UNITED STATES
NUCLEAR REGULATORY COMMISSION
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September 19, 2014

Mr. Joseph E. Pacher
Vice President R.E. Ginna Nuclear Power Plant
R.E. Ginna Nuclear Power Plant, LLC
1503 Lake Road
Ontario, NY 14519

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – SECOND ROUND OF REQUEST FOR ADDITIONAL INFORMATION CONCERNING REQUEST TO ADOPT NATIONAL FIRE PROTECTION ASSOCIATION STANDARD NFPA-805, "PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION FOR LIGHT WATER REACTOR ELECTRIC GENERATING PLANTS," (TAC NO. MF1393)

Dear Mr. Pacher

By letter dated March 28, 2013, (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13093A064), R. E. Ginna Nuclear Power Plant, LLC, (Ginna, the Licensee), submitted a license amendment request (LAR). The licensee requested approval of an amendment to the Ginna license that would facilitate transition of the fire protection-licensing basis at Ginna from Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.48(b), to 10 CFR 50.48(c), National Fire Protection Association Standard NFPA 805 (NFPA 805).

The proposed amendment requests the U.S. Nuclear Regulatory Commission (NRC) staff's review and approval for adoption of a new fire protection licensing basis which complies with the requirements in 10 CFR 50.48(a) 10 CFR 50.48(c), and Regulatory Guide 1.205, Revision 1, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants," December 2009 (ADAMS Accession No. ML092730314).

The NRC staff has reviewed the information provided by Ginna in its application, and the NRC staff also participated in an audit from August 26 to August 29, 2013, and has determined that additional information is needed to complete its review.

Enclosed is the NRC staff's second round of request for additional information (RAI).

A clarification conference call was held between the NRC staff and the licensee on August 7, 2014, and it was agreed that the licensee's staff would respond to all RAIs (except for RAI 44.01.b) by September 26, 2014. During a telephone call held on September 12, 2014, licensee staff agreed to respond to RAI 44.01.b contingent upon the NRC staff's timely review (2-3 weeks) and feedback to the licensee regarding its submittal of September 26, 2014.

Please note that review efforts on this task are being continued and additional RAIs may be forthcoming.

Please contact me at (301) 415-1476 or via email to Mohan.Thadani@nrc.gov, if you have any questions on this issue.

Sincerely,

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

cc w/encl: Distribution via Listserv

REQUEST FOR ADDITIONAL INFORMATION

LICENSE AMENDMENT REQUEST TO ADOPT

NATIONAL FIRE PROTECTION ASSOCIATION STANDARD 805

PERFORMANCE-BASED STANDARD FOR FIRE PROTECTION

FOR LIGHT-WATER REACTOR ELECTRIC GENERATING PLANTS

EXELON CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

(TAC NO. MF1393)

Probabilistic Risk Assessment (PRA) RAI 03.01

In a letter dated February 28, 2014 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML14065A033), the licensee responded to PRA RAI 03. In the response to PRA RAI 03 the licensee stated that the Fire PRA model includes credit for the upgraded Westinghouse shutdown seal (SDS) that is outlined in Revision 1 to Pressurized Water Reactor Owners Group Topical Report WCAP-17100-P/NP, "PRA Model for the Westinghouse Shut Down Seal" (ADAMS Accession Number ML110770017). Given the July 26, 2013, 10 CFR Part 21 notification by Westinghouse (ADAMS Accession Number ML13211A168) concerning defects with the SDS performance, discuss if there have been any new developments regarding the status of the SDS performance. The response to PRA RAI 03 notes that an alternative modification may be implemented if necessary to meet the risk acceptance criteria. The NRC recognizes that License Amendment Request (LAR) Attachment S, Table S-3, Implementation Item 9 would eventually require the risk acceptance guidelines to be met and is intended as a verification of an acceptable existing design, not a placeholder for a future evaluation. Please address the following:

a. Indicate the type of Westinghouse SDS model to which the Topical Report (TR) is to be applied. Provide relevant information from technical design documents, testing evaluations, draft topical reports, etc., that support the incorporation and quantification of the SDS performance in the Ginna Fire PRA model. Clearly indicate what is being credited from the TR. As appropriate, justify any assumptions for new risk reduction credit or retention of credit previously assumed via WCAP-17100-P/NP, Rev. 1, consistent with its final safety evaluation (SE) (ADAMS Accession Number ML110880526), or other NRC endorsed technical bases. In addition, clearly describe and justify deviations, if any, from the TR.

b. If the credit is dependent upon plant operational experience with the upgraded SDS, discuss to what extent credit can and will be taken prior to installation/upgrade and completion of any required operational duration necessary to justify such credit.

c. If the RCP shutdown seal reliability is not known or determined (i.e., there is no technical basis, such as engineering evaluation, Topical Report and/or vendor test data), then perform a sensitivity study to remove any credit for the RCP shutdown seal. If the RG 1.174 risk acceptance criteria cannot be met, then alternative modification(s) for transition may be considered; however, Table S-2 of the LAR would need to be updated accordingly and the re-evaluation submitted.

PRA RAI 05.01

Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants" (ADAMS Accession Number ML092730314) (RG 1.205) provides guidance on the acceptability of previously approved recovery action risk. According to LAR, Attachment W, Table W-4, the delta core damage frequency (ΔCDF) for the additional risk of recovery actions is 1.01E-5/yr. In a letter dated December 17, 2013 (ADAMS Accession Number ML13353A417), the licensee responded to PRA RAI 05. According to the licensee's response, no recovery actions were credited in the delta risk evaluation as being previously approved. Clarify whether this means that previously approved recovery actions were excluded from the additional risk of recovery actions, or whether they were included as all other recovery actions. If they were excluded, provide an estimate of the additional risk of previously approved recovery actions.

PRA RAI 11.01

In the letter dated February 28, 2014, (ADAMS Accession No. ML14065A033), the licensee responded to PRA RAI 44 indicating that changes were made to the VFDR modeling, which impacted the delta risk calculations. Please address the following with respect to the noted change:

a. "VFDR modeling was updated to account for higher spurious actuation likelihoods for loss of coolant accident (LOCA) related issues . . ." Explain this statement.

b. It is stated in response to PRA RAI 44 that, ". . . due to the increase in spurious actuation likelihood, deterministic resolution is no longer credited." Please explain how this relates to the table of modifications discussed in the response to PRA RAI 01 in the letter dated December 17, 2013, (ADAMS Accession No. ML13360A200).

c. Explain the following statement from the response to PRA RAI 44: "This change only impacted the delta risk calculations. It did not affect the risks associated with the transition plant. Only the compliant plant risk is impacted."

PRA RAI 11.02

In the letter dated December 17, 2013, the licensee response to PRA RAI 11 provided a high-level description of the method for evaluating delta risk for fire scenarios. However, further detailed information is requested for evaluating the main control room (MCR) fire related risk.

a. Describe for the MCR risk analysis whether a screening conditional core damage probability (CCDP)/conditional large early release probability (CLERP) approach was used or if fire scenarios were developed and quantified using the Fire PRA model logic.

b. Discuss the approach used to develop and apportion frequencies to the main control board (MCB) fire scenarios. If back panels are included in the MCB provide justification. Also, indicate if the "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, Final Report" (ADAMS Accession Number ML052580075) (NUREG/CR-6850) Bin 4 frequency was used for the MCB back panels.

c. Note whether MCR abandonment scenarios were modeled for loss of habitability and for loss of control. Describe the treatment of each in the MCR risk analysis. For the particular fire scenarios noted in response to PRA RAI 32 in which a full panel or a full panel and adjacent panels may be lost, do such fire scenarios represent a loss of control MCR abandonment scenario?

d. Provide the range of CCDPs and CLERPs that have been developed. In addition, explain how the process and the range of estimates developed relate to the MCR abandonment scenario bins below:

    i. Scenarios where few functions fail as a result of the fire aside from MCR habitability, and successful shutdown is relatively uncomplicated by the fire scenario.

    ii. Scenarios where the fire could cause some recoverable functional failures or spurious operations that complicate the shutdown.

    iii. Scenarios where the fire induced failures cause great difficulty for shutdown by failing multiple functions and/or causing complex spurious operations.

In addition, in the letter dated December 17, 2013, the response to PRA RAI 07 appears to indicate no reliance on previously approved recovery actions taken at the intermediate building emergency level instrumentation panel (IBELIP) and the auxiliary building emergency local instrumentation panel (ABELIP) primary control stations (PCSs) to keep one train free from fire damage.

a. Describe how IBELIP and ABELIP PCSs are credited in the MCR risk analysis and any key assumptions related to them. For example, how are they credited with respect to fire damage and any impact on the use of these PCSs or their associated safe shutdown equipment as described in Attachment C of the LAR.

b. Are these PCSs assumed to not be affected by fire scenarios and ready for use if the new recovery actions are not successful?

PRA RAI 12.01

In the letter dated December 17, 2013, the licensee responded to PRA RAI 12, addressing, in part, the methodology for applying screening human reliability analysis (HRA) values to in-MCR actions for fire scenarios, which do not require MCR abandonment due to equipment impact or the MCR environment. The methodology is different from both the HRA screening approach for in-MCR actions discussed in the "EPRI/NRC-RES Fire Human Reliability Analysis Guidelines" (ADAMS Accession Number ML093350494) (NUREG-1921) and NUREG/CR-6850 guidance. The response notes that in-MCR human failure events (HFEs) in dominant cutsets were reviewed and, where justifiable, a detailed HRA was performed. The NRC staff's review of the applied HRA screening approach notes the lack of the timing element. The screening approaches for NUREG-1921 and for NUREG/CR-6850 both recognize that certain fire scenarios may require some action(s) in the short term while the fire may be on-going. As such, these approaches assign higher HRA screening probabilities for action(s) in the first hour or the first half-hour, for example, than action(s) which may be taken at a later time. If this time dependent aspect of the screening probability were to be used for in-MCR fires, which do not result in abandonment, then different dominant cutsets and HFEs may be identified than those found using the HRA screening described in the response to PRA RAI 12. Please clarify whether the HRA screening methodology for the in-MCR actions considers timing dependence or update the screening probabilities to incorporate time dependence and explain your methodology.

PRA RAI 19.01

In the letter dated February 28, 2014, the sensitivity analyses results presented in the licensee's response to PRA RAI 19 includes the requested sensitivity analysis from PRA RAI 38 regarding fire ignition frequencies provided in Table 6-1 of NUREG/CR-6850. The result shown in Table 3, "Summary of Sensitivity Studies," in the PRA RAI 19 response, is a ΔCDF of 1.3E-5/yr. above the baseline delta risk of 9.36E-6/yr. This result exceeds the ΔCDF criteria in RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (ADAMS Accession Number ML003740133) (RG 1.174), in which case Frequently Asked Question 08-0048, "Revised Fire Ignition Frequencies" (ADAMS Accession Number ML092190457) (FAQ 08-0048) notes that the situation is to be addressed by considering fire protection or related measures that can be taken to provide additional defense in depth (DID). Neither the response to PRA 10 or PRA RAI 38 discusses measures that can be taken to provide additional DID. Please discuss these additional measures.

PRA RAI 22.01

In the letter dated February 28, 2014, the licensee response to PRA RAI 22.c noted that LAR Attachment W, Table W-3, shows that interfacing systems loss-of-coolant accident (ISLOCA) is a dominant contributor to the large early release frequency (LERF). The response to PRA RAI 22.c states that the updated model results, which are documented in the response to PRA RAI 44, were reviewed and ISLOCA scenarios are no longer a contributor to the LERF. Provide a discussion explaining this observation including model changes or assumptions.

PRA RAI 30.01

In the letter dated February 28, 2014, the licensee responses to PRA RAIs 15 and 30 describe the approach taken for modeling ignition sources in transient fire zones. Provide the following additional information related to fixed ignition sources such as cables and junction boxes within a transient zone.

a. Discuss the approach by which ignition frequencies, developed on a compartment or fire-zone basis, are apportioned to cables and junction boxes within a transient zone.

b. Confirm that suppression has not occurred prior to the damage of the entire tray for fires from welding and cutting and for self-induced cable fires, and prior to damage for cables associated with junction boxes.

c. Discuss how the CCDP is assigned to a junction box and whether it bounds the CCDP associated with all of the cables that enter the junction box. If not, justify the approach.

Note that FAQs 13-0005, "Cable Fires Special Cases: Self-Ignited and Caused by Welding and Cutting," (ADAMS Accession Number ML13129A348) and 13-0006, "Modeling Junction Box Scenarios in a Fire PRA" (ADAMS Accession Number ML13182A711) offer technical resolutions related to cable fires and junction boxes, respectively.

PRA RAI 32.01

In the letter dated December 17, 2013, the licensee's response to PRA RAI 32 discusses an approach for modeling MCB damage probabilities.

a. Describe if the timing assumed in the methodology is consistent or conservative with respect to the fire modeling of the MCB (e.g., the fire modeling described in the response to FM RAI 01.3.d.). Alternately, describe the overall MCB evaluation and justify if the evaluation is conservative or consistent with acceptable methods for frequency, likelihood, and CCDP.

b. Confirm that the methodology discussed is used in both the compliant plant model and the post-transition plant model. If not, discuss the methodology to be used for post-transition. Also, identify if the same equipment failures are assumed in both the compliant and post-transition model.

PRA RAI 33.01

In the letter dated December 17, 2013, the licensee response to PRA RAI 33 discussed a basis for the peer review assigning Not Applicable to supporting requirements (SRs) PRM-B3, PRM-B4, PRM-B6, PRM-B8, PRM-B9, PRM-B10, and PRM-B15. The NRC staff requests clarification of the following:

a. PRM-B3. For equipment modeled in the Fire PRA but not in the internal event (IE) PRA, can such equipment lead to a new IE, such as from a spurious actuation?

b. PRM-B6. The SR in the ASME/ANS-RA-Sa-2009 Standard for Level 1/ Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications also applies to existing accident sequences that require modification (by reference to PRM-B5) so that noting that no new event tree models were developed does not sufficiently address the non-applicability of this SR. In light of this clarification, indicate whether a peer review is necessary for PRM-B6.

c. PRM-B8. With respect to the Fire PRA success criteria, does crediting the diesel-driven pump identified in the response to PRA RAI 44, qualify as a new success criteria, which needs peer review?

PRA RAI 40.01

In the letter dated February 28, 2014, the licensee response to PRA RAI 40 noted that a review was performed and identified several common cause groups that are not tested on a staggered basis. These items had their common cause factors increased by a factor of two in the model. Based on the common cause groups, common cause failure (CCF) probability formulas, and data for the alpha factor method in NUREG/CR-5497, "Common Cause Failure Parameter Estimations" (ADAMS Accession Number ML070580356), a factor of two approximates a group size of two. However, for larger group sizes, a factor of two underestimates the CCF probability with this method. Confirm that the appropriate multiplying factor is being used for its associated group size if it belongs to a group not tested on a staggered basis.

PRA RAI 44.01

Section 2.4.3.3 of the NFPA 805 standard incorporated by reference into 10 CFR 50.48(c) states that the probabilistic safety assessment (PSA) (PSA is also referred to as PRA) approach, methods, and data shall be acceptable to the authority having jurisdiction (AHJ), which is the NRC. RG 1.205 identifies NUREG/CR-6850; NEI 04-02, Revision 2, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program Under 10 CFR 50.48(c)" (ADAMS Accession Number ML081130246) and the ongoing FAQ process as documentation acceptable to the staff for adopting a fire protection program consistent with NFPA 805.

The NRC staff identified several methods and weaknesses that were used in the Fire PRA. RAIs were provided about these methods and weaknesses and the responses have been reviewed. The staff noted that the justification provided for the methods and weaknesses listed below is not complete. In addition, in some instances, aspects of the Fire PRA model may have been incomplete at the time of the LAR submittal, but subsequently completed in response to an RAI; these are also included in the list below.

Methods and weaknesses:

- PRA RAI 03 (as clarified by PRA RAI 03.01) regarding modeling of Westinghouse shutdown seals
- PRA RAI 18 regarding suppression system impacts on equipment inclusion in the Fire PRA.
- PRA RAI 41 regarding the inclusion of the Intake structure in the Fire PRA.
- PRA RAI 42 regarding the multi-compartment analysis CDF screening.
- PRA RAI 44 regarding adding conduits in adjacent transient zones (related PRA RAI 15 and 30).
- PRA RAI 44 regarding re-evaluating compartments with a conditional core damage probability of 1.0 (related PRA RAI 22).

The following methods and weaknesses have been identified, but the NRC staff review is continuing with additional RAIs and further supporting information has been requested. Alternatively, the licensee may replace any of these methods and weaknesses with another method by modifying the Fire PRA model. In addition to methods and weaknesses, model updates under review are also included below.

Methods and weaknesses still under review:

- PRA RAI 11.02 regarding the Fire PRA update on spurious actuation likelihoods.
- PRA RAI 11.03 regarding the MCR risk analysis.
- PRA RAI 12.01 regarding main control room HRA screening methodology for in-MCR actions.
- PRA RAI 19.01 regarding the potential measures for defense in depth in considering the results of the NUREG/CR-6850 fire ignition frequencies sensitivity analysis.
- PRA RAI 22.01 regarding ISLOCA risk insights.
- PRA RAI 30.01 regarding cables and junction boxes within a transient zone.
- PRA RAI 32.01 regarding the MCB methodology for calculating conditional damage probabilities.
- PRA RAI 40.01 regarding common cause failure probability modeling of non-staggered testing.

a. For each method (i.e., each bullet) above, explain how the issue will be addressed in (i) the final composite analysis results provided in support of the LAR and (ii) the PRA that will be used at the beginning of the self-approval of post-transition changes. In addition, provide a method to ensure that all changes will be made, that a focused-scope peer review will be performed on changes that are PRA upgrades as defined in the PRA standard, and that any findings will be resolved before self-approval of post-transition changes.

b. Provide the results of a composite analysis that shows the integrated impact on the fire risk (CDF, LERF, ΔCDF, ΔLERF) after replacing identified methods and weaknesses. As the review process is concluded, additional changes to replace any method or weakness still under review may be required. In this composite analysis, for those cases where the individual issues have a synergistic impact on the results, a simultaneous analysis must be performed. For those cases where no synergy exists, a one-at-a-time analysis may be done. If the impact on the change in risk from transition is negligible, a quantitative evaluation is unnecessary.

PRA RAI 46.01

LAR Attachment S, Table S-3, Implementation Item 9 is related to updating the Fire PRA model to reflect modifications and ensure that impacts to risk estimates will be verified with acceptable methods.

a. Describe the method that will be used to ensure that any PRA upgrade will be peer-reviewed prior to using the model for post-transition change evaluations.

b. With respect to Fire PRA credited implementation items in Tables S-2 and S-3 of the LAR, verify the validity of the change-in risk (total modifications) provided in LAR Attachment W. If this verification determines that the risk metrics have changed such that the risk metrics from Attachment W are exceeded, provide justification describing how the modification is acceptable. Provide the reference to Tables S-2 and S-3 in the implementation item.

Describe an implementation item, or modification to Implementation Item 9, which describes how the Fire PRA updates identified in PRA RAI 44.01 and associated documentation, will be completed prior to self-approval.

Please contact me at (301) 415-1476 or via email to Mohan.Thadani@nrc.gov, if you have any questions on this issue.

Sincerely,

/RA/

Mohan C. Thadani, Senior Project Manager
Plant Licensing Branch I-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

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