UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

May 27, 2022

Mr. David P. Rhoades
Senior Vice President
Constellation Energy Generation, LLC
President and Chief Nuclear Officer
Constellation Nuclear
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – SUMMARY OF COMBINED REGULATORY AUDIT IN SUPPORT OF RISK-INFORMED COMPLETION TIMES IN TECHNICAL SPECIFICATIONS LICENSE AMENDMENT REQUEST AND TO ADOPT 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPIDS L-2021-LLA-0091 AND L-2021-LLA-0092)

Dear Mr. Rhoades:

By applications dated May 20, 2021, Exelon Generation Company, LLC submitted two license amendment requests (LARs) for R.E. Ginna Nuclear Power Plant. The first LAR would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21140A324). In the second LAR the licensee proposed the license condition to the Renewed Facility Operating Licenses (RFOLs) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"(ADAMS Accession No. ML21141A009).

Subsequently, on February 1, 2022 (ML22032A333), Exelon was renamed Constellation Energy Generation, LLC.

The U.S. Nuclear Regulatory Commission (NRC) staff conducted a regulatory audit to support its review of the LARs. The NRC staff issued its audit plan on August 25, 2021 (ML21222A114). A summary of this audit is enclosed.

If you have any questions, please contact me at (301) 415-2597 or by e-mail to V.Sreenivas@nrc.gov.

Sincerely,

/RA/

V. Sreenivas, Project Manager
Plant Licensing Branch I
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure:
Audit Summary

cc: Listserv

REGULATORY AUDIT SUMMARY

RELATED TO RENEWED FACILITY OPERATING LICENSE NO. DPR-18

CONSTELLATION GENERATION COMPANY, LLC

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

1.0 BACKGROUND

By applications dated May 20, 2021, Exelon Generation Company, LLC (Exelon) submitted two license amendment requests (LAR) for the R.E. Ginna Nuclear Power Plant. The first LAR would revise technical specification (TS) requirements to permit the use of risk-informed completion times (RICTs) for actions to be taken when limiting conditions for operation (LCOs) are not met (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21140A324). In the second LAR the licensee proposed the license condition to the Renewed Facility Operating Licenses (RFOLs) to allow the implementation of Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors"(ADAMS Accession No. ML21141A009).

Subsequently, on February 1, 2022 (ML22032A333), Exelon was renamed Constellation Energy Generation, LLC.

2.0 AUDIT ACTIVITIES AND OBSERVATIONS

The regulatory audit was conducted by the NRC headquarters staff consistent with the audit plan, (ML21222A114). The purpose of the audit was to (1) gain a better understanding of the LARs to enable staff's review for technical and regulatory acceptability, and (2) identify additional information required from the licensee on the docket to support the NRC staff's licensing or regulatory decision.

The NRC staff provided information needs in the audit plan, as supplemented. The NRC staff reviewed documents during the first phase of the audit that were responsive to the information needs and made available by the licensee in an electronic reading room.

The second phase of the audit was conducted virtually between the dates of September 13 -15, 2022, and subsequently the audit was closed on April 29, 2022, after follow up technical clarifications discussions. Detailed discussions were held on various topics with the audit team members and the licensee, aided by information needs identified by the staff in the audit plan, as supplemented, along with the licensee's presentations and prepared responses to audit questions. The NRC staff used the information gathered from the audit to determine whether additional information is warranted on the docket to support rendering of an NRC staff finding. The NRC staff used these meetings to confirm its understanding of the LAR, discuss the documents in the portal, and determine whether the NRC staff identified any information that needs to be submitted on the docket to complete the NRC staff's safety evaluation.

Enclosure 1 of this audit summary lists the NRC individuals that took part in or attended the audit. Enclosure 2 lists the NRC staff's audit requests and questions. Enclosure 3 lists the documents the NRC audited.

3.0 RESULTS OF THE AUDIT

The NRC staff did not make any regulatory decisions regarding the LARs during the audit. The licensee opted to provide a supplement to docket the additional information identified during the virtual audit week. The NRC staff communicated to the licensee at the virtual audit exit that upon review of the supplement the staff would determine the issuance of any request for additional information (RAI). The licensee supplemented its LARs by letters dated October 14, 2021, and April 28, 2022 (ML21287A006 and ML22118B143, respectively). The NRC staff continues to review the information provided in supplement from the licensee to determine if the NRC further needs any additional information to complete its review.

List of Audit Participants (NRC)

NRR Staff	Email	Division/Branch
V. Sreenivas	V.Sreenivas@nrc.gov	DORL/LPL1
Adrienne Brown	Adrienne.Brown@nrc.gov	DRA/APLA
Jonathan Evans	Jonathan.Evans@nrc.gov	DRA/APLA
Alex Schwab	Alexander.Schwab@nrc.gov	DRA/APLA
Bernard Grenier	Bernard.Grenier@nrc.gov	DRA/APLB
Milton Valentin-Olmeda	Milton.Valentin@nrc.gov	DRA/APLC
De (Wesley) Wu	De.Wu@nrc.gov	DRA/APLC
Vijay Goel	Vijay.Goel@nrc.gov	DEX/EEEB
Ming Li	Ming.Li@nrc.gov	DEX/EICB
Norbert Carte	Norbert.Carte@nrc.gov	DEX/EICB
Gurjendra Bedi	Gurjendra.Bedi@nrc.gov	DEX/EMIB
Kaihwa Hsu	Kaihwa.Hsu@nrc.gov	DEX/EMIB
Dan Widrevitz	Dan.Widrevitz@nrc.gov	DNRL/NVIB
David Nold	David.Nold@nrc.gov	DSS/SCPB
Angelo Stubbs	Angelo.Stubbs@nrc.gov	DSS/SCPB
Summer Sun	Summer.Sun@nrc.gov	DSS/SNSB
Joshua Wilson	Joshua.Wilson@nrc.gov	DSS/STSB
Andrea Russel	Andrea.Russell@nrc.gov	DSS/STSB
Garill Coles	Garill.Coles@pnnl.gov	PNNL
Fred Forsaty	Fred.Forsaty@nrc.gov	DSS/SNSB

List of Audit Questions and Requests

The U.S. Nuclear Regulatory Commission (NRC) staff requested the following information during its audit to support its review of the license amendment request (LAR).

PROBABILISTIC RISK ANALYSIS (PRA) LICENSING BRANCH A (APLA) PRA ACCEPTABILITY AND RISK-INFORMED APPROACH

APLA QUESTION 01 – Internal Events and Internal Flooding Peer Review [10 CFR 50.69 and TSTF-505 Applicable]

Regulatory Guide 1.200, Revision 2 (ML090410014) and Revision 3 (ML19308B636) provide guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the ASME/ANS PRA standard ASME/ANS-RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as one acceptable approach for determining the technical acceptability of the PRA.

For 10 CFR 50.69, LAR Section 3.3, states that the “Ginna FPIE PRA model was peer reviewed in June 2009 using the NEI 05-04 process, the PRA Standard (ASME/ANS RA-Sc-2007) and Regulatory Guide 1.200, Revision 1.” For TSTF-505, LAR Enclosure 2, states that the “Ginna Internal Events PRA model was peer reviewed in June 2009 using the NEI 05-04 process, the PRA Standard (ASME/ANS RA-Sc-2007) and Regulatory Guide 1.200, Revision 1.” The Ginna National Fire Protection Association (NFPA) 805 program NRC Safety Evaluation (SE) dated November 23, 2015 (ML15271A101), states that the June 2009 internal events PRA peer review was performed against the ASME RA-Sb-2005 version of the PRA Standard. Peer review Report LRT-RAM-II-049 indicates that the June 2009 internal events PRA peer review was performed relative to the ASME RA-Sb-2005 version of the PRA Standard. The Ginna NFPA 805 SE states that “the licensee performed a gap assessment between the ASME RA-Sb-2005 as clarified by RG 1.200, Revision 1 and ASME/ANS RA-Sa-2009 as clarified by RG 1.200, Revision 2,” and that the “licensee did not identify any significant issues for the Fire. Probabilistic Risk Assessments (FPRA) from this gap assessment.” However, the Ginna NFPA 805 Licensing Amendment Request (LAR) (ADAMS Accession No ML13093A066) states concerning the gaps between the two versions of the PRA Standard and RG 1.200 that the “most significant changes [in requirements] occurred in the Internal Flooding portion” and it acknowledged differences in internal event and fire PRA requirements between the two versions of the PRA Standard that did not impact the NFPA 805 application.

In light of these observations:

a. Clarify which version of PRA Standard was used in the June 2009 internal events PRA (including flooding) peer review.

b. Justify that the differences in the Supporting Requirements in the version of the PRA standard used for the June 2009 internal events peer review as clarified by RG 1.200, Revision 1 and the 2009 version of the PRA Standard endorsed by RG 1.200, Revision 2 have an inconsequential impact on the RICT calculations. Include a description of the results of a gap assessment that was performed to evaluate the impact of the gap on the RICT program.

APLA QUESTION 02 – Dispositions of PRA Model Assumptions and Sources of Uncertainty [10 CFR 50.69 and TSTF-505]

The NRC staff SE to NEI 06-09, Revision 0, specifies that the LAR should identify key assumptions and sources of uncertainty and to assess and disposition each as to their impact on the Risk Managed Technical Specifications (RMTS) application. Paragraphs 50.69(c)(1)(i) and (c)(1)(ii) of 10 CFR require that a licensee's PRA be of sufficient quality and level of detail to support the structures, systems and components (SSC) categorization process, and that all aspects of the integrated, systematic process used to characterize SSC importance must reasonably reflect the current plant configuration and operating practices, and applicable plant and industry operational experience. NUREG-1855, Revision 1, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making, Final Report," dated March 2017 (ML17062A466) presents guidance on the process of identifying, characterizing, and qualitative screening of model uncertainties.

For 10 CFR 50.69, LAR Attachment 6 provides dispositions for candidate key assumptions and sources of uncertainty for this application. LAR Section 3.2.7 states that the "conclusion of this review is that no additional sensitivity analyses are required to address Ginna PRA model specific assumptions or sources of uncertainty" for this application. For TSTF-505, LAR Enclosure 9 provides dispositions for candidate key assumptions and sources of uncertainty in Tables E9-1, E9-2, and E9-3 for this application. In most cases, the TSTF-505 LAR concludes that RICT program calculations are not impacted by the modelling uncertainty and no Risk Managements thresholds (RMAs) are required to address the uncertainty. However, for a few sources of PRA modeling uncertainty identified in the LAR, there is not enough information for NRC staff to conclude that the assumption or source of modeling uncertainty would not have an impact on the risk-informed categorization and RICT calculations. Therefore, address the following:

a) For 10 CFR 50.69, LAR Attachment 6 and TSTF-505, LAR Enclosure 9, Table E9-1 for the internal events PRA indicates that a 24 hour mean-time-to-repair (MTTR) was assumed in support system initiating event trees and was identified as a potential key source of uncertainty. The LARs states that the "use of a 24 MTTR is reasonable and follows industry convention." Supporting Requirement (SR) SY-A24 of the ASME/ANS RA-Sa-2009 PRA standard for general modeling of repair states: "*DO NOT MODEL the repair of hardware faults unless the probability of repair is justified through adequate analysis or examination of data.*" It is not clear to NRC staff whether this source of uncertainty can impact SSC categorization for 10 CFR 50.69 and TSTF-505 for RICT calculations. Therefore, address the following:

    i. Justify that the 24-hour assumption used for the MTTR in the internal events support system initiating event fault trees can be supported by review, analysis, or examinations of data.
    ii. If in response to part (a) above, the 24-hour assumption used for the MTTR in the internal events support system initiating event fault trees cannot be supported, then justify this assumption has an inconsequential impact on the SSC categorization program (i.e., 10 CFR 50.69) and TSTF-505 for RICT calculations.

    iii. As an alternative to item (ii) above, identify appropriate RMAs for this key assumption consistent with the treatment of key assumptions in NEI 06-09-A, prior to implementation of the RICT program.

b) For 10 CFR 50.69, LAR Attachment 6 and TSTF-505, LAR Enclosure 9, Table E9-3 for the FPRA states about Component Selection uncertainty that “a small set of loads associated with uncoordinated cabling were assigned bounding routes.” The LARs also state that a “bounding sensitivity analysis was performed to measure the risk associated with this “bounding routing” treatment and the results show that this source of modeling uncertainty has “no significant impact” on the application. The meaning of the phase “bounding routing” is not clear to NRC staff and the sensitivity study and results were not described in the LARs. NRC staff notes that fire impact on uncoordinated cables may cause the loss of power supplies associated with the uncoordinated circuits and failure of the loads associated with those power supplies. It is not clear how the uncoordinated cabling is modelled in the PRA or how the sensitivity study was performed to address this modeling uncertainty. Therefore, address the following:

i. Explain how uncoordinated cables were modelled in the fire PRA using the “bounding routing” approach. Include explanation of the PRA components that were assumed to fail from fire impact on the uncoordinated cables. Also, justify the “bounding routing” approach addresses PRA components that could fail from fire impact on the uncoordinated cables

ii. Describe the sensitivity study that was performed to evaluate this fire PRA modelling uncertainty. Include explanation of how the sensitivity study addressed PRA components that could fail due to fire impact on the uncoordinated circuits. Provide the quantitative results from the study that demonstrate the modelling uncertainty has an inconsequential impact on the SSC categorization and RICT calculations.

iii. If the response to part (ii) above is that sensitivity shows the modelling uncertainty could impact the RICT calculations, then identify appropriate RMAs for this key assumption consistent with the treatment of key assumptions in NEI 06-09-A, prior to implementation of the SSC categorization and RICT programs.

c) Portal Report G1-MISC-026, “Assessment of Key Assumptions and Sources of Uncertainty for PRA” contains the following statement in Table 4-1 pertaining to fire PRA for Topic #68: “Conservative treatments may lead to non-conservative results in applications involving a measured change in risk (For example, if a component is conservatively assumed to be unavailable, then the delta-risk impact may not be appropriately captured in all cases).” The disposition to this source of modeling uncertainty states that “several refinements have been performed to reduce the amount of conservatism” and that “several sensitivity studies analyses have been performed to ensure and confirm that key assumptions did not skew the fire risk results.” It I s not clear whether assumptions were made in the Ginna fire PRA that certain components or systems were conservatively assumed to be unavailable. However, NRC staff did not identify a sensitivity study that was performed to address this source of modelling uncertainty for RICT calculations which do include delta-risk calculations. Therefore, address the following:

a. Clarify whether components or systems were assumed to be unavailable or failed in the fire or internal events PRA models and whether a sensitivity study exists showing that treatment of this modeling uncertainty has an inconsequential impact on the RICT calculations. If such assumption were made, then identify the components and or system that were assumed to be unavailable or failed, and discuss the results of any sensitivity studies that were performed for this source of modelling uncertainty.

b. If components or systems were assumed to be unavailable or failed in the fire or internal events PRA models and a sensitivity study does not exist showing that treatment of this uncertainty has an inconsequential impact on the RICT calculations, then provide qualitative justification or the results of quantitative sensitivity study demonstrating that treatment of the components and/or systems assumed to be unavailable or failed have an inconsequential impact on the RICT calculations.

c. If it cannot be justified that treatment of the components and/or systems assumed to be unavailable or failed have an inconsequential impact on the RICT calculations, then propose a mechanism that ensures that the impact of these component and/or system availabilities or failures are incorporated into the PRA models before implementation of the RICT program.

APLA QUESTION 03 – Credit for FLEX Equipment and Actions [10 CFR 50.69 and TSTF-505]

The NRC memorandum dated May 30, 2017, "Assessment of the Nuclear Energy Institute 16-06, 'Crediting Mitigating Strategies in Risk-Informed Decision Making,' Guidance for Risk-Informed Changes to Plants Licensing Basis" (ML17031A269), provides the NRC's staff assessment of challenges to incorporating FLEX equipment and strategies into a probabilistic risk assessment (PRA) model in support of risk-informed decision-making in accordance with the guidance of RG 1.200, Revision 2.

With regards to equipment failure probability, in the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 8):

> The uncertainty associated with failure rates of portable equipment should be considered in the PRA models consistent with the ASME/ANS PRA Standard as endorsed by RG 1.200. Risk-informed applications should address whether and how these uncertainties are evaluated.

With regards to human reliability analysis (HRA), NEI 16-06 Section 7.5 recognizes that the current HRA methods do not translate directly to human actions required for implementing mitigating strategies. Sections 7.5.4 and 7.5.5 of NEI 16-06 describe such actions to which the current HRA methods cannot be directly applied, such as: debris removal, transportation of portable equipment, installation of equipment at a staging location, routing of cables and hoses; and those complex actions that require many steps over an extended period, multiple personnel and locations, evolving command and control, and extended time delays. In the May 30, 2017, memorandum, the NRC staff concludes (Conclusion 11):

> Until gaps in the human reliability analysis methodologies are addressed by improved industry guidance, [Human Error Probabilities] HEPs associated with actions for which the existing approaches are not explicitly applicable, such as actions described in Sections 7.5.4 and 7.5.5 of NEI 16-06, along with assumptions and assessments, should be submitted to NRC for review.

With regard to uncertainty, Section 2.3.4 of NEI 06-09, Revision 0-A, states that PRA modeling uncertainties shall be considered in application of the PRA base model results to the RICT program and that sensitivity studies should be performed on the base model prior to initial implementation of the RICT program on uncertainties which could potentially impact the results of a RICT calculation. NEI 06-09, Revision 0-A, also states that the insights from the sensitivity studies should be used to develop appropriate RMAs, including highlighting risk significant operator actions, confirming availability and operability of important standby equipment, and assessing the presence of severe or unusual environmental conditions. Uncertainty exists in PRA modeling of FLEX strategies, related to the equipment failure probabilities for FLEX equipment used in the model, the corresponding operator actions, and pre-initiator failure probabilities. Therefore, FLEX modeling assumptions can be key assumptions and sources of uncertainty for RICTs proposed in this application.

Neither the 10 CFR 50.69 or TSTF-505 LARs state or indicate that FLEX strategies are credited in the PRA models that will be used to support the RICT program or SSC categorization. TSTF-505 LAR Enclosure 4, Section 4 does refer to the availability of FLEX fuel trucks and trailers and portable fuel pumps. Portal report G1-MISC-026 “Assessment of Key Assumptions and Sources of Uncertainty for R. E. Ginna PRA” indicates that credit for FLEX modeling could be a source of generic modeling uncertainty. However, the report states that no PRA credit is modelled for the initiation of the Emergency Response Organization (ERO) and that any use of FLEX equipment would be driven by emergency and abnormal plant procedures. Accordingly, FLEX strategies do not appear to be credited in the PRA models with two noted exceptions. Report G1-MISC-026 states that “[i]n the level 2 models some credit is given for scrubbing of eluent from a steam generator tube rupture.” The report also states that the PRA could also “implicitly” credit ERO response in the Human Reliability Analysis (HRA) for failure of operators in the long term. If these exceptions are the only instances in which FLEX strategies are credited in the PRA models, then address parts (a) and (b) below. If other credit is taken for FLEX, then address parts (c) through (g) below:

a) Describe the FLEX strategy that is used to credit “some” scrubbing of effluent from a steam generator tube rupture (SGTR).

   i. Describe the equipment used to model this credit and explain whether this includes portable or temporary equipment.

   ii. If the response to part (i) above indicates that the equipment used to credit “some” scrubbing of effluent from a SGTR is portable or temporary equipment, then address this equipment using the requests in part (e) and (g) below.

   iii. Describe the operator actions needed to initiate operation of the equipment identified in part (i) above.

   iv. If the response to part (ii) above is that the equipment used to credit “some” scrubbing of effluent from a SGTR is portable or temporary equipment, then address the HRA performed to model actions identified in part (iii) above using the requests in parts (f) and (g) below for these actions.

b) Pertaining to the “implicit” credit is taken for FLEX in modeling operator failure of long-term actions.

i. Describe how “implicit” credit is taken for FLEX in modeling operator failure of long-term actions. Include explanation of the term “implicit” credit and how it is different from taking explicit credit.

ii. Justify that this modeling treatment has an inconsequential impact on the RICT calculations and the categorization of SSCs for 10 CFR 50.69. Alternatively, explain how the results of real time risk model will be used to identify RMAs prior to the implementation of the RICT program, consistent with the guidance in Section 2.3.4 of NEI 06-09, Revision 0-A.

To complete the NRC staff’s review of the FLEX strategies modeled in the PRA, the NRC staff requests the following information for the internal events PRA (includes internal floods) and fire PRA, as appropriate:

c) Discuss whether Ginna has credited FLEX equipment or mitigating actions into the Ginna internal events, including internal flooding, or fire PRA models.

   If not incorporated or their inclusion is not expected to impact the PRA results used in the RICT program and 10 CFR 50.69, no additional response is requested, and remainder of this question is not applicable.

d) Summarize the FLEX strategies including the equipment and actions that have been quantitatively credited for each of the PRA models used to support this application. Include discussion of whether the credited FLEX equipment is portable or permanently installed equipment.

e) Regarding the credited equipment:

   i. Discuss whether the credited equipment (regardless of whether it is portable or permanently installed) are like other plant equipment (i.e. SSCs with sufficient plant specific or generic industry data).

      If all credited FLEX equipment is similar to other plant equipment credited in the PRA (i.e., SSCs with sufficient plant-specific or generic industry data), responses to items ii and iii below are not necessary.

   ii. Discuss the data and failure probabilities used to support the modeling and provide the rationale for using the chosen data. Discuss whether the uncertainties associated with the parameter values are in accordance with the ASME/ANS PRA Standard as endorsed by RG 1.200, Revision 2.

   iii. Perform, justify, and provide results of specific sensitivity studies that assess impact on RICT or SSC categorization for 10 CFR 50.69 due to FLEX equipment data and failure probabilities. Part of the response include the following:

      1. For 10 CFR 50.69 and TSTF-505, justify values selected for the sensitivity studies, including justification of why the chosen values constitute bounding realistic estimates.

      2. For TSTF-505, provide numerical results on specific selected RICTs and discussion of the results.

3. For TSTF-505, describe how the results of the sensitivity studies will be used to identify RMAs prior to the implementation of the RICT program, consistent with the guidance in Section 2.3.4 of NEI 06-09, Revision 0-A.

f) Regarding human reliability analysis (HRA), address the following:

i. Discuss whether any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06.

If any credited operator actions related to FLEX equipment contain actions described in Sections 7.5.4 and Sections 7.5.5 of NEI 16-06, answer either item ii or iii below:

ii. Perform, justify and provide results of LCO specific sensitivity studies that assess impact from the FLEX independent and dependent HEPs associated with deploying and staging FLEX portable equipment on the RICTs proposed in this application and the 10 CFR 50.69 program. Response should include the following:

1. Justify independent and joint HEP values selected for the sensitivity studies provided to support 10 CFR 50.69 and TSTF-505, including justification of why the chosen values constitute bounding realistic estimates.

2. Provide numerical results on specific selected RICTs and discussion of the results.

3. Discuss composite sensitivity studies of the RICT results and SSC categorization for the operator action HEPs and the equipment reliability uncertainty sensitivity study provided in response to part (4.c.ii) above.

4. Describe how the source of uncertainty due to the uncertainty in FLEX operator actions HEPs will be addressed in the RICT and 10 CFR 50.69 programs. For TSTF-505, describe specific RMAs being proposed, and how these RMAs are expected to reduce the risk associated with this source of uncertainty.

iii. Alternatively, to item ii above, provide information associated with the following items listed in supporting requirements (SR) HR-G3 and HR-G7 of the ASME/ANS RA-Sa-2009 PRA Standard to support detailed NRC review:

1. the level and frequency of training that the operators and/or non-operators receive for deployment of the FLEX equipment (performance shaping factor (a)),

2. performance shaping factor (f), regarding estimates of time available and time required to execute the response,

3. performance shaping factor (g) regarding complexity of detection, diagnosis and decision making and executing the required response,

4. Performance shaping factor (h) regarding consideration of environmental conditions, and

5. Human action dependencies as listed in SR HR-G7 of the ASME/ANS RA-Sa-2009 PRA Standard.

g) The ASME/ANS RA-Sa-2009 PRA standard defines PRA upgrade as the incorporation into a PRA model of a new methodology or significant changes in scope or capability that impact the significant accident sequences or the significant accident progression sequences. Section 1-5 of Part 1 of ASME/ANS RA-Sa-2009 PRA Standard states that upgrades of a PRA shall receive a peer review in accordance with the requirements specified in the peer review section of each respective part of this Standard. Provide an evaluation of the model changes associated with incorporating FLEX mitigating strategies, which demonstrates that none of the following criteria is satisfied: (1) use of new methodology, (2) change in scope that impacts the significant accident sequences or the significant accident progression sequences, and (3) change in capability that impacts the significant accident sequences or the significant accident progression sequences.

APLA QUESTION 04 – In-Scope LCOs and Corresponding PRA Modeling [TSTF-505]

The NRC SE to NEI 06-09 specifies that the LAR should provide a comparison of the TS functions to the PRA modeled functions to show that the PRA modelling is consistent with the licensing basis assumptions or to provide a basis for when there is a difference. LAR Enclosure 1, Table E1-1 identifies each TS LCO proposed for the RICT program, describes whether the systems and components participating in the TS LCO are implicitly or explicitly modeled in the PRA, and compares the design basis and PRA success criteria. For certain TS LCO Conditions, the table explains that the associated SSCs are not modelled in the PRAs but will be represented using a surrogate event that fails the function performed by the SSC. For some LCO conditions, the LAR did not provide enough description for NRC staff to conclude that the PRA modeling will be sufficient for each proposed LCO Condition. Therefore, address the following:

a) TSTF-505 LAR Table E1-1 states for TS LCO 3.4.11 ("Each power-operated relief valve (**PORV**) PORV and associated block valve shall be OPERABLE") Condition C ("One block valve inoperable") and Condition D ("Both block valves inoperable") that the design basis success criteria is "PORV on the same train operable" and "2/2 PORVs operable," respectively. The LAR also states that the PRA success criterion in both cases is "1/2 block valves required to ensure isolation of affected PORV LOCA." Based on the information provided in the table is not clear whether the PRA success criterion is consistent with the design basis success criteria because the design basis success criteria is presented in terms PORV operability and not block valve operability. Therefore, address the following:

   i. Clarify what the design basis success criteria is for LCO 3.4.11 Conditions C and D in terms of the PORV block valves that are needed and whether the PRA success criteria used to model these conditions is consistent with that criteria.

   ii. If the PRA success criteria is inconsistent with the design basis success criteria, then justify the success criterion used in the PRA to model LCO 3.4.11 Conditions C and D.

b) TSTF-505 LAR Table E1-1 states for TS LCO 3.7.4 (“Two ARV lines shall be OPERABLE”) Condition A (“One ARV line inoperable”) that the design basis success criteria are “1/2 ARVs operable AND Opposite train block valve operable.” The PRA success criteria presented for this LCO condition does not address the required operability of the Atmospheric Relief Valve (ARV) block valves, though the comment column for this table entry states that the “Block valve can be represented by a surrogate.” ARV block valves function to isolate a stuck open ARV are not modelled for SGTR but are modelled to prevent both SGs from blowing down during a feedwater event.” The “Function Required” column of this entry does not explicitly state what the design basis function of the ARVs block values is for this LCO condition and whether they have more than one function (e.g., isolation of stuck open ARV and isolation of SG during feedwater event). Therefore, clarify what the design basis function of the ARVs block is for this LCO condition and what surrogate modelling would be used in the PRA to represent an inoperable ARV block valve.

c) LAR Table E1-1 indicates for TS LCO 3.7.5 (“Two motor driven AFW (MDAFW) trains, one turbine driven AFW (TDAFW) trains, and two standby AFW (SAFW) trains shall be OPERABLE”) Condition F (“Both SAFW Trains inoperable”) that the design basis success criteria are “1/2 TDAFW flow paths operable AND 1/2 MDAFW trains operable OR 2/2 TDAFW flow paths operable OR 2/2 MDAFW trains operable.” The table indicates that the PRA success criteria for this LCO condition is “1/2 Standby AFW pumps to either SG required for decay heat removal.” The PRA success criterion is presented for this LCO condition in terms of the number of SAFW pumps needed for successful decay heat removal is not the same as the design basis success criteria which is presented in terms of the number of TDAFW flow paths and MDAFW trains needed. (It might be implied from the LAR table that the PRAs model the success criteria for the TDAFW and MDAFW pumps consistent with the design basis success criteria presented for this LCO condition and in addition credits the SAFW pumps for decay heat removal based on thermal hydraulic analysis.) Therefore, address the following:

   i. Explain how the PRA success criteria used for LCO 3.7.5.F which models the criteria in term of SAFW pumps is equivalent to design basis success criteria presented for this LCO condition which presents the criteria in terms of MDAFW and TDAFWs trains.

ii. If the design basis and PRA success criteria are not equivalent, then explain the basis for the PRA success criteria and justify that LOC 3.7.5.F can be modelled in the PRAs.

APLA QUESTION 05 – Total Risk and Accounting for the SOKC [10 CFR 50.69 and TSTF-505]

RG 1.174 provides the risk acceptance guidance for total core damage frequency (CDF) (1E-04 per year) and large early release frequency (LERF) (1E-05 per year). NRC staff notes based on RG 1.174 and Section 6.4 of NUREG-1855, Revision 1, for a Capability Category II risk evaluation, the mean values of the risk metrics (total and incremental values) need to be compared against the risk acceptance guidelines. The mean values referred to are the means of the probability distributions that result from the propagation of the uncertainties on the PRA input parameters and model uncertainties explicitly reelected in the PRA models. In general, the point estimate CDF and LERF obtained by quantification of the cutset probabilities using mean values for each basic event probability does not produce a true mean of the CDF and LERF. Under certain circumstances, a formal propagation of uncertainty may not be required if it can be demonstrated that the state of knowledge (SOKC) is unimportant (i.e., the risk results are well below the acceptance guidelines).

For 10 CFR 50.69, LAR Attachment 2 presents the total internal events and internal fire baseline CDF and LERF. The Ginna TSTF-505 LAR presents more total risk information than just the total internal events and fire CDF and LERF. TSTF-505 LAR Enclosure 5, Section 2 states that the total CDF and LERF values presented in Table E5-1 for Ginna are "point estimate values" which are likely lower than the mean CDF and LERF values. The total CDF and LERF values presented in Table E5-1 include the seismic hazard contribution based on the seismic penalty values that will be used in the RICT calculations, but do not include the high wind hazard contribution. LAR Enclosure 4, Section 4 presents a high winds CDF penalty of 1E-05 per year and LERF penalty of 2E-06 per year for use in the RICT calculations for all plant configurations except those associated with LCO 3.7.5.F, 3.6.2.C, and 3.6.3.E. (For these exceptions, the CDF penalty could be as high as 7E-05 per year and the LERF penalty could be as high as 5E-06 per year.) NRC staff notes given the high wind risk contribution and the potential risk increase due to a possible PRA model update in response to information requests (e.g., requests concerning update of the fire PRA to incorporate internal event F&O resolutions) that the total risk could be higher than shown in the LARs. Therefore, the total CDF could potentially approach the RG 1.174, Revision 3 guidelines of 1E-05 per year when the total mean LERF is used accounting for the SOKC, the high winds risk contribution is included, and potential risk increases associated with model updates performed in response to NRC requests are considered. Therefore, address the following:

a) Demonstrate, that after the total mean internal events and fire CDF and LERF values are calculated to account for the SOKC, the high winds risk contribution is included, and potential risk increases associated with model updates performed in response to NRC requests are considered, the total risk for Ginna is in conformance with RG 1.174 risk acceptance guidelines (i.e., CDF < 1E-04 and LERF < 1E-05 per year). Include identification of the fire PRA parameters that are assumed to correlated in the parametric uncertainty analysis of fire events.

b) Alternatively, propose a mechanism that ensures calculation of the mean internal events and fire CDFs and LERFs to account for the SOKC, the high winds risk contribution is included, and potential risk increases associated with model updates performed in response to NRC requests are considered prior to implementation of the RICT program. The mechanism must also ensure confirmation that the updated total CDF and LERF values are still in conformance with the RG 1.174 risk acceptance guidance (i.e., CDF < 1E-04 and LERF < 1E-05 per year) prior to implementation of the RICT program or SSC categorization.

c) Discuss how the SOKC will be addressed for the RICT program and SSC categorization, and how this treatment is consistent with NUREG-1855, Revision 1 when the risk increase associated with SOKC is considered.

APLA QUESTION 06: Open Phase Condition [10 CFR 50.69 and TSTF-505]

In response to the January 30, 2012, Open Phase Condition (OPC) event at the Byron Generating Station, the NRC issued Bulletin 2012-01, "Design Vulnerability in Electric Power System" (ML12074A115). As part of the initial Voluntary Industry Initiative (VII) for mitigation of the potential for the occurrence of an OPC in electrical switchyards (ADAMS Accession No. ML22091A281) licensees have made the addition of an Open Phase Isolation System (OPIS). As per SRM-SECY-16-0068, "Interim Enforcement Policy For Open Phase Conditions In Electric Power Systems For Operating Reactors", the NRC staff was directed to ensure that licensees have appropriately implemented OPIS and that licensing bases have been updated accordingly. Inspections of OPIS by NRC staff are currently underway. From revised voluntary initiative and resulting industry guidance in NEI 19-02 on estimating OPC and OPIS risk, it is understood that the risk impact of an OPC can vary widely dependent on electrical switchyard configuration and design. In light of this observation, provide the following information:

a. A discussion of the risk impact of OPC and OPIS at Ginna Nuclear Plant

b. Discuss if the risk impact of OPC and OPIS have been or, are to be, incorporated as part of the plant Model of Record (MOR). If so, provide the following:
   i. The schedule for the inclusion of OPC and OPIS to the MOR.
   ii. The impact, if any, to key assumptions and sources of uncertainty.
   iii. A discussion of the HRA methods and assumptions used for OPIS alarm manual response.
   iv. The impact to external events, e.g., fire, seismic, flooding, high winds, tornado, other external events, etc.
   v. A discussion of the risk impact of inadvertent OPIS actuation and justification for its exclusion.

c. If OPC and OPIS are not planned to be included in the MOR, provide justification why the risk impact is not included by performing either a qualitative or sensitivity analysis.

PRA LICENSING BRANCH B (APLB) FIRE PRA QUESTIONS

APLB QUESTION 01 – Update of Fire PRA with Internal Event F&O Resolutions [10 CFR 50.69 and TSTF-505]

Regulatory Guide 1.200, Revision 2 (ML090410014) provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the ASME/ANS PRA standard ASME/ANS-RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," as one acceptable approach for determining the technical acceptability of the PRA. The primary results of peer review are the F&Os recorded by the peer review team and the subsequent resolution of these F&Os. The primary results of peer review are the F&Os recorded by the peer review team and the subsequent resolution of these F&Os. A process to close finding-level F&Os is documented in Appendix X to the Nuclear Energy Institute (NEI) guidance documents NEI 05-04, NEI 07-12, and NEI 12- 3, titled "NEI 05-04/07-12/12-06 Appendix X: Close-out of Facts and Observations (F&Os)" (ADAMS Package Accession No. ML17086A431), which was accepted by the NRC in a letter dated May 3, 2017 (ML17079A427).

LAR Enclosure 2, Section 3 states that Independent Assessments were performed in 2017 and 2020 to closeout internal events PRA F&Os after the model was updated to resolve F&Os from the 2009 full-scope peer review. LAR Enclosure 2, Section 4 states that the last full-scope peer review of the fire PRA was performed in June 2012 which is significantly before the internal events PRA F&O closure reviews in 2017 and 2020. The LAR does not indicate when the modeling updates to the internal events PRA to resolve F&Os occurred and whether applicable modeling updates were also performed for the fire PRA. Given that internal events PRA provides the modeling foundation for the fire PRA, it is not clear to NRC staff whether F&O resolutions made to the internal events PRA to close F&Os that could impact the fire PRA were incorporated into the fire PRA. Therefore, address the following:

a) Confirm that all internal events PRA modeling updates performed to resolve F&Os that could impact fire risk were incorporated into the fire PRA.

b) If it cannot be confirmed in response to part (a) above that all internal events modeling updates performed to resolve F&Os that could impact fire risk were incorporated into the fire PRA, then propose a mechanism that ensures that all internal events modeling updates performed to resolve F&Os that could impact fire risk are incorporated into the fire PRA prior to implementation of the RICT program. Alternatively, justify that all the internal events modeling updates performed to resolve F&Os have an inconsequential impact on the RICT calculations.

APLB QUESTION 02 – Fire PRA Uncertainty Associated with Methods [10 CFR 50.69 and TSTF-505]

RG 1.200 states "NRC reviewers, [will] focus their review on key assumptions and areas identified by peer reviewers as being of concern and relevant to the application." Some concerns are not always readily identified in F&Os by the peer review teams but are considered potential key assumptions by the NRC staff because using more defensible and less simplified assumptions could substantively affect the fire risk and overall risk profile of the plant. The NRC staff notes that the calculated results of the PRAs are used directly to calculate a RICT.

LAR, Enclosure 9, Section 4 indicates that the Ginna fire PRA was developed using methods presented NUREG/CR-6850 and "other more recent NUREGs, e.g., NUREG-7150… and published "frequently asked question" FAQs) for fire PRA." Moreover, LAR Attachment 2 presents the following from 5.5.18 of the Technical Specifications for the RICT program: "Methods to assess the risk from extending the Completion Times must be PRA methods used to support [this license amendment], or other methods approved by the NRC for generic use; and any change in the PRA methods to assess risk that are outside these approval boundaries require prior NRC approval." NRC staff notes that since (or about the same time as) the Ginna NFPA 805 program NRC SE (ML15271A101) was issued November 23, 2015, further NRC endorsed guidance was issued that can impact the fire PRA modelling and potentially the RICT calculations. These fire PRA studies and methods include:

- NUREG-2178, Volume 1, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) Volume 1: Peak Heat release Rates and Effect of Obstructed Plume" dated April 2016 (ML16110A140).
- NUREG-2178, Volume 2, "Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE) Volume 2: Fire Modeling Guidance for Electrical Cabinets, Electric Motors, Indoor Dry Transformers, and the Main Control Board," dated June 2020 (ML20168A655).
- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities DELORESVEWFIRE), dated December 2016 (ML16343A058).
- NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," dated January 2015 (ML15016A069).

Therefore, for each of the above NRC-accepted fire PRA methods and studies address the following:

a) Explain whether the cited fire PRA guidance has been incorporated into the Ginna fire PRA and, as applicable, summarize the changes made to the fire PRA model. Indicate whether this change was PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4 and qualified by RG 1.200, Revision 3, along with a justification for the determination. If this change constitutes a PRA upgrade, then discuss the focused-scope (or full-scope) peer review(s) that was performed to evaluate the change and provide any open F&Os and associated dispositions from this peer review(s) in accordance with RG 1.200, Revision 3.

b) If the response to part (a) above indicates that the cited fire PRA guidance has not been incorporated into the Ginna fire PRA, then justify why application of the fire PRA guidance would have an inconsequential impact on the (1) TSTF-505 RICT calculations, and (2) estimated total CDF and total LERF. As part of this justification, identify any fire PRA methodologies used in the Ginna fire PRA that are no longer accepted by the NRC staff (e.g., guidance provided in FAQ 08-0046, “Closure of National Fire Protection Association 805 Frequently Asked Question 08-0046 Incipient Fire Detection Systems,” ML093220426, has been retired by letter dated July 1, 2016, (ML16167A444). Provide technical justification for its use in the fire PRA supporting the 10 CFR 50.69 program and RICT calculations, along with evaluating the significance of its use on the risk estimates provided in Enclosure 5 of the LAR.

c) If in response to part (b) above, the cited fire PRA guidance has not been incorporated into the Ginna fire PRA and it cannot be justified that the application of the cited fire PRA guidance would have an inconsequential impact on the 10 CFR 50.69 program, TSTF-505 RICT calculations, and estimated total CDF and total LERF, then propose a mechanism that ensures the cited fire PRA guidance (or other NRC acceptable methods) will be integrated into the Ginna fire PRA prior to implementation of the RICT program and 10 CFR 50.69 program. If this fire PRA update is determined to be a PRA model upgrade per the ASME/ANS PRA standard, then include in this mechanism a process for conducting a focused-scope peer review and ensure any findings are closed by using an approved NRC process.

APLB QUESTION 03 – NFPA 805 Committed Modifications and Implementation Items [10 CFR 50.69 and TSTF-505]

RG 1.200, Revision 3 and NEI 06-09, Revision 0-A state that the PRA models which support the risk-informed program must be maintained consistent with the as-built, as-operated plant. The Ginna NFPA 805 program NRC SE dated November 23, 2015 (ML15271A101) cites commitments made in Attachment S of the Ginna NFPA 805 LAR to perform plant modifications and complete implementation items (e.g., updated fire response procedures) before fully transitioning to the NFPA 805 program. These plant improvements were used to offset the risk increase associated with transitioning to the NFPA 805 program and show that the risk acceptance guidelines in RG 1.174, Revision 3 are met. It is not clear to NRC staff whether the promised NFPA 805 plant modifications and implementation items have been completed and whether credited but uncompleted improvements can impact the RICT program. In light of these observations, address the following:

a) Confirm that the fire PRA model used to support the RICT program and 10 CFR 50.69 reflects the as-built, as-operated plant (e.g., do not credit NFPA 805 plant modifications or implementation items that are not yet complete).

b) If in response to part (a), it cannot be confirmed that the fire PRA models used to support the RICT program and 10 CFR 50.69 reflect the as-built, as-operated plant, then justify that the modeling credit for NFPA 805 plant modifications and/or implementation items not yet completed but credited in the fire PRA do not have a consequential impact on the RICT calculations or 10 CFR 50.69.

c) As an alternative to part (b) above, propose a mechanism that ensures that fire PRA models used to support the RICT calculations or 10 CFR 50.69 reflect the as-built, as-operated plant (e.g., do not credit NFPA 805 plant modifications or implementation items that are not complete) prior to implementation of the RICT program or 10 CFR 50.69.

PRA LICENSING BRANCH C (APLC) EXTERNAL HAZARDS QUESTIONS

APLC QUESTION 01 – Alternate Seismic Approach **[10 CFR 50.69]**

Paragraph (b)(2)(ii) of 10 CFR 50.69 requires, for license amendment, a description of measures taken to assure the level of detail of the systematic processes that evaluate the plant. This includes the internal events at power PRA required by 10 CFR 50.69(c)(1)(i) as well as the risk analyses used to address external events.

The LAR proposes to use the alternative seismic approach for Tier 1 plants based on insights from Electric Power Research Institute (EPRI) 3002017583 and other qualitative considerations. The NRC staff understands that EPRI 3002017583 is an updated version of EPRI 3002012988 that was reviewed in conjunction with its review of the Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2, LAR for adoption of 10 CFR 50.69 (precedent) dated November 28, 2018 (ML18333A022). The staff has not endorsed EPRI 3002012988 as a topical report for generic use. As such, each licensee needs to perform a plant-specific review for applicability of the Tier 1 alternative seismic approach. The NRC staff reviewed and approved CCNPP's alternative seismic approach, which was based on information for Tier 1 plants included in the EPRI report and information provided in the supplements to the CCNPP LAR, as described in Safety Evaluation dated February 28, 2020 (ML19330D909). Accordingly, address the following:

a) Identify differences (if any) that may exist between the proposed alternative seismic approach and the NRC staff approval of the precedent documented in the CCNPP safety evaluation, including any Ginna specific considerations.

b) If there are differences identified in response to part (a) above, then justify that in light of these differences that the proposed approach (1) meets the criteria for using the alternative approach, (2) will appropriately determine the safety significance of any SSCs whose seismic-induced failures led directly to core damage and large early release in conjunction with other elements of the 10 CFR 50.69 risk-categorization approach, and (3) the seismic risk contribution would not solely result in any additional SSC being categorized.

<u>APLC Question 02 – Risk contribution of a seismic event [10 CFR 50.69]</u>

In Title 10 of the *Code of Federal Regulation* (CFR) 50.69(b)(2)(ii) requires a description of the measures taken to assure that the quality and level of detail of the systematic processes that evaluate the plant for internal and external events during normal operation, low power, and shutdown (including the plant-specific probabilistic risk assessment (PRA), margins-type approaches, or other systematic evaluation techniques used to evaluate severe accident vulnerabilities) are adequate for the categorization of SSCs.

Section 3.2.3, “Seismic Hazards,” of the enclosure to the LAR states that “low seismic CDF and LERF estimates lead to reasonable confidence that seismic risk contributions would allow reducing an HSS to LSS…” Section 2.2.2 of the EPRI report identifies the contribution of seismic to total plant risk as a basis for the use of the proposed alternate seismic approach. Further, insights in the EPRI report are derived from the full spectrum of the seismic hazard (i.e., the entire hazard curve). The LAR does not provide information to support the claim that the plant-specific seismic risk is a small contribution to the total plant risk and thereby, the applicability of the proposed alternate seismic approach to the licensee. Based on Enclosure 5 of the Ginna TSTF-505 LAR, seismic CDF is about 7% of total CDF and seismic LERF is about 68% of total LERF. It appears that SLERF is not a small contribution to the total plant risk.

Justify that the plant specific seismic LERF risk is low relative to the overall plant LERF risk such that the categorization results will not be significantly impacted to support the applicability of the proposed alternate seismic approach.

<u>APLC QUESTION 03 – Overall Use of NEI 00-04 Figure 5-6 and Considerations of Extreme Wind or Tornadoes and Ice Cover Hazard [10 CFR 50.69]</u>

NEI 00-04, Revision 0, Figure 5-6 provides guidance to be used to determine SSC safety. The same document, states, in part, that if it can be shown that the component either did not participate in any screened scenarios or, even if credit for the component was removed, the screened scenario would not become unscreened, then it is considered a candidate for the LSS category.

LAR Section 3.2.4 states that “[a]ll external hazards, except for seismic, were screened for applicability to Ginna per a plant-specific evaluation.” LAR Attachment 4 lists all hazards as screened except internal events, internal flooding, internal fire, and seismic events for which there are PRA models or in the case of the seismic hazard an alternate approach is used. Except for the external flooding hazard entry in the Attachment 4 table of the LAR, the guidance in NEI 00-04, Figure 5-6 regarding SSCs that play a role in screening a hazard is not discussed in the LAR. Therefore, it appears to NRC staff that at the time an SSC is categorized it may not be evaluated using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard (except the external flooding hazard) because that evaluation has already been made. The NRC staff notes that plant changes, operational experience, and identified errors or limitations in the PRA models could potentially impact the conclusion that an SSC is not needed to screen an external hazard. Therefore, address the following:

a) Clarify whether an SSC will be evaluated during categorization of the SSC using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard.

b) If SSCs will not be evaluated using the guidance in NEI 00-04, Figure 5-6 to ensure that the SSC are not credited in screening an external hazard at the time of categorization because that evaluation has already been made, then explain how plant changes, plant or industry operational experience, and identified errors or limitations that could change that decision are addressed.

With regards to the extreme wind or tornado hazard, the LAR appears to indicate that screening of this hazard was determined in part on the success of tornado missile barriers “after upgrades to several of the barriers are made.” Attachment 7 of the LAR indicates that several identified upgrades and modifications are needed to protect against 3-inch pipe missiles generated by tornadoes. The LAR refers to these identified upgrades and modifications as “commitments,” but it is not clear to the NRC staff what mechanism ensures that these commitments will be completed prior to implementation of the 10 CFR 50.69 program. It is also not clear whether SSCs, including those associated with the cited upgrades, will be evaluated during risk-informed categorization using the guidance in NEI 00-04, Figure 5-6 to confirm that the SSC is not credited in screening an external hazard. Therefore, address the following:

c) Clarify whether the tornado missile plant upgrades and modifications discussed in the LAR to protect against 3-inch pipe missiles generated by tornadoes are needed to support the screening of the extreme wind or tornado hazard. If they are needed for screening, then propose a mechanism to ensure that the cited upgrades and modifications will be completed prior to implementation of the 10 CFR 50.69 program.

d) Confirm that SSCs will be evaluated using the guidance in NEI 00-04, Figure 5-6 to determine whether SSCs are credited in screening the extreme wind or tornado hazard during 10 CFR 50.69 risk-informed categorization.

With regards to the ice cover hazard, Attachment 4 of the LAR indicates the ice cover hazard (i.e., accumulation of frozen water on bodies of water such as lakes, rivers or on structures, systems, and components) is screened based on the criteria defined in Attachment 5 as “C1” (Event damage potential is < events for which the plant was designed) and “C4” (Event is included in the definition of another event). However, Section 5.0 of the SE for the Ginna Individual Plant Examination of External Events (IPEEE) (ML003773799) states that: “The licensee reported that in an earlier plant modification, the power for the heaters on the cooling water intake screens on Lake Ontario had been increased to protect against ice formation (slush).” Therefore, it appears there is a potential for ice to form on the cooling water intake screens in the winter that could potentially fail the cooling water supply for such systems as the Ultimate Heat Sink particularly if the heaters (or power to the heaters) are unavailable. It appears that there might be SSCs (e.g., the heaters) credited in screening this hazard.

e) If any SSCs are credited in screening the ice cover hazard, then confirm that SSCs will be evaluated using the guidance in NEI 00-04, Figure 5-6 to ensure that the SSCs credited in screening the ice cover hazard at the time of 10 CFR 50.69 risk-informed categorization are identified including heaters and power supplies.

APLC QUESTION 04 – Evaluation of Seismic Induced Loss of Offsite Power [TSTF-505]

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A (ML12286A322), states that the “impact of other external events risk shall be addressed in the [Risk Managed Technical Specifications] RMTS program,” and explains that one method to do this is by “performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated [Risk-Informed Completion Time] RICT." The NRC staff's safety evaluation for NEI 06-09 (ML071200238) states that "Where [probabilistic risk assessment] PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

Section 4 of Enclosure 4 to the LAR does not address the incremental risk associated with seismic-induced loss of offsite power (LOOP) that may occur following the design basis seismic event. The accident scenarios associate with seismically-induced (and therefore unrecoverable) LOOP frequency could already be addressed to some extent in the internal events PRA for unrecovered LOOP events, but this is not explained either.

Demonstrate that seismic-induced LOOP will have an inconsequential impact on the RICT calculations.

APLC QUESTION 05 – High Winds Penalty Factors [TSTF-505]

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A (ML12286A322), states that the "impact of other external events risk shall be addressed in the [Risk Managed Technical Specifications] RMTS program," and explains that one method to do this is by "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated [Risk-Informed Completion Time] RICT." The NRC staff's safety evaluation for NEI 06-09 (ML071200238) states that "Where [probabilistic risk assessment] PRA models are not available, conservative or bounding analyses may be performed to quantify the risk impact and support the calculation of the RICT."

In Section 4 of Enclosure 4 to the LAR, the licensee provides its extreme winds analysis, and stated that the high wind hazards can be screened from consideration for the TSTF-505 application, except tornado missiles for certain maintenance configurations. However, the licensee does not provide the basis for these high wind penalty factors. The LAR states that the CDF penalty is 1E-05 per year and the LERF penalty is 2E-06 per year for all plant configurations associated with LCO conditions encompassed in the RICT program with the exceptions of LCO conditions 3.7.5.F, 3.6.2.C, and 3.6.3.E. For these exceptions, other penalty factors are presented but no explanation of how these additional penalty factors were derived is provided in the LAR.

Enclosure 2 of the LAR provides a completion time of 7 days for the original LCO 3.7.5.F. Table E1-2 of Enclosure 1 to the LAR shows that a completion time of RICT estimate is 1.4 days for LCO 3.7.5.F, which is much lower than its original completion time.

In light of above observations, address the following:

a) Discuss the calculational basis for each of the extreme wind and tornado CDF and LERF penalty factors presented in the LAR for use in the RICT calculations.

b) Justify that each penalty factor proposed represents a reasonable bounding value per the guidance in NEI 06-09.

c) Discuss the completion times for LCO 3.7.5.F, in which the estimated RICT is lower than its original CT. Is the lower estimated RICT caused by the conservative penalty factor of delta CDF (7E-5 /yr) for LCO 3.7.5.F?

Answers to these questions will support the staff's review of the licensee's RICT program implementation and technical basis for seismic penalty factors.

APLC QUESTION 06 – External Flooding [TSTF-505]

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A, states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by documenting prior to the RMTS program that external events that are not modeled in the PRA are not significant contributors to configuration risk. The SE for NEI 06-09 states that "[o]ther external events are also treated quantitatively, unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk."

LAR Enclosure 4, Section 5 concluded that Ginna's external flooding hazard is screened out because the maximum event (i.e., a LIP (Local Intense Precipitation) combined with a River Flood) that could impact the plant has an exceedance frequency of < 1E-06 per year. However, NRC staff notes that, according to Section 6.1 of the Focused Evaluation report (ML17069A004), the site protection against flooding events depends on the "combination of permanent and temporary passive flood protection barriers to prevent ingress of flood waters in areas with key SSCs." The report refers to temporary portable flood barriers at the Auxiliary Building and Standby Auxiliary Feedwater Pump Building Annex and installed water-resistant doors at the Battery and Diesel Generator Rooms. The LAR did not describe any RMAs to ensure that the flood protection features, which are integral to flood protection and important for screening of external flooding, continue to be available and functional during the proposed RICTs.

Identify and justify the mechanism that will be used to ensure that the temporary portable flood barriers will be in installed and the water-resistant doors will be closed during a flood event to prevent impact on risk significant equipment.

APLC QUESTION 07 – Ice Cover Hazard [TSTF-505]

Section 2.3.1, Item 7, of NEI 06-09, Revision 0-A, states that the "impact of other external events risk shall be addressed in the RMTS program," and explains that one method to do this is by documenting prior to the RMTS program that external events that are not modeled in the PRA are not significant contributors to configuration risk. The SE for NEI 06-09 states that "[o]ther external events are also treated quantitatively, unless it is demonstrated that these risk sources are insignificant contributors to configuration-specific risk." LAR Enclosure 4, Table E4-4 indicates the ice cover hazard (i.e., The accumulation of frozen water on bodies of water such as lakes and rivers or on structures, systems, and components) is screened based on the Table E4-5 criteria of "C1" (Event damage potential is < events for which the plant was designed) and "C4" (Event is included in the definition of another event). Section 5.0 of the SE for the Ginna Individual Plant Examination of External Events (IPEEE) states that: "The licensee reported that in an earlier plant modification, the power for the heaters on the cooling water intake screens on Lake Ontario had been increased to protect against ice formation (slush)." Therefore, it appears there is a potential for ice to form on the cooling water intake screens in the winter that could potentially fail the cooling water supply for such systems as the

Ultimate Heat Sink, particularly if the heaters (or power to the heaters) are unavailable.

It is not clear to the NRC staff how the criteria cited above are used to screen this hazard event for all plant configurations encompassed in the RICT program. Section 6 of LAR Enclosure 4 states for configurations allowed by the RICT program that “hazards for which the ability to achieve safe shutdown may be impacted by one or more plant configurations, the impact of the hazard to particular SSCs is assessed and a basis for the screening decision applicable to configurations impacting those SSCs is provided.” Accordingly, given that ice cover could impact the ability to achieve safe shutdown, especially for certain configurations, address the following:

a) Explain how the “C1” and “C4” screening are used to screen the ice cover hazard from consideration for impact on RICT calculations given that ice cover events appear to be anticipated and could be a contributor to a core damage accident particularly for certain plant configurations. Include discussion of the heaters used to keep the ice clear from the cooling water intake screens.

b) If the screening criteria cited in the LAR are not sufficient to screen the ice cover hazard from consideration for impact on the RICTs for all plant configurations encompassed in the RICT program, then justify screening the ice cover hazard using another basis.

c) If it cannot be justified that the Ice Cover hazard can be screened for impact on the RICT calculations, then explain how the RICT program will mitigate or prevent the impact of the ice cover hazard during a RICT application.

INSTRUCTIONS: INSTRUMENTATION AND CONTROLS (EICB) AND TECH. SPEC. (STSB) BRANCHES'

EICB Question 01: 7-1 Page 15 of 359 of the LAR (ML21140A324) includes "INSERT RICT NOTE 1" however, this insert does not appear to be used in the LAR. Please explain.

EICB Question 02: 7-2 LAR Enclosure 1 Table E1-1 seems to depict several loss of TS required functions, please clarify whether these items are or are not a loss of function.

Tech Spec (TS)	Function	Design Success Criteria	Explanation
3.3.1.K	(10b) RCP breaker position (Two Loops)	One open breaker per RCP	TS Table 3.3.1-1 identifies one channel per RCP. If this channel is lost, then the associated TS function is lost. Since Condition K applies to other functions, should the proposed TS change be in to "Insert RICT Note 2"?
3.3.1.N	(10a) RCP breaker position (Single Loop)	One open breaker per RCP	TS Table 3.3.1-1 identifies one channel per RCP. If this channel is lost, then the associated TS function is lost. Should the proposed TS change be in to "Insert RICT Note 2"?
3.3.1.P	(14) (b) Turbine Stop Valve Closure	Two of two channels	A design success criteria of "two of two" seems like a loss of TS function if one is out. Since Condition P applies to other functions, should the proposed TS change be in to "Insert RICT Note 2"?
3.3.1.U	(18) Reactor Trip Breaker Undervoltage and Shunt Trip Mechanisms	One trip mechanism per RTB	Needs some discussion to understand.
3.3.2.B	(6f) Auxiliary Feedwater-Trip of Both Main Feedwater Pumps	Two of two channels per MFW Pump	Need explanation of logical arrangement (e.g., voting) of channels.
3.3.2.H	Containment Spray; (2a) Manual Initiation	(2a) Two of two pushbuttons	A design success criteria of "two of two" seems like a loss of TS function if one is out. Since Condition H applies to other functions, should the proposed TS change be in to

Tech Spec (TS)	Function	Design Success Criteria	Explanation
			“Insert RICT Note 2”? The TS Bases state: “The operator can initiate CS at any time from the control room by simultaneously depressing two CS actuation pushbuttons…the inoperability of either pushbutton fails both trains of manual initiation.”
3.3.5.A	Containment Radiation Signal from either of 2 channels: Gaseous: one of one channel Particulate: one of one channel		A design success criteria of “one of one” seems like a loss of TS function if one is out. Should the proposed TS change be in to “Insert RICT Note 2”? The TS Bases state: “Two containment radiation monitoring channels are provided as input to the containment ventilation isolation. The two radiation detectors are of different types: gaseous (R-12), and particulate (R-11). Both detectors will respond to most events that release radiation to containment. However, analyses have not been conducted to demonstrate that all credible events will be detected by more than one monitor. Therefore, for the purposes of this LCO the two channels are not considered redundant. Instead, they are treated as two one-out-of-one Functions.”

EICB Question 03: 7-3 LAR Attachment 1, page 3 pf 6 states:

> “TS 3.3.5.A.1- One rad monitor inoperable. Per UFSAR Section 6.2.4.3, there is no loss of function if R-11 or R-12 become inoperable. These radiation monitors actuate Containment Ventilation Isolation (CVI), for the mini-purge valves. CVI serves as a backup to the Containment Isolation (CI) signal, and is not specifically credited in the accident analysis.”

This is a TS loss of function. If the function is in the TS, it does not matter whether the function is credited in the accident analysis. Generally. if a function is not credited in the accident analysis, then the loss of the function is most likely a reduction in defense-in-depth.

SNSB BRANCH:

SNSB-1: TS 3.4.11.B and TS 3.4.11.C – One PORV inoperable and One Block Valve Inoperable

Ginna Technical Specification Bases 3.4.11 indicates that the pressurizer PORV's and block valves have many safety functions, including (1) providing flow path for depressurization control during a SGTR event, and (2) terminating a small break LOCA (SBLOCA) in the event a pressurizer PORV fails to reclose following actuation.

Discuss whether the following two plant configuration cases would be allowed for the RICT application or not. If they not allowed, specify the references that disallow these cases. If they are allowed, discuss if the cases would result in a loss of the associated function (LOF) discussed above. If the LOF would occur, discuss, and justify the eligibility of Case 1 and Case 2 for applying the RICT program in accordance with the guidelines of the NRC-approved NEI-06-09. Specifically, Condition 3 in the SE approving NEI-06-09 (ADAMS No. ML12286A322) imposes a restriction that when an LOF of specific safety function for the affected TS system occurs, the RMTS cannot be applied.

Response to NRC Audit Questions Attachment 1 License Amendment Request to Adopt Page 2 of 6 Risk Informed Completion Times TSTF-505 Docket No. 50-244
a. Case 1: one PORV is inoperable and closed (allowed by TS 3.4.11.B) and one blocked valve on the other PORV line is inoperable and closed (allowed by TS 3.4.11.C), and
b. Case 2: one PORV is inoperable and open (allowed by TS 3.4.11.B) and one PORV associated blocked valve is inoperable and open (allowed by TS 3.4.11.C).

Additional Comment
Page TS B 3.4.11-3 states that "The PORV is required to be OPERABLE to mitigate the effects associated with an SGTR and its block valve must be OPERABLE to limit the potential for a small break loss-of-coolant accident through the flow path".

Justify the use of the block valve to control pressurizer pressure for an SGTR to show that the Area of Responsibility, assuming the PORV for pressurizer pressure control, in Ginna UFSAR Section 15.6.3 remains valid. The requested information should show that the relief capacity of the block valve is equivalent to the PORV, adequate steps of using the block valve for pressurizer pressure control during an SGTR are available in the emergency response procedures or similar procedures, and the operator training data show the required manual actions using the block valve can be completed within the time frames assumed in the SGTR for pressure control using the PORV specified in the following UFSAR Tables:

1. Table 15.6-3: MARGIN TO OVERFILL ANALYSIS (Page 211 of 276)
· PORV opened at 47.4 minutes from the SGTR initiation and
· PORV closed at 47.9 minutes.
2. Table 15.6-5: OFFSITE RADIATION DOSE ANALYSIS (Page 213 of 276)
· PORV opened at 73.2 minutes and
· PORV closed at 74.1 minutes.

The licensee stated that the modified TS 3.4.11 is to ensure that there will always be one OPERABLE PORV train, not relying on a block valve for pressure control, in order to implement RICT.

As shown in the modified TS changes above, the operator has an option to perform current RA B.3 and RA C.2, or perform the added RAs (in red). Is there any assurance that the operator will perform the added RAs to meet the intent such that there will always be one OPERABLE PORV train? Should “stats” be “status”?

SNSB-2: TS 3.4.11.C.2 and TS 3.4.11.D.2 – Two PORV BLOCK Valves Inoperable
The PORV block valves are used to terminate an SBLOCA in the event a pressurizer PORV fails to reclose following actuation. Page 4 of Attachment 1 to the LAR states that for TS 3.4.11.C.2 and TS 3.4.11.D.2 allowing two block valves inoperable the current completion time to terminate the loss of function (LOF) is 72 hours.

Discuss whether the plant configuration for TS 3.4.11.C.2 and TS 3.4.11.D.2 would result in an LOF or not. If an LOF would not occur, discuss, and justify the determination. If an LOF would occur, discuss, and justify the eligibility of TS 3.4.11.C.2 and TS 3.4.11.D.2 for applying the RICT program in accordance with the guidelines of NRC-approved NEI-06-09. Specifically, Condition 3 in the SE approving NEI-06-09 imposes a restriction that when an LOF of specific safety function for the affected TS system occurs, the RMTS cannot be applied.

Additional Comment
Initial SNSB-2 response did not include information addressing the applicability of the RMTS to the following plant configurations which were discussed in the response to audit questions:
a. One (TS 3.4.11.C) or Both Block Valves Failed Open (and TS 3.4.11.D)
b. One PORV Failed Open (TS 3.4.11.B) and Two Block Valves Failed Open (TS 3.4.11.D)
c. One PORV Failed Closed (TS 3.4.11.B) and Two Block Valves Inoperable (with the One Associated with the Inoperable PORV Failed Open and the Second One on the Other Line Failed Closed (TS 3.4.11.D))

SNSB-3: TS 3.5.1.A.1 and TS 3.5.1.B.1 – One accumulator Inoperable
Accumulators are part of the emergency core cooling system and provide injection of water for core cooling during an LOCA to meet the acceptance criteria of 10 CFR 5.46. Page 4 of Attachment 1 to the LAR states for a large break cold leg LOCA, one accumulator is assumed to spill out the break, while the other provides the required core cooling. Therefore, TS 3.5.1.A.1 and TS 3.5.1.B.1 allowing one inoperable accumulator constitutes an LOF for the specified LOCA.

Since TS 3.5.1.A.1 and TS 3.5.1.B.1 would result in an LOF, discuss, and justify the eligibility of TS 3.5.1.A.1 and TS 3.5.1.B.1 for applying the RICT program in accordance with the guidelines of NRC-approved NEI-06-09. Specifically, Condition 3 in the SE approving NEI-06-09 imposes a restriction that when an LOF of specific safety function for the affected TS system occurs, the RMTS cannot be applied.

SNPB-4: TS 3.7.1.A – One or more Main Steam Safety Valve MSSVs inoperable
Demonstrate that the limiting design basis UFSAR Ch. 15 transients have been analyzed, and shown that main steam pressure can be maintained below 110% of design pressure, assuming inoperability of MSSV(s).

Response to NRC Audit Questions Attachment 1 License Amendment Request to Adopt Page 6 of 6 Risk Informed Completion Times TSTF-505 Docket No. 50-244.

SNSB-5: TS 3.7.4 – Atmospheric Relief Valves (ARVs)
The design basis safety function for the ARVs is cooling the unit to the residual heat removal (RHR) entry conditions for various evets such as the steam generator tube rupture (SGTR). Ginna Final Safety Analysis Report (FSAR), Section 15.6.3 discussed the analysis of the SGTR showing that a loss of function (LOF) of the ARVs would not occur. Page 179 of FSAR Chapter 15 stated in particular for the dose release limiting case that "Following termination of tube flow, the intact steam generator's ARV is assumed to cool down the plant at the maximum allowable rate of 100 degrees F/hour to a residual heat removal system in-service temperature of 330 degrees F."

The licensee is proposed to apply the risk-informed competition time (RICT) program to TS 3.7.4 Condition A that allows one required ARV line inoperable. Condition A would result in only one required ARV line operable. If an SGTR occurs in the SG with the remaining operable ARV line, the unaffected SGs cannot cooldown the plant because it has no operable ARV and would not be bounded by the limiting case in FSAR, Section 15.6.3.

The NRC-approved NEI-06-09 Rev. 0-A provides guidelines for application of the RICT program. Specifically, Condition 3 in the NRC safety evaluation (Page 10 of 93 in ML12286A322) approving NEI-06-09 imposes a applicability restriction that when an LOF of specific safety function for the affected TS system occurs, the RMTS cannot be applied. Provide further analysis to demonstrate how the LCO 3.7.4 ARV safety function will be maintained during a SGTR event with the ARV on the unaffected steam generator inoperable.

List of Audited Documents

The U.S. Nuclear Regulatory Commission (NRC) staff audited, in a sampling manner, the following licensee documents during its review of the license amendment request. These documents were provided on the electronic reading room with restricted access for the staff review. The list does not include documents that are in NRC's Agencywide Documents Access and Management System (ADAMS).

No. #	LAR	Name of Document
1	TSTF-505	100kw-DG-HRA.doc
2	TSTF-505	100kw-GG-HRA.pdf
3	50.69	5069-APLA-02c-Response-RevA.pdf
4	50.69	5069-APLA-02c-to-03g-Response-RevC.pdf
5	TSTF-505	APLA-1ab-Response-Rev1.pdf
6	TSTF-505	APLA-1b-Response-Rev2.pdf
7	TSTF-505	APLA-3ab-Response-Rev0 – final.pdf
8	TSTF-505	APLA-4abc-Response-Rev0 final.pdf
9	TSTF-505	Follow-up TCA and TSA questions.mht
10	TSTF-505	G1-MISC-035 r0-Signed.pdf
11	50.69	Ginna 50.69-RICT Audit-APLA-2a SSIE MTTR-Response.pdf
12	50.69	Ginna 50.69-RICT Audit-APLA-5 SOKC-final.pdf
13	50.69	Ginna 50.69-RICT Audit-APLA-6 Open Phase – Response
14	TSTF-505	OP-AA-102 106 REV 007
15	TSTF-505	Portable-Alt-RCS-Injection-Pump (002) .pdf
16	TSTF-505	Portable-Alt-RCS-Injection-Pump (002) .pdf
17	TSTF-505	Portable-Alt-RCS-Injection-pump.doc
	Audit Plan -Item 1	• 1BT132299.006.058.100.pdf • G1-MISC-023 r0.pdf • LTR-RAM-II-09049.pdf • LTR-RAM-II-12-066-Ginna Peer Review.pdf

No. #	LAR	Name of Document
18	Audit Plan-Item 2	• G1-FQ-F001-Quantification-Rev-4-Final-all-Sigs.pdf • G1-MISC-026 Ginna 50.69 and 505 Uncertainties Rev 0 – Final.pdf • G1-UNC-F001-REG Uncertainty-and-Sensitivitiy-(UNC)-Notebook-r3.pdf
19	Audit Plan-Item 3	G1-MISC-026 Gina 50.69 and 505 Uncertainties Rev – 0 Final.pdf
20	Audit Plan-Item 4	No documents
21	Audit Plan-Item 5	G1-FQ-F001-Quantification-Rev-4-Final-all-Sigs.pdf
22	Audit Plan-Item 6	No documents
23	Audit Plan-Item 7	• DA-CE-17-001 combined with Att AC r1.pdf • DA-ME-21-001_RICT TORNADO MISSILE_REV_0_FINAL_3-29-2-21.pdf • FHR-FLOOD-FREQ-signed.pdf • G1-MISC-021 External Hazard_SIGNED_Revision 0.pdf • Ginna Tornado Missile Vulnerability Report Final.pdf • Ginna _Soil_Seismic Hazard and Screening Report [11-27-2013].pdf • IPEEENUREG1407.pdf • RG016230-high-Winds-and-Transportation.pdf • RSM-112013-037 Ginna Seismic Hazard and Screening Report [11-27-2013].pdf
24	Audit Plan-Item 8	• ER-AA-569-1001-R004.pdf • ER-AA-569-1002-R003.pdf • ER-AA-569-1003-F-01-R006.pdf • ER-AA-569-1003-F-R006.pdf • ER-AA-569-1004-R006.pdf • ER-AA-569-1005-R004.pdf • ER-AA-569-1006-R003.pdf • ER-AA-569-1007-R005.pdf • ER-AA-569-R004.pdf
25	Audit Plan-Item 9	• A-601.16, Rev-009, PCR-20-03882, JKZ.pdf • OP-AA-108-117 REV 006.pdf • OP-AA-201-012-1001 REV 004.pdf • WC-AA-101-1006 Rev 4 On-Line Risk Management and Assessment.pdf
26	Audit Plan-Item 10	• ER-AA-600-1015-R020.pdf • ER-AA-600-1061-R007.pdf
27	Audit Plan-Item 11	Drawings • 03201-0101, Rev 026, 120V AC INSTRUMENT BUS ONE-LINE DIAGRAM • 03202-0102, Rev 027, 125V POWER DISTRIBUTION SYSTEM

No. #	LAR	Name of Document
		• 33013-0623.1, Rev 030, MAIN ONE LINE OPERATING DIAGRAM • 33013-0623.2, Rev 014, MAIN ONE LINE OPERATING DIAGRAM • 33013-1231.2.Rev 000.pdf, TURBINE DRIVEN AUXILLIARY FEEDWATER STEAM SIDE • 33013-1233.Rev 038.pdf, CONDENSATE LOW PRESSURE FW HEATERS (CDST) • 33013-1234, Rev 047.pdf, CONDENSATE STORAGE (CDST) • 33013-1236.2.Rev 024.pdf, FEEDWATER (FW) • 33013-1236.3.Rev 003.pdf, FEEDWATER (FW) • 33013-1237 Rev 074.pdf, AUXILIARY FEEDWATER (FW) • 33013-1242.pdf. Rev 005, FIRE PROTECTION RELAY AND MULTIPLEXOR ROOMS FP P • 33013-1245.pdf, Rev 036, AUXILIARY COOLANT COMPONENT COOLING WATER (AC) • 33013-1246.1.pdf, Rev 017, AUXILIARY COOLANT COMPONENT COOLING WATER (AC) • 33013-1246.2.pdf, Rev 015, AUXILIARY COOLANT COMPONENT COOLING WATER (AC) P • 33013-1247.pdf, Rev 049.01, AUXILIARY COOLANT RESIDUAL HEAT REMOVAL (AC) • 33013-1248.pdf, Rev 050, STATION SERICE COOLING SPENT FUEL POOL COOLING (AC) • 33013-1250.1.pdf Rev 070 STATION SERVICE COOLING WATER SAFETY RELATED (SW) • 33013-1250.2.pdf Rev 054 STATION SERVICE COOLING WATER SAFETY RELATED (SW) • 33013-1251.2.pdf Rev 034 STATION SERVICE COOLING WATER NON-SAFETY RELATED (SW) • 33013-1252.pdf Rev 032 CONDENSATE • 33013-1256.pdf Rev 024 TECHNICAL SUPPORT CENTER HVAC • 33013-1258.pdf Rev 026, REACTOR COOLANT PRESSURIZER (RC) • 33013-1259.pdf Rev 017, MISCELLANEOUS LIQUID WASTE LIQUID (WD) • 33013-1260.pdf Rev 027, REACTOR COOLANT (RC) • 33013-1261.pdf Rev 048 CONTAINMENT SPRAY (SI) • 33013-1262.1.pdf Rev 038 SAFETY INJECTION

No. #	LAR	Name of Document
		AND ACCUMULATORS (SI) • 33013-1263.pdf Rev 010, RCS OVERPRESSURE PROTECTION NITROGEN ACCUMULATORY SYSTEM • 33013-1265.1.pdf Rev 012, CHEMICAL AND VOLUME CONTROL SYSTEM CHARGING (CVCS) • 33013-1265.2.pdf Rev 027, AUXILIARY BUILDING CHEMICAL VOLUME CONTROL SYSTEM CHARGING (CVCS) • 33013-2539.pdf Rev 033 main0line0one-line-electrical • Ginna Drawing index.pdf
28	Audit Plan-Item 12	• 3013-2539-Rev33-main-line-one-line-electrical.pdf • P-11.pdf, Rev 012 DC Elec Dist • P-12, Rev 29, AC Elec Dist
29	Audit Plan-Item 13	• ATT-18.1.pdf Rev-003, PCR-19-01612 • E-0, PCR-18-05340, Rev-049, JKZ • ECA-0.0.odf Rev-044, PCF-18-01589 • ER-DG.1.pdf, Rev-020, PCR-20-09235, JKZ
30	Audit Plan-Item 14	G1-PRA-003.pdf R4 SC final
31	Additional	• APLA-APLB-LAR-Text-from0G1 LAR 008.pdf • EEEB Q3.pdf • EEB RAI Responses.pdf • EICB 02 Response rev 2.pdf • Ginna-SNSB Audit Questions for TSTF-505 adoption.pdf • TSTF-505-50-69-Supplemental-LAR-APLA-APLB-REV01-for-NRC.pdf
32	50.69	5069-APLB-01-Response-RevB.pdf
33	50.69	5069-APLB-02-Response-RevB.pdf
34	50.69	5069-APLB-03-Response-RevB.pdf
35	50.69 and TSTF-505	APLC-01.pdf
36	50.69 and TSTF-505	APLC-02 Ginna Audit Response Seis Risk Contrib_Final.pdf
37	50.69 and TSTF-505	APLC-03 Rev1.pdf
38	50.69 and TSTF-505	APLC-04 Ginna Audit Response_Seis-LOOP_Final.pdf
39	50.69 and TSTF-505	APLC-05 090921.pdf
40	50.69 and TSTF-505	APLC-06-Response-Rev.0.pdf
41	50.69 and TSTF-505	APLC-07 Rev.1.pdf
42	50.69 and TSTF-505	APLC-07.pdf
43	TSTF-505	EICB 02 Responses.pdf
44	TSTF-505	EICB 03 Response.pdf

No. #	LAR	Name of Document
45	TSTF-505	EICB-01-Response.pdf
46	TSTF-505	3-28 Comment and Response.pdf
47	TSTF-505	3-30 Comments and Responses.pdf
48	TSTF-505	33013-1353.3.pfd Rev 11
49	TSTF-505	Discussion and Response to RAI-SNSB-4.pdf TS 3.7.1.A
50	TSTF-505	Ginna (DRAFT RAI RESPONSE comments -4) Adoption of TSTF.pdf
51	TSTF-505	Ginna (DRAFT RAI RESPONSE comments -4) Adoption of TSTF.pdf (Word file)
52	TSTF-505	Ginna 505 Supplement 2 Attachment 1
53	TSTF-505	Ginna – TSTF505 Comments dated 3-28-2022 on RAI Response dated 3-7-2022
54	TSTF-505	M-51.11.pdf
55	TSTF-505	Responses to 3-7.pdf
56	TSTF-505	T-18B

SUBJECT: R. E. GINNA NUCLEAR POWER PLANT – SUMMARY OF COMBINED REGULATORY AUDIT IN SUPPORT OF RISK-INFORMED COMPLETION TIMES IN TECHNICAL SPECIFICATIONS LICENSE AMENDMENT REQUEST AND TO ADOPT 10 CFR 50.69, “RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS” (EPIDS L-2021-LLA-0091 AND L-2021-LLA-0092) DATED MAY 27, 2022

DISTRIBUTION:
PUBLIC
PM File Copy
RidsNrrDorlLpl1 Resource
RidsRgn1MailCenter Resource
RidsACRS_MailCTR Resource
RidsNrrPMGinna Resource
RidsNrrLAKZeleznock Resource
RidsNrrDexEicb Resource
RidsNrrDssStsb Resource
RidsNrrDssSnsb Resource
RidsNrrDraApla Resource
AMcCraw, OEDO

ADAMS Accession No. ML22124A316

OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LAiT	NRR/DRA/APLA/BC
NAME	VSreenivas	KZeleznock	RPascarelli
DATE	05/03/2022	05/06/2022	04/01/2022
OFFICE	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM	
NAME	JDanna	VSreenivas	
DATE	05/26/2022	05/27/2022	

OFFICIAL RECORD COPY