auxiliary Feedwater Pump suction are modeled. One is the service water system. The other is the fire water system. The water treatment system and condenser hotwells are also available for makeup but are not modeled. The service water makeup is piped to the suction of the aux. feedwater pumps. The operator is required to open the MOV on the suction of the pump from the service water system. Makeup from the fire water system is accomplished by running a hose stored in the operator station to the CST.	**Minimal Impact** – The fault tree model represents the sources of water to the AFW pump suction. The water source from the water treatment system has insufficient flow to support AFW pump operation. However, the water source associated with the condenser hotwells is not credited in the model resulting in a minor conservatism.	The assumption has a potentially conservative impact on the calculated RICT for the AFW system, since AFW may provide some mitigating capability if service water and fire protection are not available. This would have a very small effect since it would only be credited after all 6 SW pumps, the DDFP and MDFP all fail to provide water to the suction of the AFW pumps. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.
Operator actions to control AFW flow later in an accident sequence are not explicitly modeled in the AFW system fault trees.	**Potential Impact** - Operator actions are necessary to prevent the AFW system from overfilling the steam generators as their pressures decrease and AFW flow increases. This was not modeled since there is a long time available and the function would be alarmed. In addition, the operator would have to successfully supply an alternate source of water to the suction of the AFW pumps (not automatic) and then forget to control flow or check steam generator level.	Although non-conservative, subsequent failure to maintain a plant parameter after initial success is judged to have a negligible impact on the results. Therefore, no additional considerations are required to address this source of uncertainty in the RICT Program.

ENCLOSURE 10

Point Beach Units 1 and 2
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Program Implementation

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1.0 Purpose

Section 4.0, Item 10 of the NRC Final Safety Evaluation (Reference 1) for NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0 (Reference 2), requires that the license amendment request (LAR) provide a description of the implementing programs and procedures regarding plant staff responsibilities for the Risk Managed Technical Specifications (RMTS) Implementation, and specifically discuss the decision process for risk management action (RMA) implementation during a Risk-Informed Completion Time (RICT). Several procedures and processes are detailed in other enclosures that are not repeated in this enclosure addressing Probabilistic Risk Assessment (PRA) Model Update, Cumulative Risk Assessment, Monitoring Program, and Risk Management Actions.

This enclosure provides a description of the implementing programs and the administrative controls and procedures regarding the plant staff responsibilities for the RICT Program, including training of plant personnel, and specifically discusses the decision process for RMA implementation during extended Completion Times (CT).

2.0 Revision Summary

Initial Issuance.

3.0 Evaluation

3.1 RICT PROGRAM AND PROCEDURES

NextEra will develop a program description and implementing procedures for the RICT Program. The program description will establish the management responsibilities and general requirements for risk management, training, implementation, and monitoring of the RICT program. More detailed procedures will provide specific responsibilities, limitations, and instructions for implementing the RICT program. The program description and implementing procedures will incorporate the programmatic requirements for Risk Managed Technical Specifications included in NEI 06-09.

The Operations Department is responsible for compliance with the Technical Specifications (TS) and will be responsible for implementation of RICTs and risk management actions (RMA). Entry into the RICT program will require management approval prior to pre-planned activities and as soon as practical following emergent conditions.

The procedures for the RICT program will address the following attributes consistent with NEI 06-09:

- Plant management positions with authority to approve entry into the RICT Program.
- Important definitions related to the RICT Program.
- Departmental and position responsibilities for activities in the RICT Program.
- Plant conditions for which the RICT Program is applicable.
- Limitations on implementing RICTs under voluntary and emergent conditions.

- Implementation of the RICT Program 30-day back stop limit.
- Use of the Configuration Risk Management Program (CRMP) tool.
- Guidance on recalculating RICT and risk management action time within 12 hours or within the most limiting front-stop CT after a plant configuration change.
- Requirements to identify and implement RMAs when the Risk Management Action Time (RMAT) is exceeded or is anticipated to be exceeded.
- Guidance on the use of RMAs including the conditions under which they may be credited in RICT calculations.
- Conditions for exiting a RICT.
- Requirements for training on the RICT Program.
- Documentation requirements related to individual RICT evaluations, implementation of extended CTs, and accumulated annual risk.

3.2 RICT Program Training

The scope of training for the RICT Program will include rules for the new TS program, CRMP software, TS Actions included in the program, and procedures. This training will be conducted for the following NextEra personnel:

3.2.1 <u>Site Personnel</u>

- Operations Director
- Operations Personnel (Licensed and Non- Licensed)
- Operations Training
- Selected Maintenance Personnel
- Engineering
- Other Selected Management

3.2.2 <u>Corporate Personnel</u>

- General Manage Nuclear Support and Regulatory Affairs
- Director Regulatory Affairs
- Licensing Management and Personnel
- Director Online Work Management
- Manager Online Work Management North
- Work Week Coordinators
- Cycle Coordinator
- Director Outage North
- Outage Supervisors North
- Planning and Scheduling Personnel
- Probabilistic Risk Assessment Manager and Personnel
- Training Management and Personnel
- Other Selected Management

Training will be carried out in accordance with NextEra training procedures and processes. These procedures were written based on the Institute of Nuclear Power Operations (INPO) Accreditation (ACAD) requirements, as developed and maintained by the National Academy for Nuclear Training. NextEra has planned three levels of training for implementation of the RICT Program. They are described below:

3.2.3 Level 1 Training

This is the most detailed training. It is intended for the individuals who will be directly involved in the implementation of the RICT Program. This level of training includes the following attributes:

- Specific training on the revised TS
- Record keeping requirements
- Case studies
- Hands-on experience with the CRMP tool for calculating RMAT and RICT
- Identifying appropriate RMAs
- Common cause failure considerations
- Other detailed aspects of the RICT Program

3.2.4 Level 2 Training

This training is applicable to supervisors, managers, and other personnel who need a broad understanding of the RICT Program. It is significantly more detailed than level 3 training (described below), but it is different from level 1 training in that hands-on time with the CRMP tool and case studies are not included. The concepts of the RICT Program will be taught, but this group of personnel will not be qualified to perform the tasks for actual implementation of the RICT Program.

3.2.5 Level 3 Training

This training is intended for the remaining personnel who require an awareness of the RICT Program. These employees need basic knowledge of RICT Program requirements and procedures. This training will cover RICT Program concepts that are important to disseminate throughout the organization.

4.0 References

1. Letter from the NRC to NEI, “Final Safety Evaluation for Nuclear Energy Institute (NEI) Topical Report (TR) NEI 06-09, ‘Risk-Informed Technical Specifications Initiative 4B, Risk-Managed Technical Specifications (RMTS) Guidelines’ (TAC No. MD4995)”, dated May 17, 2007 (ADAMS Accession No. ML071200238),.

2. NEI Topical Report NEI 06-09-A, “Risk-Informed Technical Specifications Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines”, Revision 0, dated October 2012 (ADAMS Accession No. ML12286A322).