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

January 26, 2015

Mr. Vito Kaminskas
Site Vice President - Nuclear Generation
DTE Electric Company
Fermi 2 - 280 OBA
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE FERMI 2 LICENSE RENEWAL APPLICATION – SET 19 (TAC NO. MF4222)

Dear Mr. Kaminskas:

By letter dated April 24, 2014, DTE Electric Company (DTE or the applicant) submitted an application pursuant to Title10 of the *Code of Federal Regulations* Part 54, to renew the operating license NPF-43 for Fermi 2, for review by the U.S. Nuclear Regulatory Commission (NRC or the staff). The staff is reviewing the information contained in the license renewal application and has identified, in the enclosure, areas where additional information is needed to complete the review.

These requests for additional information were discussed with Ms. Lynne Goodman, and a mutually agreeable date for the response is within 40 days from the date of this letter. If you have any questions, please contact me at 301-415-3301 or e-mail Daneira.Melendez-Colon@nrc.gov.

Sincerely,

***/RA/***

Daneira Meléndez-Colón, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosure:
Requests for Additional Information

cc w/encl: ListServ

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**ADAMS Accession No.: ML15015A194** ***Concurred via e-mail**

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SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE FERMI 2, LICENSE RENEWAL APPLICATION – SET 19 (TAC NO. MF4222)

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**FERMI 2**
**LICENSE RENEWAL APPLICATION**
**REQUESTS FOR ADDITIONAL INFORMATION SET 19**
**(TAC NO. MF4222)**

**RAI 4.3.1.2-1**

Background:

License Renewal Application (LRA) Section 4.3.1.2 describes the applicant's time-limited aging analysis (TLAA) evaluation for the feedwater nozzle fracture mechanics analysis. The LRA states that transient numbers 10 and 11 in LRA Table 4.3-1 reflect the scram transients that are considered in the analysis. As stated in LRA Table 4.3-1, transient number 10 is for "SCRAM-turbine generator trip," and transient number 11 is for "SCRAM-all others."

Issue:

The existing fracture mechanics analysis is based on General Electric Report GE-NE-523-22-0292, "Updated NUREG-0619 Feedwater Nozzle Fatigue Crack Growth Analysis, Enrico Fermi Nuclear Power Plant, Unit 2," which was submitted to the NRC by letter dated July 29, 1992; and Design Calculation 5922, "NUREG 0619 RPV Feedwater Nozzle Crack Growth Reevaluation," which was submitted to the NRC by letter dated June 24, 1998. In both of these documents, power reductions were counted as scram transients for the fracture mechanics analysis. In particular, Design Calculation 5922 states that power reductions to below 50 percent power were counted as scrams. It is not clear from the information in LRA Table 4.3-1 as to whether the projections for transient numbers 10 and 11 include these power reductions.

Request:

Indicate whether power reductions to below 50 percent power are included in transient numbers 10 and 11 in LRA Table 4.3-1 as reflected in the above documents, along with any associated changes to your TLAA evaluation. If such power reductions are not included in these transients, provide justification as to why they are not accounted for in the transient projections for this TLAA.

**RAI 4.3.1.2-2**

Background:

LRA Section 4.3.1.2 describes the applicant's TLAA evaluation for the feedwater nozzle fracture mechanics analysis. The LRA projects that the 620 transients assumed in the existing analysis will not be exceeded during the period of extended operation. As such, the applicant dispositioned this TLAA in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 54.21(c)(1)(i) to demonstrate that the existing analysis remains valid for the period of extended operation.

Issue:

The existing fracture mechanics analysis is based on General Electric Report GE-NE-523-22-0292, “Updated NUREG-0619 Feedwater Nozzle Fatigue Crack Growth Analysis, Enrico Fermi Nuclear Power Plant, Unit 2,” and Design Calculation 5922, “NUREG 0619 RPV Feedwater Nozzle Crack Growth Reevaluation.” The calculations in GE-NE-523-22-0292 are based on a total of 648 transients. However, the results of these calculations do not meet the acceptance standard in Generic Letter (GL) 81-11, “BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking (NUREG-0619),” dated February 29, 1981. That is, GE-NE-523-22-0292 does not demonstrate that a postulated 0.25-inch flaw would grow to less than 1 inch in 40 years. The calculations in Design Calculation 5922 meet the acceptance standard in GL 81-11; however, these calculations are based on a total of 496 transients. Since neither of these existing calculations is based on a total of 620 transients, it is not clear as to why the transient projections for this TLAA were compared against this value to demonstrate that the existing analysis remains valid for the period of extended operation.

Request:

Show how the existing fracture mechanics analysis is based on a total of 620 transients. Alternatively, demonstrate that the projections for startup and shutdown transients plus scram transients, inclusive of power reductions to below 50 percent power, are less than the 496 transients that were used in Design Calculation 5922. Otherwise, provide and justify a different demonstration for this TLAA pursuant to 10 CFR 54.21(c)(1).

**RAI 4.2.4-1**

Background:

LRA Section 4.2.4 describes the TLAA for upper-shelf energy of the reactor pressure vessel beltline materials. The LRA states that the upper-shelf energy values for the beltline materials were evaluated for the period of extended operation using the guidance in NRC Regulatory Guide (RG) 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” dated May 1988.

Issue:

LRA Table 4.2-4 provides values for the various parameters used in the revised upper-shelf energy analysis. However, this table appears to be incomplete because it does not provide the un-irradiated upper-shelf energy value or the percent decrease in upper-shelf energy value for the N-16 water level instrumentation nozzle. In addition, it is not clear from the information in this table as to what methodology was used to establish the un-irradiated upper-shelf energy value and calculate the percent decrease in upper-shelf energy value for this component.

Request:

(a) Provide the un-irradiated upper-shelf energy value for the N-16 water level instrumentation nozzle and identify and justify the methodology that was used to establish this value.

(b) Provide the percent decrease in upper-shelf energy that is projected for the N-16 water level instrumentation nozzle and identify and justify the methodology that was used to calculate this value.

Alternatively, provide an explanation as to why LRA Table 4.2-4 omits the un-irradiated upper-shelf energy and percent decrease in upper-shelf energy values for the N-16 water level instrumentation nozzle.

**RAI 4.2.4-2**

Background:

LRA Section 4.2.4 describes the TLAA for upper-shelf energy of the reactor pressure vessel beltline materials. LRA Table 4.2-4 indicates that Regulatory Position 2.2 from RG 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials," was used to determine the percent decrease in upper-shelf energy values for the vertical and girth weld materials at Fermi 2 based on data from the boiling water reactor integrated surveillance program (ISP). LRA Table 4.2-5 provides fluence and upper-shelf energy values from the ISP for plate and weld materials.

Issue:

The LRA does not provide the relevant ISP data that is representative of the Fermi 2 reactor pressure vessel vertical and girth welds identified in LRA Table 4.2-4.

Request:

(a) Provide all host reactor capsule testing data from the ISP that apply to the upper-shelf energy analyses for the vertical and girth weld materials in the Fermi 2 reactor pressure vessel. Specifically, for each host reactor capsule, provide: (a) the inside diameter and ¼ T fluence values for all irradiated capsules, (b) the specific material heats that apply to the Fermi 2 vertical and girth weld materials, and (c) for those specific material heats, the un-irradiated upper-shelf energy values, measured upper-shelf energy values, and the copper, nickel, phosphorous, and silicon values. This response may be accomplished by appropriate references to documents that are available in the NRC's Agencywide Documents Access and Management System (i.e., ADAMS).

(b) Explain how the ISP data were applied to the upper-shelf energy calculations for the Fermi 2 vertical and girth weld materials. As part of this response, indicate whether there is a direct match between the material heats in the host reactors and those material heats that were used to fabricate vertical welds 2-307 A, B, and C; vertical welds 15-308 A, B, C, and D; and girth weld 1-313 at Fermi 2. If there is not a direct match, justify the basis for applying the ISP data to the upper-shelf energy calculations for these welds.

**RAI 4.2.4-3**

Background:

LRA Section A.2.1.4 provides the Updated Final Safety Analysis Report (UFSAR) supplement summarizing the TLAA for upper-shelf energy of the reactor pressure vessel beltline materials.

Issue:

The proposed description for this TLAA does not compare the results of the revised upper-shelf energy analysis against the upper-shelf energy requirements in 10 CFR Part 50, Appendix G. As such, the description does not contain adequate information regarding the basis for the demonstration for this TLAA made pursuant to 10 CFR 54.21(c)(1)(ii).

Request:

Provide justification as to why the results of the TLAA are not included in the UFSAR supplement. Otherwise, revise LRA Section A.2.1.4, as appropriate, to include a comparison of the results of the revised upper-shelf energy analysis against the upper-shelf energy requirements in 10 CFR Part 50, Appendix G.