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

August 25, 2021

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

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – AUDIT PLAN IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST REGARDING TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B" AND 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2021-LLA-0091 AND EPID L-2021-LLA-0092)

Dear Mr. Rhoades:

By letters dated May 20, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21140A324 and ML21141A009), Exelon Generation Company, LLC (Exelon) submitted license amendment requests (LARs) for R.E. Ginna Nuclear Power Plant (Ginna). In its LARs, Exelon proposes changes to its license to adopt risk informed completion times, and to adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors".

During the initial review of the LARs, the U.S. Nuclear Regulatory Commission (NRC) staff identified several items that require further clarification and detailed explanations. The NRC staff will conduct a regulatory audit to support its review of the LARs in accordance with the enclosed audit plan. A regulatory audit is a planned activity that includes the examination and evaluation of primarily non-docketed information. The audit will be conducted to increase the NRC staff's understanding of the LARs and identify information that will require docketing to support the NRC staff's regulatory findings.

The combined audit will be conducted using video conferencing and a web portal (also known as an eDocs portal, online portal, electronic portal, ePortal, electronic reading room) from September 13 to September 16, 2021. The logistics and scope of this audit were discussed with your staff on August 12, 2021. The audit plan is enclosed.

If you have any questions, please contact me by telephone 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 Plan
Attachment 1: Initial Audit Material Request
Attachment 2: Audit Questions

cc: Listserv

SUBJECT: R.E. GINNA NUCLEAR POWER PLANT – AUDIT PLAN IN SUPPORT OF REVIEW OF LICENSE AMENDMENT REQUEST REGARDING TSTF-505, REVISION 2, "PROVIDE RISK-INFORMED EXTENDED COMPLETION TIMES - RITSTF INITIATIVE 4B" AND 10 CFR 50.69, "RISK-INFORMED CATEGORIZATION AND TREATMENT OF STRUCTURES, SYSTEMS AND COMPONENTS FOR NUCLEAR POWER REACTORS" (EPID L-2021-LLA-0091 AND EPID L-2021-LLA-0092) DATED AUGUST 25, 2021

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OFFICE	NRR/DORL/LPL1/PM	NRR/DORL/LPL1/LAiT	NRR/DRA/APLA/BC	NRR/DSS/SNSB/BC
NAME	VSreenivas	KZeleznock	RPascarelli	SKrepel
DATE	08/10/2021	08/13/2021	07/14/2021	08/12/2021
OFFICE	NRR/DEX/EEEB/BC	NRR/DORL/LPL1/BC	NRR/DORL/LPL1/PM	
NAME	SRay	JDanna	VSreenivas	
DATE	08/12/2021	08/24/2021	08/24/2021	

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

COMBINED AUDIT PLAN

REGARDING RISK INFORMED COMPLETION TIMES -TSTF-505 AND

TO ADOPT 10 CFR 50.69 LICENSE AMENDMENT REQUESTS

EXELON GENERATION COMPANY, LLC

R.E. GINNA NUCLEAR POWER PLANT

DOCKET NOS. 50-244

1.0 BACKGROUND

By letters dated May 20, 2021 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML21140A324 and ML21141A009), Exelon Generation Company, LLC (Exelon) submitted license amendment requests (LARs) for R.E. Ginna Nuclear Power Plant (Ginna). In its LARs, Exelon proposes changes to its license to adopt risk informed completion times, and to adopt 10 CFR 50.69, "Risk-informed categorization and treatment of structures, systems and components for nuclear power reactors".

The proposed amendments would modify technical specification (TS) requirements to permit the use of risk informed completion times in accordance with Technical Specifications Task Force (TSTF)-505, Revision 2, "Provide Risk-Informed Extended Completion Times - RITSTF Initiative 4b," (ADAMS Accession No. ML18183A493). The proposed amendments would also modify the Ginna licensing basis by the addition of a license condition to allow for the implementation of the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Part 50.69, "Risk-Informed Categorization and Treatment of Structures, Systems and Components for Nuclear Power Reactors."

2.0 REGULATORY AUDIT BASES

A regulatory audit is a planned license or regulation-related activity that includes the examination and evaluation of primarily non-docketed information. The audit is conducted with the intent to gain understanding, to verify information, and to identify information that will require docketing to support the basis of a licensing or regulatory decision. Performing a regulatory audit is expected to assist the NRC staff in efficiently conducting its review and gaining insights to the licensee's processes and procedures. Information that the NRC staff relies upon to make the safety determination must be submitted on the docket.

3.0 SCOPE

The audit team will view the documentation and calculations that provide the technical support for the LARs. The scope of the NRC staff's audit will focus on the following subjects:

- Understand how the licensee's proposed program implements TSTF-505 and conforms to NRC-endorsed guidance in Nuclear Energy Institute (NEI) report NEI 06-09, Revision 0-A, "Risk-Informed Technical Specification Initiative 4b, Risk-Managed Technical Specification Guidelines."

- Understand how the licensee's proposed program implements 10 CFR 50.69, SSC Categorization to NRC-endorsed guidance in NEI 00-04, Revision 0, "SSC Categorization Guideline", as endorsed by RG 1.201, Revision 1 "Guidelines for Categorizing Structures, Systems, and Components in Nuclear Power Plants According to Their Safety Significance." :

- Gain a better understanding of the detailed calculations, analyses, and bases underlying the LARs and confirm the staff's understanding of the LARs.

- Gain a better understanding of plant design features and their implications for the LARs.

- Identify any information needed to enable the staff's evaluation of the technical acceptability of the probabilistic risk assessment (PRA) used for these applications.

- Identify any information needed to enable the staff's evaluation of whether the proposed changes challenge design-basis functions or adversely affect the capability or capacity of plant equipment to perform design-basis functions.

- Identify questions and requests that may become formal requests for additional information (RAIs) per NRR Office Instruction LIC-115, "Processing Requests for Additional Information."

The NRC staff will audit the PRA methods that the licensee would use to determine the risk impact from which the revised completion times for TSTF-505 would be obtained, including the licensee assessments of internal events (including internal flooding) and fire PRAs. The NRC will also audit the licensee's quantification of risk from significant external events, whether the licensee uses PRA or bounding methods, and the licensee's evaluation of defense-in-depth.

In addition, the audit team will request to discuss these topics with Exelon's subject matter experts. The NRC staff will conduct this audit under the guidance provided in NRR Office Instruction LIC-111, "Regulatory Audits," Revision 1 (ADAMS Accession No. ML19226A274).

4.0 INFORMATION AND OTHER MATERIAL NECESSARY FOR THE REGULATORY AUDIT

The NRC staff will request information and interviews throughout the audit period. The NRC staff will use an "audit items list" to identify the information (e.g., methodology, process information, and calculations) to be audited and the subjects of requested interviews and meetings. The NRC staff requests the licensee to have the requested audit information listed in the audit items list to be readily available and accessible for the NRC staff's review via a web portal.

The NRC staff requests the licensee to have the information referenced in Attachment 1 of this audit plan available and accessible for the NRC staff's review via a web portal within two weeks

of the date of this audit plan. The NRC staff requests that any supplemental information requested be available and accessible for the NRC staff's review within one week of the date of the NRC's notification to the licensee of the new requests.

The staff acknowledges and will observe appropriate handling and protection of proprietary information made available for the audit. The NRC staff will not remove non-docketed information from the audit site or web portal.

5.0 AUDIT TEAM

The following are the NRC audit team members and their respective areas of focus during the audit:

NRR Staff	Email	50.69	505	RTR[1]	QA[2]	Division/Branch
V. Sreenivas, DORL Project Manager	V.Sreenivas@nrc.gov		✓			DORL/LPL1
Adrienne Brown	Adrienne.Brown@nrc.gov		✓	✓		DRA/APLA
Jonathan Evans	Jonathan.Evans@nrc.gov		✓		✓	
Alex Schwab	Alexander.Schwab@nrc.gov			✓		
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		✓			
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

6.0 LOGISTICS

The audit will be conducted using video conferencing and a web portal (also known as an eDocs portal, online portal, electronic portal, ePortal, or electronic reading room) from September 13 to September 16, 2021. During the audit, information should be shared using video conference and telephone conference. The audit kick-off meeting will be held at 10:00 am

[1] RTR, Reactor Technical Reviewer: lead staff member responsible for their technical scope of review.
[2] QA:, Quality Assurance: staff member responsible for review of the quality of RTR, and provides concurrence

on September 13, 2021. A detailed audit schedule and topics to be discussed will be provided to the licensee and confirmed prior to the onset of the audit.

Representatives of Exelon are requested to be available for video or audio conferences on the days and times on the audit schedule below. The NRC project manager will coordinate any changes to the audit schedule and location with the licensee. The NRC staff requests remote access to the available documents listed in Section 4.0.

Audit Schedule		
Date	Time	Subject
September 13, 2021	10:00 am – 3:00pm	Audit Kick-Off Meeting and Scheduled Discussions
September 14, 2021	10:00 am – 3:00pm	Scheduled Discussions
September 15, 2021	10:00 am – 3:00pm	Scheduled Discussions
September 16, 2021	10:00 am – 3:00pm	Scheduled Discussions and Audit Exit Meeting

7.0 SPECIAL REQUESTS

The following conditions associated with the web portal must be maintained throughout the duration so that the NRC staff has access to the web portal:

- The web portal will be password-protected, and separate passwords will be assigned to the NRC staff who are participating in the audit.
- The web portal will be sufficiently secure to prevent the NRC staff from printing, saving, downloading, or collecting any information from the web portal.
- Conditions of use of the web portal will be displayed on the login screen and will require acknowledgment by each user.

Exelon should provide username and password information directly to the NRC staff. The NRC project manager will provide Exelon the names and contact information of the NRC staff who will be participating in the audit. All other communications should be coordinated with the NRC project manager. NRC staff access to the web portal will be terminated 30 days after the end of the regulatory audit.

8.0 DELIVERABLES

On completion of the audit, the staff will prepare an audit summary report within 60 days that will be entered as an official agency record in ADAMS. If the NRC staff identifies information during the audit that is needed to support its regulatory decision, the staff will issue RAIs to the licensee after the audit.

ITEM	AUDIT REQUEST
1	Reports of peer reviews (full-scope and focused-scope), self-assessments, and Facts & Observations (F&Os) closure reviews for the internal events, internal flooding, and fire PRAs cited in LaSalle's LAR dated May 20, 2021; • LTR-RAM-II-09-049, "RG 1.200 PRA Peer Review Against the ASME PRA Standard Requirement for the R.E. Ginna Station Probabilistic Risk Assessment", August 2009. • LTR-RAM-II-12-066, "Fire PRA Peer Review Against the Fire PRA Standard Supporting Requirements from Section 4 of the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessments for Nuclear Power Plant Applications", August 6, 2012. • For the Ginna Fire Probabilistic Risk Assessment" 1BT132299.006.058.100, "Ginna Nuclear Power Plant PRA Finding-Level Fact and Observation Technical Review", August 2017. • G1-MISC-023, Revision 0, "Risk Management Finding Level F&O Independent Assessment Ginna", March 2020.
2	Uncertainty notebooks for the Ginna internal events, internal flooding, and fire PRAs related to PRA model assumptions and sources of uncertainty; • G1-FQ-F001, Revision 4, "Fire PRA Notebook –Fire Risk Quantification (FQ)", August 2019. • G1-UNC-F001, Revision 3, "Fire PRA Notebook Uncertainty and Sensitivity Analysis (UNC), Revision 3, July 2015. • G1-MISC-026, Revision 0, "Assessment of Key Assumptions and Sources of Uncertainty for the R.E. Ginna PRA," April 2020.
3	Documentation of evaluation of the generic and plant specific uncertainties with respect to the Ginna LAR dated May 20, 2021;
4	PRA notebooks for the modeling of FLEX equipment and FLEX human error probabilities, if credited in the PRA;
5	Results of the fire PRA and resolution of F&Os; • G1-FQ-F001, Revision 4, "Fire PRA Notebook –Fire Risk Quantification (FQ)", August 2019.
6	PRA notebooks for the modeling of FLEX equipment and FLEX human error probabilities credited in the PRA
7	External hazards analysis to include • Report G1-MISC-021, "Ginna External Hazards Assessment," Revision 0, April 2021. • Reports-2016-0519 TMP, "Tornado/Wind Generated Missile Vulnerability Evaluation, Tornado Missile Project (TMP), Ginna Station," Revision 1, November 2017. • DA-CE-17-001, "Tornado Missile Protection Structural Barriers, ECP-17-000388," Revision 0, May 2018. • DA-ME-21-001, "Assessment of Tornado Missile Barriers for Ginna RICT," Revision 0, March 2021. • IPEEE High Winds and Transportation Report, from Robert C. Mecredy, Rochester Gas and Electric Corporation, to USNRC, August 19, 1998. • Enclosure to CF 18139-30157, Project Number: 1041, "Ginna Seismic Hazard and Screening Report", November 2013.

ITEM	AUDIT REQUEST
	• Exelon Design Analysis: FHR-FLOOD-FREQ, "Fukushima Flood Hazard Reevaluation - Flood-Frequency Analysis for Localized and Stream Flooding," Revision 0, August 2020.
8	Available 10 CFR 50.69 SSC categorization program procedures (e.g., categorization review and adjustment process, decision criteria for Independent Decision-Making Panel (IDP)); and
9	Any draft or final RICT program procedures (e.g., for risk management actions, PRA functionality determination, and recording limiting conditions for operation) • Exelon Procedure OP-AA-201-012-1001, "Operations On-Line Fire Risk Management" • A-601.16, “Online Fire Risk Management”. • Exelon Procedure WC-AA-101-1006, "On-Line Risk Management and Assessment" • Exelon Procedure OP-AA-108-117, "Protected Equipment Program."
10	Plant and PRA configuration control procedures • ER-AA-600-1015, "FPIE PRA Model Update." • ER-AA-600-1061, "Fire PRA Model Update and Control.”
11	Relevant design documentation, e.g., single line diagrams of the electrical power distribution systems and piping and instrumentation diagrams
12	Load list for each safety-related bus
13	Plant procedures related to the risk management action for the electrical power systems, if available
14	Any other supporting documentation that the licensee may determine is responsive to the staff’s above information requests; • RA-003, Rev 4, “Ginna Nuclear Power Plant Probabilistic Risk Assessment • Success Criteria Notebook”, December 2020

ATTACHMENT 2: AUDIT QUESTIONS

R.E. GINNA NUCLEAR POWER PLANT
LICENSE AMENDMENT REQUEST (LAR) TO
ADOPT TSTF-505, REVISION 2, AND 10 CFR 50.69
DOCKET NUMBERS 50-244
FACILITY OPERATING LICENSE NOS. DPR-18

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 (ADAMS Accession No. ML090410014) and Revision 3 (ADAMS Accession No. 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 (ADAMS Accession No. ML1527A101) 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 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 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 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, Main Report," dated March 2017 (ADAMS Accession No. 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 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 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 states 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, 2020a 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" (ADAMS Accession No. 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 (ADAMS Accession No. ML090410014).

With regards to equipment failure probability, in the May 30, 2017 memo, 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 memo, 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 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 STGR 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 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" (ADAMS Accession No. ML12074A115). As part of the initial Voluntary Industry Initiative (VII) for mitigation of the potential for the occurrence of an OPC in electrical switchyards[3], 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[4] and resulting industry guidance in NEI 19-02[5] 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.

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[3] Anthony R. Pietrangelo to Mark A. Satorius, Ltr re: "Industry Initiative on Open Phase Condition - Functioning of Important-to-Safety Structures, Systems and Components (SSCs)", 10/9/2013 (ADAMS Accession No. ML13333A147).

[4] Doug True to Ho Nieh, Ltr re: "Industry Initiative on Open Phase Condition, Revision 3", 6/6/19 (ADAMS Accession No. ML19163A176)

[5] Nuclear Energy Institute, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights", NEI 19-02 R0, April 2019 [ADAMS Accession No. ML19122A321].

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 (ADAMS Accession No. 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 12-13, 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 (ADAMS Accession No. ML 17079A427).

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 (ADAMS Accession No. ML1527A101) 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML20168A655).
- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities DELORESVEWFIRE), dated December 2016 (ADAMS Accession No. ML16343A058).
- NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," dated January 2015 (ADAMS Accession No. 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," ADAMS Accession No. ML093220426, has been retired by letter dated July 1, 2016, (ADAMS Accession No. 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 (ADAMS Accession No. ML1527A101) 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 (ADAMS Accession No. ML 18333A022). 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 (ADAMS Accession No. 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.

APLC Question 02 – Risk contribution of a seismic event [10 CFR 50.69]

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.

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]

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) (ADAMS Accession No. 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 of10 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 (ADAMS Accession No. 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 (ADAMS Accession No. ML071200238) states that "[w]here [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 (ADAMS Accession No. 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 (ADAMS Accession No. ML071200238) states that "[w]here [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 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 (ADAMS Accession No. 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."

ATTACHMENT 2: AUDIT QUESTIONS

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.

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 Closur	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

			"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.