From: Surinder Arora
Sent: Tuesday, July 30, 2024 2:30 PM
To: Eric Frank
Cc: Roxanne Vonhabsburg; Derek C Corrin
Subject: Fermi 2: Audit Questions from the Audit Team Supporting Review of the LAR to Adopt TSTF-505 (EPID: L-2024-LLA-0034)
Attachments: Fermi 2 TSTF-505 LAR Audit Questions.pdf

Importance: High

Dear Eric Frank,

I am emailing you the attached audit questions based on the staff's review of your amendment request to adopt TSTF-505 (ML24081A326) and the documents that have been uploaded by your staff on the audit portal to support the review of your license amendment request.

Please review these questions and let me know when your staff will be prepared to meet with the audit team to discuss the responses to the staff's questions. Please let me know so that we can plan a three-day audit meeting sometime in September. Please consider a window from Tuesday to Thursday for this meeting and let me know which week of September suits your staff.

Your prompt response will be highly appreciated. Feel free to contact me if you have any questions.

Thanks,
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Project Manager,
Fermi 2 and Dresden 2 & 3
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AUDIT QUESTIONS

LICENSE AMENDMENT REQUEST TO REVISE TECHNICAL SPECIFICATIONS TO

ADOPT TSTF-505, REVISION 2

DTE ELECTRIC COMPANY

FERMI POWER PLANT, UNIT 2

DOCKET NO. 50-341

By application dated March 21, 2024 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML24081A326), DTE Electric Company (DTE, the licensee) submitted a license amendment request (LAR) for Fermi Power Plant, Unit 2 (Fermi 2). The amendment 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. The proposed changes are based on Technical Specifications Task Force (TSTF) Traveler TSTF-505, Revision 2, dated July 2, 2018 (ADAMS Accession No. ML18183A493). The U.S. Nuclear Regulatory Commission (NRC) issued a final revised model safety evaluation (SE) (ADAMS Accession No. ML18269A041) approving TSTF-505, Revision 2, on November 21, 2018.

The NRC staff has determined that the response to the following audit questions is needed to complete its review.

Probabilistic Risk Assessment (PRA) Licensing Branch A (APLA) Fermi 2 TSTF-505 Audit Questions

APLA Question 01 – Open Fire PRA Facts and Observations (F&O) Closeout

Regulatory Guide (RG) 1.200, Revision 2[1] provides guidance for addressing PRA acceptability. RG 1.200, Revision 2, describes a peer review process using the American Society of Mechanical Engineers/American Nuclear Society (ASME/ANS) PRA Standard ASME/ANS-RA-Sa-2009[2], as one acceptable approach for determining the technical acceptability of the PRA. The primary results of a 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

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[1] Regulatory Guide 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment for Risk-Informed Activities," Revision 2, dated March 2009 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML090410014).

[2] American Society of Mechnical Engineers (ASME) and American Nuclear Society (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," dated February 2009, New York, NY (Copyright).

NEI 05-04, NEI 07-12, and NEI 12-13[3], which was accepted by the NRC in a letter dated May 3, 2017[4].

Section 4 of Enclosure 2 to the LAR states that, regarding the Fire PRA model, a full-scope peer review was conducted in August 2020 and that a Finding Level F&O Independent Assessment (IA) was conducted in November 2020. The LAR continues by stating that the IA Team (IAT) report mistakenly identified an F&O, 3-13, as a finding level F&O, when in fact the final peer review report identified it as a suggestion level F&O. The IAT report appears to state that 25 finding level F&Os were closed, whereas the LAR appears to state 23 F&Os were closed by this IAT.

The NRC staff notes that the full-scope peer review final description and resolution to a finding level F&O plays a critical role for the IAT members to assess closure of a finding level F&O. Based on the inconsistencies stated in the LAR regarding this issue, it is unclear to the NRC staff if the IAT closure review was conducted with the final approved full-scope peer review findings description and resolution.

a) Clarify if the IAT, at the time of their review, were provided the final approved description and resolution recommendations for the fire PRA finding level F&Os. Include in this discussion the cause of the discrepancy in the IAT report of the number of finding level F&Os.

b) For the finding level F&Os, if any, descriptions and resolutions that were provided to the IAT for their review that did not match the final peer review report provide the following information:

    i. Identify each of the impacted F&Os.

    ii. For each F&O listed in part i, provide justification that the use of the description and resolution provided to the IAT, that was not the final approved version did not significantly impact the IAT closure determination.

    iii. For those F&O closures identified in part ii that cannot be justified, provide a disposition for each F&O on its impact to this application.

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[3] Anderson, V. K., Nuclear Energy Institute, letter to Stacey Rosenberg, U.S. Nuclear Regulatory Commission, "Final Revision of Appendix X to NEI 05-04/07-12/12-1[3], Close-Out of Facts and - Observations (F&Os)," dated February 21, 2017 (ADAMS Accession No. ML17086A431).
[4] Giitter, J., and Ross-Lee, M.J., U.S. Nuclear Regulatory Commission, letter to Krueger, G., Nuclear Energy Institute, "U.S. Nuclear Regulatory Commission Acceptance on Nuclear Energy Institute Appendix X to Guidance 05-04, 07-12, and 12-13, Close-Out of Facts and Observations (F&Os)", May 3, 2017 (ADAMS Accession No. ML17079A427).

APLA Question 02 – Determinations of Key Assumptions and Sources of Uncertainty

Section 2.3.4 of NEI 06-09[5] states that PRA modeling uncertainties be considered in application of the PRA base model results to the RICT program. The NRC SE[6] for NEI 06-09 states that this consideration is consistent with Section 2.3.5 of RG 1.177, Revision 1, "An Approach for Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications" (ADAMS Accession No. ML100910008). NEI 06-09 further states 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 and that sensitivity studies should be used to develop appropriate compensatory risk management actions (RMAs).

a) Item 9 in Table 4 of 01T013-RPT-04, "Assessment of Key Assumptions and Sources of Uncertainty for the Fermi 2 TSTF-505 Application," regarding the assumptions for quasi steady failure threshold methods to determine failure pressure is a point estimate (or 'best estimate'). The disposition continues by stating that the failure pressure value can utilize a probability density function (pdf) that can produce a mean value.

   The NRC staff notes that pdf derived means can be non-conservative when compared to point estimates. It is also unclear what processes were used to determine the point estimate, such as MAAP or an industry evaluation.

   i. Clarify if the point estimate is the result of a deterministic evaluation, such as MAAP or industry evaluation.

   ii. Justify the use of the point estimate instead of the derived mean value from the related pdf. Include in this discussion the mean value and if it is conservative or non-conservative when compared to the point estimate.

   iii. If the point estimate value is non-conservative, then provide justification that the use of the point estimate does not impact this application.

b) Item 12 in Table 4 of 01T013-RPT-04, "Assessment of Key Assumptions and Sources of Uncertainty for the Fermi 2 TSTF-505 Application," regarding the thermal-induced failure of the RCS boundary performed several MAAP sensitivities and that the range of effects were captured. However, there is no mention of the impact on the results based on these sensitivities.

   i. Provide discussion of the impact of these sensitivity studies for this application.

   ii. If the sensitivities demonstrate a significant impact on this application than propose how this source of uncertainty will be addressed in accordance with Section 2.3.4 of NEI-06-09.

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[5] Bradley, B., Nuclear Energy Institute, letter to S. D. Stuchell, U.S. Nuclear Regulatory Commission, NEI 06-09, "Risk Informed Technical Specifications Initiative 4b; Risk Managed Technical Specifications (RMTS) Guidelines," Revision 0-A, October 2012 (ADAMS Accession No. ML122860402).

[6] Golder, J. M., U.S. Nuclear Regulatory Commission, letter to B. Bradley, Nuclear Energy Institute, "Final Safety Evaluation for Nuclear Energy Insittute (NEI) Topical Report (TR) NEI 06-09, 'Risk-Infomed Techncial Specification Initiative 4b, Risk-Managed Technical Specifications (RMTS) Guidelines' (TAC No. MD4995)," May 17, 2007 (ADAMS Accession No. ML071200238).

APLA Question 03 – Open Phase Condition

Section C.1.4 of RG 1.200 states the base (e.g., Model of Record) PRA is to represent the as-built, as-operated plant to the extent needed to support the application. The licensee is to have a process that identifies updated plant information that necessitate changes to the base PRA model.

In response to the January 30, 2012, Open Phase Condition (OPC) event at the Byron Generating Station, the NRC staff had issued Bulletin 2012-01[7]. As part of the initial Voluntary Industry Initiative (VII) for mitigation of the potential for the occurrence of an OPC in electrical switchyards[8], licensees have made the addition of an Open Phase Isolation System (OPIS). As per SRM-SECY-16-0068[9], the NRC staff was directed to ensure that the licensees have appropriately implemented OPIS and that licensing bases have been updated accordingly. From the revised voluntary initiative[10] and resulting industry guidance in NEI 19-02[11] 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 response to the NRC audit portal request regarding OPC and OPIS, the licensee stated that neither OPC nor OPIS is modeled in any Fermi 2 PRA model but has been entered into their tracking system (PSA WR 2058) for future evaluation. Considering these observations, provide the following information:

a) For Fermi 2, discuss the evaluation of the risk impact associated with OPC events including the likelihood of OPC initiating plant trips and the impact of those trips on PRA-modeled SSCs. Also, explain whether an OPIS has been installed and if it has been installed, then discuss its functionality and any operator actions needed to operate the system or needed in response to the system.

b) Provide justification that the exclusion of this failure mode and mitigating system does not impact RICT calculations.

c) As an alternative to Part (b), propose a mechanism to ensure that OPC-related scenarios are incorporated into the application PRA models prior to implementing the RICT.

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[7] U.S. NRC Bulletin 2012-01, "Design Vulnerability in Electric Power System" (ADAMS Accession No. ML12074A115).

[8] 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)," dated October 9, 2013 (ADAMS Accession No. ML13333A147).

[9] U.S. NRC SRM-SECY-16-0068, "Interim Enforcement Policy for Open Phase Conditions in Electric Power Systems for Operating Reactors," dated March 9, 2017 (ADAMS Accession No. ML17068A297).

[10] Doug True to Ho Nieh, Ltr re: "Industry Initiative on Open Phase Condition, Revision 3," dated June 6, 2019 (ADAMS Accession No. ML19163A176).

[11] Nuclear Energy Institute (NEI) 19-02, "Guidance for Assessing Open Phase Condition Implementation Using Risk Insights," Revision 0, April 2019 (ADAMS Accession No. ML19122A321).

APLA Question 04 – Impact of Seasonal Variations

The Tier 3 requirement of RG 1.177, Revision 2, "Plant-Specific, Risk-Informed Decisionmaking: Technical Specifications," dated January 2021, stipulates that a licensee should develop a program that ensures that the risk impact of out-of-service equipment is appropriately evaluated prior to performing any maintenance activity.

Section 2.3.4 of NEI 06-09-A states, in part, that:

> If the PRA model is constructed using data points or basic events that change as a result of time of year or time of cycle…, then the RICT calculation shall either 1) use the more conservative assumption at all time, or 2) be adjusted appropriately to reflect the current (e.g., seasonal or time of cycle) configuration for the feature as modeled in the PRA.

Enclosure 8 of the LAR states that seasonal dependencies are addressed in the Real-Time Risk (RTR) model. However, it does not appear to specify the modeling adjustments needed to account for seasonal variations and what kind of adjustments will be made. Therefore, address the following to clarify the treatment of seasonal and time of cycle variations:

a) Explain how the RICT calculations address changes in PRA data points, basic events, and SSC operability constraints as a result of extreme weather conditions, seasonal variations, other environmental factors, or time of cycle. Also, explain how these adjustments are made in the configuration risk management program (CRMP) model and how this approach is consistent with the guidance in NEI 06-09-A and its associated NRC final SE.

b) Describe the criteria used to determine when PRA adjustments due to extreme weather conditions, seasonal variations, other environmental factors, or time of cycle variations need to be made in the CRMP model and what mechanism initiates these changes.

APLA Question 05 – Performance Monitoring

The NRC SE for NEI 06-09-A, states in part: "The impact of the proposed change should be monitored using performance measurement strategies." NEI 06-09-A considers the use of NUMARC 93-01, Revision 4F, "Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants" (ADAMS Accession No. ML18120A069), as endorsed by RG 1.160, "Monitoring the Effectiveness of Maintenance at Nuclear Power Plants," Revision 4 (ADAMS Accession No. ML18220B281), for the implementation of the Maintenance Rule. NUMARC 93-01, Section 9.0, contains guidance for the establishment of performance criteria.

In addition, the NEI 06-09-A methodology satisfies the five key safety principles specified in RG 1.177, Revision 2, relative to the risk impact due to the application of a RICT. Moreover, NRC staff position C.3.2 provided in RG 1.177, Revision 2, for meeting the fifth key safety principle acknowledges the use of performance criteria to assess degradation of operational safety over a period. It is unclear how the licensee's RICT program captures performance monitoring for the SSCs within the scope of the RMTS program.

Attachment 8 to the LAR mentions the use of NUMARC 93-01, as endorsed by RG 1.182, "Assessing and Managing Risk Before Maintenance Activities at Nuclear Power Plants," in performing maintenance risk evaluations. However, Enclosure 11 to the LAR mentions that the existing Maintenance Rule monitoring programs under 10 CFR 50.65 provide for the impacts of the RICT Program but does not appear to mention NUMARC 93-01.

a) Confirm that the Fermi Maintenance Rule program incorporates the use of performance criteria to evaluate SSC performance as described in NUMARC 93-01, as endorsed by RG 1.160.

b) Alternatively, describe the approach or method used by Fermi 2 for SSC performance monitoring, as described in NRC staff position C.3.2 of RG 1.177, Revision 2, for meeting the fifth key safety principle. In the description, include criteria (e.g., qualitative, or quantitative) along with the appropriate risk metrics, and explain how the approach and criteria demonstrate the intent to monitor the potential degradation of SSCs in accordance with the NRC SE for NEI 06-09-A.

APLA Audit Question 06 – In-Scope LCOs and Corresponding PRA Modeling

The NRC's safety evaluation for NEI 06-09-A specifies that the LAR should provide a comparison of the TS functions to the PRA modeled functions to show that the PRA modeling is consistent with the licensing basis assumptions or to provide a basis for when there is a difference. Table E1-1 of LAR Enclosure 1 identifies each Limiting Condition for Operation (LCO) in the TSs proposed for inclusion in the RICT program. The table also describes whether the systems and components covered by the LCO are modeled in the PRA and, if so, presents both the design success criteria and PRA success criteria. For certain LCOs, the table explains that the associated SSCs are not modeled in the PRAs but will be represented using a surrogate event that fails the function performed by the SSC. For some LCOs, the LAR did not provide an adequate description for the NRC staff to conclude that the PRA modeling will be sufficient.

a) Regarding TS LCO 3.3.1.1.A/B, Table E1-1 states that, for reactor protection system instrumentation not modeled, that an item that is modeled will be used as a surrogate. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.3.1.1.A/B.

b) Regarding TS LCO 3.3.5.1.B/C, Table E1-1 states that, for emergency core cooling system instrumentation inputs that are not modeled, that an item that is modeled will be used as a surrogate, with relay logic failure provided as an example. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.3.5.1.B/C.

c) Regarding TS LCO 3.3.5.1.D, Table E1-1 states that, for emergency core cooling system instrumentation for the high pressure core injection trip on suppression pool high water level instrumentation that is not modeled, that an item that is modeled will be used as a surrogate or failure of the frontline system. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.3.5.1.D.

d) Regarding TS LCO 3.3.5.1.E/F, Table E1-1 states that, for reactor protection system instrumentation automatic depressurization system (ADS) initiation logic and instrumentation not modeled, that the emergency operating procedures direct manual inhibition of the ADS and an instrument that is modeled will be used as a surrogate. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.3.5.1.E/F.

e) e) Regarding TS LCO 3.6.1.3.A, Table E1-1 states that, for primary containment isolation valves not modeled, that either a containment bypass or isolation failure that is modeled will be used as a surrogate. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.6.1.3.A.

f) Regarding TS LCO 3.7.2.A, Table E1-1 states that, for ultimate heat sink reservoir not modeled, that items that are modeled will be used as a surrogate. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.7.2.A.

g) Regarding TS LCO 3.7.6.A, Table E1-1 states that, for steam jet air ejector not modeled, that an item that is modeled will be used as a surrogate. It is unclear to the NRC staff what the specific surrogate is and how it is either conservative or bounding.

   Identify the surrogate and provide justification that the surrogate conservatively bounds TS LCO 3.7.6.A.

APLA Audit Question 07 – PRA Update Process

Section 2.3.4 of NEI 06-09-A specifies, "criteria shall exist in PRA configuration risk management to require PRA model updates concurrent with implementation of facility changes that significantly impact RICT calculations."

LAR Enclosure 7 states that if a plant change or a discovered condition is identified and can have significant impact on the RICT calculations, then an unscheduled update of the PRA models will be implemented. More specifically, the LAR states that if the plant changes meet specific criteria defined in the plant PRA and update procedures, including criteria associated with consideration of the cumulative risk impact, then the change will be incorporated into applicable PRA models without waiting for the next periodic PRA update. The LAR does not explain under what conditions an unscheduled update of the PRA model will be performed or the criteria defined in the plant procedures that will be used to initiate the update.

Considering these observations, describe the conditions under which an unscheduled PRA update (i.e., more than once every two refueling cycles) would be performed and the criteria that would be used to require a PRA update. In the response, define what is meant by “significant impact to the RICT Program calculations.”

APLA Audit Question 08 – PRA Model Uncertainty Analysis Process

The NRC staff SE to NEI 06-09-A specifies that the LAR should identify key assumptions and sources of uncertainty and assess and disposition each as to its impact on the RICT application. Section 2.3.4 of NEI 06-09-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 that could potentially impact the results of a RICT calculation. 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 qualitatively screening model uncertainties.

LAR Enclosure 9, Section 3 (Assessment of Internal Events PRA Epistemic Uncertainty Impacts) indicates that the process used to evaluate sources of uncertainty in the internal events PRA follows the guidance in NUREG-1855, Revision 1, and EPRI TR-1016737, “Treatment of Parameter and Model Uncertainty for Probabilistic Risk Assessments.” The LAR states that for the internal events PRA both plant-specific sources of uncertainty from the PRA notebooks and generic sources of uncertainty compiled by Electric Power Research Institute (EPRI) were considered. However, it is not clear to the NRC staff what EPRI reports were used to identify the generic sources of uncertainty for the internal events PRA, including internal flooding and large early release frequency (LERF).

Furthermore, LAR Enclosure 9, Section 3 presents one source of modeling uncertainty for the internal events PRA (i.e., involving the joint human error probability floor) and states that this uncertainty does not present a significant impact on the Fermi 2 RICT calculations. No other assessment of candidate assumptions or sources of uncertainty for the internal events PRA was provided in the LAR. From PRA Notebook 01T013-RPT-04 (Assessment of Key Assumptions and Sources of Uncertainty for the Fermi 2 TSTF-505 Application), it appears that a master compilation of plant-specific and generic industry PRA modeling assumptions and sources of uncertainty was screened using a set of criteria to determine that none of the applicable PRA modeling assumptions and sources of uncertainty are “key” to this application. However, it is not clear to NRC staff what evaluation criteria were used to consistently evaluate plant-specific and generic sources of uncertainty to conclude that none are “key” for this application.

Considering these observations, provide the following information:

a) Identify the EPRI reports that were used to identify the generic sources of uncertainty for the internal events PRA, including internal flooding and LERF.

b) Describe and justify the criteria used to consistently evaluate a comprehensive list of internal events PRA modeling assumptions and sources uncertainty (including those associated with plant-specific features, modeling choices, and generic industry concerns) to conclude none are “key” to the Fermi 2 RICT program.

PRA Licensing Branch B (APLB) Fermi Unit 2 TSTF-505 Audit Questions

APLB Question 01 – Use of Unacceptable Methods

The LAR provides the history of the Fire PRA (FPRA) peer review but does not appear to discuss methods used in the FPRA. Methods may have been used in the FPRA that deviate from guidance in NUREG/CR-6850, "EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities," (ADAMS Accession Nos. ML052580075, ML052580118, and ML103090242), or other acceptable guidance (e.g., frequently asked questions (FAQs), NUREGs, or interim guidance documents).

a) Identify methods used in the FPRA that deviate from guidance in NUREG/CR-6850 or other acceptable guidance.

b) If such deviations exist, then justify their use in the FPRA and impact on the RICT program.

c) As an alternative to item b above, add an implementation item to replace those methods with a method acceptable to NRC prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

APLB Question 02 – Reduced Transient Heat Release Rates (HRRs)

The key factors used to justify using transient fire reduced heat release rates (HRRs) below those prescribed in NUREG/CR-6850 are discussed in the June 21, 2012, letter from Joseph Giitter, U.S. Nuclear Regulatory Commission, to Biff Bradley, NEI, "Recent Fire PRA Methods Review Panel Decisions and Electrical Power Research Institute (EPRI) 1022993, “Evaluation of Peak Heat Release Rates in Electrical Cabinet Fires," (ADAMS Package Accession No. ML12171A583).

If any reduced transient HRRs below the bounding 98% HRR of 317 kW from NUREG/CR-6850 were used, discuss the key factors used to justify the reduced HRRs. Include in this discussion:

a) Identification of the fire areas where a reduced transient fire HRR is credited and what reduced HRR value was applied.

b) A description for each location where a reduced HRR is credited, and a description of the administrative controls that justify the reduced HRR including how location-specific attributes and considerations are addressed. Include a discussion of the required controls for ignition sources in these locations and the types and quantities of combustible materials needed to perform maintenance. Also, include discussion of the personnel traffic that would be expected through each location.

c) The results of a review of records related to compliance with the transient combustible and hot work controls.

APLB Question 03 – Treatment of Sensitive Electronics

FAQ 13-0004, "Clarifications on Treatment of Sensitive Electronics" (ADAMS Accession No. ML13322A085) provides supplemental guidance for application of the damage criteria provided in Sections 8.5.1.2 and H.2 of NUREG/CR-6850, Volume 2, for solid-state and sensitive electronics.

a) Describe the treatment of sensitive electronics for the FPRA and explain whether it is consistent with the guidance in FAQ 13-0004, including the caveats about configurations that can invalidate the approach (i.e., sensitive electronics mounted on the surface of cabinets and the presence of louver or vents).

b) If the approach cannot be justified to be consistent with FAQ 13-0004, then justify that the treatment of sensitive electronics has no impact on the RICT calculations.

c) As an alternative to item b above, add an implementation item to replace the current approach with an acceptable approach prior to the implementation of the RICT program. Include a description of the replacement method along with justification that it is consistent with NRC accepted guidance.

APLB Question 04 – Obstructed Plume Model

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," (ADAMS Accession No. ML15056A144) contains refined peak HRRs, compared to those presented in NUREG/CR-6850, and guidance on modeling the effect of plume obstruction. Additionally, NUREG-2178 provides guidance that indicates that the obstructed plume model is not applicable to cabinets in which the fire is assumed to be located at elevations of less than one-half of the cabinet.

a) If obstructed plume modeling was used, then indicate whether the base of the fire was assumed to be located at an elevation of less than one-half of the cabinet.

b) Justify any modelling in which the base of an obstructed plume is located at less than one half of the cabinet's height.

c) As an alternative to item b above, add an implementation item to remove credit for the obstructed plume model in the FPRA prior to the implementation of the RICT program.

APLB Question 05 – Well-Sealed Motor Control Center (MCC) Cabinets

Guidance in FAQ 08-0042 from Supplement 1 of NUREG/CR-6850 applies to electrical cabinets below 440V. With respect to Bin 15 as discussed in Chapter 6, it clarifies the meaning of "robustly or well-sealed." Thus, for cabinets of 440V or less, fires from well-sealed cabinets do not propagate outside the cabinet. For cabinets of 440V and higher, the original guidance in Chapter 6 remains and requires that Bin 15 panels which house circuit voltages of 440V or greater are counted because an arcing fault could compromise panel integrity (an arcing fault could burn through the panel sides, but this should not be confused with the high energy arcing fault type fires)." FPRA FAQ 14-0009, "Treatment of Well-Sealed MCC Electrical Panels Greater than 440V" (ADAMS Accession No. ML15113A241) provides the technique for evaluating fire damage from MCC cabinets having a voltage greater than 440V. Therefore, propagation of fire outside the ignition source panel must be evaluated for all MCC cabinets that house circuits of 440V or greater.

a) Describe how fire propagation outside of well-sealed MCC cabinets greater than 440V is evaluated.

b) If well-sealed cabinets less than 440V are included in the Bin 15 count of ignition sources, provide justification for using this approach as this is contrary to the guidance.

APLB Question 06 – Influence Factors for Transient Fires

NUREG/CR-6850, Section 6, “Fire Ignition Frequencies,” and FAQ 12-0064 “Hot Work/Transient Fire Frequency Influence Factors" (ADAMS Accession No. ML12346A488) describe the process for assigning influence factors for hot work and transient fires. Provide the following regarding application of this guidance:

a) Indicate whether the methodology used to calculate hot work and transient fire frequencies applies influencing factors using NUREG/CR-6850 guidance or FAQ 12-0064 guidance.

b) Indicate whether administrative controls are used to reduce transient fire frequency, and if so, describe and justify these controls

c) Indicate whether you have any combustible control violations and discuss your treatment of these violations for the assignment of transient fire frequency influence factors. For those cases where you have violations and have assigned an influence factor of 1 (Low) or less, indicate the value of the influence factors you have assigned and provide your justification.

d) If you have assigned an influencing factor of "0" to Maintenance, Occupancy, or Storage, or Hot Work for any fire physical analysis units (PAUs) provide justification.

e) If a weighting factor of "50" was not used in any fire PAU, provide a sensitivity study that assigns weighting factors of "50" per the guidance in FAQ 12-0064.

APLB Question 07 - Fire Scenario Treatment of the Main Control Board (MCB)

Traditionally, the cabinets on front face of the MCB have been referred to as the MCB for purposes of FPRA. Appendix L of NUREG/CR-6850, (ADAMS Accession Nos. ML052580075) provides a refined approach for developing and evaluating those fire scenarios. FPRA FAQ 14-0008, “Main Control Board Treatment dated July 22, 2014 (ADAMS Accession No. ML14190B307) clarifies the definition of the MCB and effectively provides guidance for when to include the cabinets on the back side of the MCB as part of the MCB for FPRA. It is important to distinguish between MCB and non-MCB cabinets because misinterpretation of the configuration of these cabinets can lead to incomplete or incorrect fire scenario development. This FAQ also provides several alternatives to NUREG/CR-6850 for using Appendix L to treat partitions in an MCB enclosure. Therefore, address the following:

a) Briefly describe the main control room MCB configuration, and use the guidance in FAQ 14-0008, to determine whether cabinets on the rear side of the MCB are a part of the MCB. Provide your justification using the FAQ guidance.

b) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure using the definition in FAQ 14-0008, then confirm that guidance in FAQ 14-0008 was used to develop fire scenarios in the MCB and determine the frequency of those scenarios.

c) If the cabinets on the rear side of the MCB are part of a single integral MCB enclosure and the guidance in FAQ 14-0008 was not used to develop fire scenarios involving the MCB, then provide a description of how the fire scenarios for the backside cabinets are developed and an explanation of how the treatment aligns with NRC accepted guidance.

d) If in response to parts (c) above, the current treatment of the MCB and those cabinets on the rear side of the MCB cannot be justified using NRC accepted guidance, then justify that the treatment has no impact on the RICT calculations. Alternatively, propose a mechanism that ensures that the FPRA is updated to treat the MCB enclosure consistent with the guidance in FAQ 14-0008, prior to implementation of the RICT program.

APLB Question 08 – FPRA Methods for Outdated FPRA and Peer Review

The LAR states in part, the Internal FPRA model was developed consistent with NUREG/CR-6850 and only utilizes NRC approved methods. As part of the ongoing PRA maintenance and update process described in the LAR, the licensee will address Internal FPRA methods approved by the NRC since the development of the Internal FPRA. Furthermore, in the LAR, the licensee specifies that a full-scope FPRA model peer review was performed in 2013.

There have been numerous changes to the FPRA methodology since the last full scope peer review of the FPRA. The integration of NRC-accepted FPRA methods and studies described below that are relevant to this submittal could potentially impact the TSTF-505 results and/or the CDF and LERF. NRC has issued updated guidance for aspects of FPRA that supplant earlier guidance issued by NRC.

- NUREG-2180, "Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities," (DELORES-VEWFIRE)," (ADAMS Accession No. ML16343A058) regarding the updated approach to credit incipient fire detections systems.
- NUREG-2169, "Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database," (ADAMS Accession No. ML15016A069) regarding changes in fire ignition frequencies and non-suppression probabilities.
- NUREG/CR-7150, "Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE)," Volume 2, "Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure," (ADAMS Accession No. ML14141A129) regarding possible increases in spurious operation probabilities.

- NUREG-2230, “Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinets Fires in Nuclear Power Plants,” (ADAMS Accession No. ML20157A148) regarding electrical cabinet fires
- NUREG-2178, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire (RACHELLE-FIRE), (ADAMS Accession No. ML20168A655) regarding heat release rates (Volume 2).

Section 2.5.5 of Regulatory Guide (RG) 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” provides guidance that indicates additional analysis is necessary to ensure that contributions from the above influences would not change the conclusions of the LAR.

a) Provide a detailed justification for why the integration of the above NRC accepted FPRA methods and studies would not significantly impact the RICT calculation. As part of this justification, identify potential FPRA methodologies used in the FPRA that are no longer accepted by the NRC staff. Provide technical justification for methods in Fermi 2 FPRA not accepted by the staff and evaluate the significance of their use on the RICT estimates.

OR

b) Alternatively, if the above guidance has been implemented in Fermi 2 FPRA, provide the following:

   i. Indicate whether the changes to the FPRA are PRA maintenance or a PRA upgrade as defined in ASME/ANS RA-Sa-2009, Section 1-5.4, as qualified by RG 1.200, Revision 2, “Acceptability of Probabilistic Risk Assessment Results for Risk-Informed Activities,” ADAMS Accession No. ML090410014) along with justification for the determination.

   iii. Discuss the focused scope (or full scope) peer review(s) that was performed to evaluate the changes that were determined in Part b.i. above to constitute a PRA upgrade and provide the date for when the peer review(s) was performed and for when the peer review report(s) that evaluated the incorporation of the method(s) was approved.

PRA Licensing Branch C (APLC) Fermi Unit 2 TSTF-505 Audit Questions

APLC QUESTION 01 – Frazil Ice

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 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 RICT." The NRC staff's safety evaluation for NEI 06-09 (ML071200238) states that "where 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 6 of enclosure 4 of the LAR, the licensee describes the process in identifying and evaluating other external hazards for screening. Table E4-4 of this enclosure lists the hazards evaluated and their screening disposition and explanation. The NRC staff notes that the frazil ice hazard has been noted as a significant hazard for nuclear power plants in the north. However, it does not appear Enclosure 4 addresses frazil ice. Address the following:

a) Clarify if frazil ice accumulation at Fermi 2 is a recognized external hazard for the plant. Include in this discussion any analysis of frazil ice impact on Fermi 2 plant operations.

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

c) If it cannot be justified that the frazil ice hazard can be screened for impact on the RICT calculations, then explain how the RICT program will account for the impact of the frazil ice hazard during a RICT implementation.

APLC QUESTION 02 – Evaluation of Seismic-Induced Loss of Offsite Power

Section 2.3.1, Item 7, of NEI 06-09-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 "performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT." The NRC staff's safety evaluation for NEI 06-09 states that "where 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 3.8 of enclosure 4 of the LAR, the licensee discusses the incremental risk associated with a seismic-induced loss of offsite power (LOOP) that may occur following a design basis seismic event. The licensee concludes that the 24-hour non-recovered seismic-induced LOOP frequency is 0.2 percent of the 24-hour non-recovered LOOP frequency determined by the Fermi FPIE PRA model. However, the LAR does not provide the value for either of the two frequencies.

Provide the value for the frequency of the 24-hour non-recovered seismic-induced LOOP and 24-hour non-recovered LOOP.

APLC QUESTION 03 – Seismic Probabilistic Risk Assessment (SPRA)

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 “performing a reasonable bounding analysis and applying it along with the internal events risk contribution in calculating the configuration risk and the associated RICT.” The NRC staff’s safety evaluation for NEI 06-09 states that “where 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 3.1 of enclosure 4 of the LAR, the licensee states that “since the Fermi 2 seismic PRA (SPRA) may not be employed in the TSTF-505 program, an alternative approach is taken here to provide an estimate of SCDF for use in TSTF-505 risk calculations.” The NRC staff found no record of a SPRA for Fermi 2, but the sentence in Enclosure 4 of the LAR, Section 3.1 seems to state that Fermi 2 does have a SPRA. Address the following:

a) Clarify if Fermi 2 has an SPRA.

b) If Fermi 2 has an SPRA, explain why it may not be employed in the TSTF-505 program.

c) If Fermi 2 has an SPRA, demonstrate that the proposed alternative seismic approach is conservative or bounding for all configurations as compared to the SPRA.

Electrical Engineering Branch (EEEB) Fermi Unit 2 TSTF-505 Audit Questions

EEEB Question 1– TS LCO 3.8.1, Conditions A and B

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. LCO 3.8.1, Conditions A and B are for the inoperability of one and two emergency diesel generators (EDGs) in one division, respectively.

According to UFSAR Section 8.3.1.1.2, "Busing Arrangements", there are two redundant engineered safety feature (ESF) divisions supplied by a total of four 4.16 kV buses with two buses assigned to each division. Each division has two load groups with its A.C. loads divided amongst those two load groups. Each load group is supplied, when necessary, by its own EDG. To supply one division of ESF loads both EDGs for that division must be operational. According to UFSAR Section 8.3.1.2.2.1, "Safety Design Basis", a single failure that could cause loss of EDGs for one ESF division would not prevent safe reactor shutdown. If a design basis accident (DBA) happened at this moment, then the plant could be safely shutdown by EDGs for one ESF division.

The Design Success Criteria (DSC) in LAR Table E1-1 for TS LCO 3.8.1 Conditions A and B presently requires "two of four EDGs" when it should clearly state to require at least "two EDGs for one division". Please clarify or explain this inconsistency.

EEEB Question 2– TS LCO 3.8.1, Condition F

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. LCO 3.8.1, Condition F is the inoperability of "One offsite circuit inoperable" AND "One or both EDGs in one division."

According to UFSAR Section 8.3.1.1.2, "Busing Arrangements", there are two redundant ESF divisions supplied by a total of four 4.16 kV buses with two buses assigned to each division. Each division has two load groups with its A.C. loads divided amongst those two load groups. Each load group is supplied, when necessary, by its own EDG. To supply one division of ESF loads both EDGs for that division must be operational. According to UFSAR Section 8.3.1.2.2.1, "Safety Design Basis", a single failure that could cause loss of EDGs for one ESF division would not prevent safe reactor shutdown. If a DBA happened at this moment, then the plant could be safely shutdown on EDGs for one ESF division.

The DSC in LAR Table E1-1 for TS LCO 3.8.1 Condition F presently requires two qualified offsite circuits, but it should clearly state to require remaining "one offsite circuit" and "two EDGs in one division." Please clarify or explain this inconsistency.

EEEB Question 3 – TS LCO 3.8.1, Conditions D and E

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided. LCO 3.8.1, Conditions D and E are for the inoperability of one and two qualified offsite circuits, respectively.

According to UFSAR Section 8.2.1.3, "Offsite Power Supply to the Plant from the Switchyards", 345/4.16/4.16-kV Transformer SS65's one secondary winding supplies power to the recirculation pump motor-generator sets. The other winding supplies approximately half of the balance-of-plant (BOP) loads and all the Division II Class 1E loads. Transformer 13.2/4.16-kV

SS64 supplies Division I Class 1E power requirements and the remainder of BOP loads. The transformer SS64 receives its power 120/13-2kV Transformer 1 via 13.2 kV-bus 1-2B. An alternate feed to bus 1-2B is provided by 120/13.8/13.8-kV three winding transformer CTG-11 which is a step-up transformer for the four-gas turbine peaking units located near Fermi 1. TS Bases, B3.8.1, "AC Sources – Operating, BACKGROUND" indicates the Class 1E AC distribution system is divided into Division I and Division II so loss of one division does not prevent the minimum safety functions from being performed. Each division is connected to an offsite power supply and two EDGs. An offsite circuit is defined as consisting of all breakers, transformers, switches, interrupting devices, cabling, and controls required to transmit power from the offsite transmission network to the onsite Class 1E ESF bus or buses. According to UFSAR Section 8.3.1.2.2.1, "Safety Design Basis", a single failure that could cause the loss of a division would not prevent safe reactor shutdown.

The DSC in LAR Table E1-1 for TS LCO 3.8.1 Conditions D and E presently requires "Two qualified circuits between the Offsite transmission network and the onsite Class 1E Distribution System" when only a single offsite circuit to one division is required for safe shutdown. Please clarify or explain this inconsistency.

EEEB Question 4 – LAR Table E1-2

GDC 17 requires, in part, that both offsite and onsite electrical power systems be provided.

According to UFSAR Section 8.2.1.3, "Offsite Power Supply to the Plant from the Switchyards", there are two offsite power circuits one for each ESF division. Also, according to UFSAR Section 8.3.1.1.2, "Busing Arrangements", there are two redundant ESF divisions supplied by a total of four 4.16 kV buses with two buses assigned to each division. Each bus supplies a load group and is powered by a single EDG.

LAR Table E1-2 indicates a RICT of 30 days for loss of one or both EDGs in one division, and a RICT of 3.9 days for loss of one offsite circuit. Please clarify or explain why the loss of one EDG or both EDGs is more risk tolerant than loss of one offsite circuit.

Technical Specifications Branch (STSB) Unit 2 TSTF-505 Audit Questions

STSB Question 1

In attachment 2 of the LAR, Condition B of TS Example 1.3-8 is proposed as follows:

CONDITION	REQUIRED ACTION	COMPLETION TIME
B. Required Action and associated Completion Time not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 5.	6 hours 36 hours

NRC staff recognizes that the licensee's proposed change is consistent with the NUREG-1433 TS markups in TSTF-505, Revision 2. However, it has been brought to staff's attention that some of the TSTF-505 markups contain errors. The proposed completion time for required action B.1 and proposed required action B.2 are not consistent with BWR STS and Fermi 2's other TS examples in this section. Discuss why these values are appropriate or provide a revision.

STSB Question 2

In Attachment 2 of the LAR, the proposed change to add a risk informed completion time (RICT) to Fermi 2 TS required action 3.3.5.1.E.2 (Place channel in trip) is, in part:

> 96 hours from discovery of inoperable channel concurrent with [high pressure coolant injection] HPCI or reactor core isolation cooling (RCIC) inoperable
>
> <u>*OR*</u>
>
> *In accordance with the Risk Informed Completion Time Program*
> …

NRC staff recognizes that the licensee's proposed change is consistent with the NUREG-1433 TS markups in TSTF-505, Revision 2. However, it has been brought to staff's attention that some of the TSTF-505 markups contain errors, introducing potential for licensee actions to be less conservative than the original intent of the requirements. To modify completion times that include the phrase "from discovery," the RICT shall start at discovery instead of the time the TS Action statement is entered, or the normal "time zero." This requirement is not clear when the RICT statement is separated from the "from discovery" statement. This is concept is also discussed in the description section of Fermi 2 TS 1.3, "Completion Times."

To provide clarity, discuss how the proposed change ensures that the time of entry for the condition will be from discovery or revise the placement of the proposed RICT for TS Required Action 3.3.5.1.E.2 between "96 hours" and "from discovery." Also, provide a similar discussion or change for proposed change for required action 3.3.5.1.F.2, which is formatted similarly.

STSB Question 3

In the LAR, the licensee proposed to add a RICT to TS required action 3.6.1.3.C.2 (verify the affected penetration flow path is isolated.) The associated frontstop completion time "once per 31 days" is not fixed like those allowed in TSTF-505, Revision 2; it is a periodic performance of an action. Provide justification for this variation.

STSB Question 4

Current Fermi 2 TS required action 3.6.1.3.D.2 contains a "once per 31 days" completion time. Discuss whether it would be appropriate to add the "following isolation" phrase similar to the other proposed additions to required actions 3.6.1.3.A.2 and 3.6.1.3.C.2 per TSTF-505 rev. 2.

STSB Question 5

TS required action 3.8.1.A.6 (restore emergency diesel generator (EDG) to operable status) would add a RICT to a completion time that is allowed, in part, due to the availability of an alternate power source as a compensatory measure. Adding a RICT to required action A.6 includes consideration of how associated requirements would be affected. Provide a discussion of how required action A.6 would interact with these TS 3.8.1 required actions:

- A.3 (verify the status of CTG 11-1),
- A.5 (restore availability of CTG 11-1),
- B.4 (restore one EDG in the division to operable status), and
- F.2 (restore both EDGS in the division to operable status).

STSB Question 6

In attachment 5 to the LAR, the cross-reference table describes what Fermi 2 TS 3.5.1 condition B (one low pressure coolant injection (LPCI) pump in both LPCI subsystems inoperable) is equivalent to in TSTF-505, rev. 2. The table indicates that it is equivalent to standard TS 3.5.1.B which states, "Required Action and associated Completion Time of Condition A not met." Discuss further why this is a similar requirement to Fermi 2 TS 3.5.1.B or provide a revised table entry.

STSB Question 7

Table E1-1 of Enclosure 1 of the LAR includes descriptions of the design success criteria (DSC) of TS proposed for inclusion in the RICT program. Given the inoperable equipment in the TS condition, the DSC should represent the minimum set of remaining credited equipment sufficient to perform TS safety function while in a RICT. According to Table E1-1, the following TS conditions appear to include a loss of function; please clarify the associated DSCs:

a) TS 3.5.1.E (HPCI System inoperable)
b) TS 3.5.3.A (RCIC System inoperable)