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

February 4, 2014

Mr. Joseph H. Plona
Senior Vice President
and Chief Nuclear Officer
DTE Electric Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: RELOCATION OF PRESSURE AND TEMPERATURE CURVES TO A PRESSURE TEMPERATURE LIMITS REPORT (TAC NO. MF0446)

Dear Mr. Plona:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated December 21, 2012, as supplemented by letters dated July 9, 2013, and October 17, 2013.

The amendment revises the Fermi 2 Technical Specification (TS) Section 1.1, Definitions, TS Section 3.4.10, [Reactor Coolant System] Pressure and Temperature (P/T) Limits, and TS Section 5.6, Reporting Requirements, by replacing the existing reactor vessel heatup and cooldown rates limits and the P/T limit curves with references to the Pressure and Temperature Limits Report (PTLR) at Fermi 2. P/T limit curves and supporting information representing operation to 24 and 32 effective full power years were included.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures:
1. Amendment No. 195 to NPF-43
2. Safety Evaluation

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

DTE ELECTRIC COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 195
License No. NPF-43

1. The U.S. Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Detroit Edison Company[1] (the licensee) dated December 21, 2012, as supplemented by letters dated July 9, 2013, and October 17, 2013, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

[1] Note: The Detroit Edison Company name was changed to DTE Electric Company as of January 1, 2013.

Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 195, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DTE Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented within 60 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert D. Carlson, Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the License and Technical Specifications

Date of Issuance: February 4, 2014

ATTACHMENT TO LICENSE AMENDMENT NO. 195

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Facility Operating License and Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
License Page 3	License Page 3
1.1-5	1.1-5
3.4.23	3.4-23
3.4-24	3.4-24
3.4.25	3.4-25
3.4-26	3.4-26
3.4-27	3.4-27
3.4-28	3.4-28
3.4-28a	---
3.4.28b	---
3.4-28c	---
3.4-28d	---
3.4-28e	---
5.0-22	5.0-22

(4) DTE Electric Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material such as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) DTE Electric Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) DTE Electric Company, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

DTE Electric Company is authorized to operate the facility at reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 195 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated into this license. DTE Electric Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DTE Electric Company shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between DTE Electric Company and

1.1 Definitions (continued)

MINIMUM CRITICAL POWER RATIO (MCPR)	The MCPR shall be the smallest critical power ratio (CPR) that exists in the core for each type of fuel. The CPR is that power in the assembly that is calculated by application of the appropriate correlation(s) to cause some point in the assembly to experience boiling transition, divided by the actual assembly operating power.
MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE-OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the UFSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR)	The PTLR is the unit-specific document that provides the reactor vessel pressure and temperature limits, including heatup and cooldown rates, for the current reactor vessel fluence period. These pressure and temperature limits shall be determined for each fluence period in accordance with TS 5.6.8.

(continued)

3.4 REACTOR COOLANT SYSTEM (RCS

3.4.10 RCS Pressure and Temperature (P/T) Limits

LCO 3.4.10 RCS pressure, RCS temperature, RCS heatup and cooldown rates, and the recirculation pump starting temperature requirements shall be maintained within the limits specified in the PTLR.

APPLICABILITY: At all times.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. ---------NOTE--------- Required Action A.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in MODES 1, 2, and 3.	A.1 Restore parameter(s) to within limits. AND A.2 Determine RCS is acceptable for continued operation.	30 minutes 72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1 Be in MODE 3. AND B.2 Be in MODE 4.	12 hours 36 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. ---------NOTE--------- Required Action C.2 shall be completed if this Condition is entered. --------------------- Requirements of the LCO not met in other than MODES 1, 2, and 3.	C.1	Initiate action to restore parameter(s) to within limits.	Immediately
	<u>AND</u>		
	C.2	Determine RCS is acceptable for operation.	Prior to entering MODE 2 or 3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.10.1	------------------NOTE------------------- Only required to be performed as applicable during RCS heatup and cooldown operations and RCS inservice leak and hydrostatic testing. --- Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR.	30 minutes

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.2	Verify RCS pressure and RCS temperature are within the criticality limits specified in the PTLR.	Once within 15 minutes prior to control rod withdrawal for the purpose of achieving criticality
SR 3.4.10.3	-------------------NOTE------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the bottom head coolant temperature and the reactor pressure vessel (RPV) steam space coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump
SR 3.4.10.4	-------------------NOTE------------------ Only required to be met in MODES 1, 2, 3, and 4 during recirculation pump startup. -- Verify the difference between the reactor coolant temperature in the recirculation loop to be started and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to each startup of a recirculation pump

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.5	------------------NOTE------------------ Only required to be met during a THERMAL POWER increase or recirculation flow increase in MODES 1 and 2 with one idle recirculation loop when THERMAL POWER is ≤ 30% RTP or when operating loop flow is ≤ 50% rated loop flow. -- Verify the difference between the bottom head coolant temperature and the RPV steam space coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to a THERMAL POWER increase or recirculation flow increase
SR 3.4.10.6	------------------NOTE------------------ Only required to be met during a THERMAL POWER increase or recirculation flow increase in MODES 1 and 2 with one non-isolated idle recirculation loop when THERMAL POWER is ≤ 30% RTP or when operating loop flow is ≤ 50% rated loop flow. -- Verify the difference between the reactor coolant temperature in the idle recirculation loop and the RPV coolant temperature is within the limits specified in the PTLR.	Once within 15 minutes prior to a THERMAL POWER increase or recirculation flow increase

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.4.10.7	------------------NOTE------------------ Only required to be performed when tensioning the reactor vessel head bolting studs. -- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR when the reactor vessel head bolt studs are under tension.	30 minutes
SR 3.4.10.8	------------------NOTE------------------ Not required to be performed until 30 minutes after RCS temperature ≤ 80°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	30 minutes
SR 3.4.10.9	------------------NOTE------------------ Not required to be performed until 12 hours after RCS temperature ≤ 100°F in MODE 4. -- Verify reactor vessel flange and head flange temperatures are within the limits specified in the PTLR.	12 hours

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5.6 Reporting Requirements (continued)

5.6.6 Safety Relief Valve Challenge Report

The main steam line Safety Relief Valve (SRV) Report documenting all challenges to SRVs during the previous calendar year shall be submitted by April 30 of each year.

5.6.7 PAM Report

When a report is required by Condition B or F of LCO 3.3.3.1, "Post Accident Monitoring (PAM) Instrumentation," a report shall be submitted within the following 14 days. The report shall outline the action taken, the cause of the inoperability, and the plans and schedule for restoring the instrumentation channels of the Function to OPERABLE status.

5.6.8 Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)

a. RCS pressure and temperature limits for heatup, cooldown, low temperature operation, criticality, and inservice leakage and hydrostatic testing as well as heatup and cooldown rates shall be established and documented in the PTLR for the following:

   1. Limiting Condition for Operation Section 3.4.10, "RCS Pressure and Temperature (P/T) Limits."

   2. Surveillance Requirement Section 3.4.10, "RCS Pressure and Temperature (P/T) Limits."

b. The analytical methods used to determine the RCS pressure and temperature limits shall be those previously reviewed and approved by the NRC, specifically those described in the following document:

   1. NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, June 2009.

c. The PTLR shall be provided to the NRC upon issuance for each reactor vessel fluence period and for any revision or supplement thereto.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 195 TO FACILITY OPERATING LICENSE NO. NPF-43

DTE ELECTRIC COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated December 21, 2012 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML13004A134), as supplemented by letters dated July 9 (ADAMS Accession No. ML13191A748), and October 17, 2013 (ADAMS Accession No. ML13291A363), Detroit Edison Company[2] (the licensee) submitted a request for changes to the Fermi 2 Technical Specifications (TSs) related to the Pressure-Temperature (P/T) limits. The supplements dated July 9 and October 17, 2013, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the U.S. Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination noticed in the *Federal Register* on April 9, 2013 (78 FR 21167).

The proposed TS changes would relocate the Fermi 2 P/T limits from the TS Limiting Conditions for Operation (LCOs) to a licensee-controlled Pressure and Temperature Limits Report (PTLR). The licensee's submittal includes a proposed PTLR that incorporates new P/T limit curves representing facility operation through 24 and 32 effective full power years (EFPY). The PTLR was developed based on the NRC-approved methodology of proprietary General Electric-Hitachi (GEH) Topical Report NEDC-33178P-A, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves," Revision 1, dated June 2009, as referenced in the proposed TS revision.

The NRC staff reviewed the licensee's submittal to determine: (1) whether the proposed TS changes meet the criteria for implementation of a PTLR, as defined in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," and NRC-approved Technical Specifications Task Force (TSTF)-419-A, "Revise PTLR Definition and References in Improved Standard Technical Specification 5.6.6, Reactor Coolant System PTLR," (2) whether the proposed PTLR meets the seven technical criteria for an acceptable PTLR, as defined in GL 96-03, and (3) whether the proposed 24 and 32 EFPY P/T limits contained in the PTLR were correctly generated in accordance with the NRC-approved methodology of GEH Topical Report NEDC-33178-P-A,

[2] The Detroit Edison Company name was changed to DTE Electric Company as of January 1, 2013.

and are incompliance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, as required by GL 96-03 in order to implement the PTLR.

2.0 REGULATORY EVALUATION

The NRC has established requirements in Part 50 of Title 10 of the *Code of Federal Regulations* (10 CFR Part 50) to protect the integrity of the reactor coolant pressure boundary in nuclear power plants. The staff evaluates the acceptability of a facility's proposed P/T limits based on the following NRC regulations and guidance: Appendix G, "Fracture Toughness Requirements," to 10 CFR Part 50; Appendix H, "Reactor Vessel Material Surveillance Program Requirements," to 10 CFR Part 50; Regulatory Guide (RG) 1.99, Revision 2 (Rev. 2), "Radiation Embrittlement of Reactor Vessel Materials;" Generic Letter (GL) 92-01, Rev. 1, "Reactor Vessel Structural Integrity;" GL 92-01, Rev. 1, Supplement 1, "Reactor Vessel Structural Integrity;" and Standard Review Plan (SRP) Section 5.3.2. Appendix G to 10 CFR Part 50 requires that facility P/T limits for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the linear elastic fracture mechanics methodology of Appendix G to Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code). Appendix H to 10 CFR Part 50 establishes requirements related to facility RPV material surveillance programs. RG 1.99, Rev. 2, contains methodologies for determining the increase in transition temperature and the decrease in upper-shelf energy resulting from neutron radiation.

GL 92-01, Rev. 1 requested that licensees submit the RPV data for their plants to the NRC staff for review, and GL 92-01, Rev. 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations. SRP Section 5.3.2 provides an acceptable method for determining the P/T limits for ferritic materials in the beltline of the RPV based on the ASME Code Appendix G methodology.

The most recent version of Appendix G to Section XI of the ASME Code, which has been endorsed in 10 CFR 50.55a, and therefore by reference in 10 CFR Part 50, Appendix G, is the 2010 Edition of the ASME Code. This edition of Appendix G to Section XI of the ASME Code incorporates the provisions of ASME Code Case N-588, "Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels," and ASME Code Case N-640, "Alternative Reference Fracture Toughness for Development of P/T Limit Curves." Additionally, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is at or above 20 percent of the preservice hydrostatic test pressure.

GL 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," establishes the information that must be included in an acceptable PTLR methodology and in an acceptable PTLR. The PTLR must also comply with Technical Specification Task Force (TSTF) 419-A, which documents revised guidance for a plant's PTLR. Subsequent changes in the methodology must be approved by the NRC by a license amendment; 10 CFR 50.59 does not apply.

In 10 CFR 50.36, "Technical specifications," the NRC established its regulatory requirements related to the content of TSs. Pursuant to 10 CFR 50.36, TSs are required to include items in the following five specific categories related to station operation: (1) safety limits, limiting safety system settings, and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. The amendment

revises the Fermi 2 TS Section 1.1, Definitions, TS Section 3.4.10, [Reactor Coolant System] Pressure and Temperature (P/T) Limits, and TS Section 5.6, Reporting Requirements, by replacing the existing reactor vessel heatup and cooldown rates limits and the P/T limit curves with references to the PTLR at Fermi 2.

10 CFR Section 50.60 imposes fracture toughness and material surveillance program requirements, which are set forth in 10 CFR 50 Appendices G and H.

10 CFR 50, Appendix G establishes fracture toughness requirements. In the Definitions section of Appendix G, paragraph G.II D(ii) states, "For the reactor vessel beltline materials, RT_{NDT} must account for the effects of neutron radiation." In the Requirements section, paragraph G.IV.A states in part, "...the values of RT_{NDT} and Charpy upper-shelf energy must account for the effects of neutron radiation, including the results of the surveillance program of appendix H of this part." The effects of neutron radiation are determined, in part, by estimating the neutron fluence on the reactor vessel.

RG 1.190 describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence with respect to the General Design Criteria contained in Appendix A of 10 CFR Part 50. In consideration of the guidance set forth in RG 1.190, GDC 14, 30, and 31 are applicable. GDC 14, "Reactor Coolant Pressure Boundary," requires the design, fabrication, erection, and testing of the reactor coolant pressure boundary so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 30, "Quality of Reactor Coolant Pressure Boundary," requires among other things, that components comprising the reactor coolant pressure boundary be designed, fabricated, erected, and tested to the highest quality standards practical. GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," pertains to the design of the reactor coolant pressure boundary, stating:

> The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions, (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

3.0 TECHNICAL EVALUATION

3.1 Licensee's Evaluation

The revised P/T limits are based on application of the GE methodology to Fermi 2. NEDC-33178P-A, Revision 1, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (the GE methodology) provides the approved generic GE methodology for generating P/T limits based on the plant-specific adjusted reference temperature (ART). NEDO-33178-A, Revision 1 (ADAMS Accession No. ML092370487) is the non-proprietary version of the report. The GE methodology provides

beltline and generic upper vessel and bottom head P/T limit curves that are shifted by the plant-specific ART, as well as guidance on the application of the ASME Code, Appendix G and 10 CFR Part 50, Appendix G.

For the RPV beltline material, the licensee identified weld heat 13253/12008 as the limiting beltline weld material for Fermi 2. A limiting plate material was not identified due to the differences between the Integrated Surveillance Program target material and the Fermi 2 material. The acceptability of this was addressed in that the estimated ART of the limited weld was significantly above that of the beltline plates. ART values were calculated for 24 and 32 EFPYs. The licensee noted that the N16 water level instrument (WLI) nozzle was evaluated using material properties estimated from the operating boiling water reactor (BWR) forging data, due to a lack of Fermi 2 specific data. The parameters used to determine the licensee's ART values for the limiting materials at the one-quarter of the RPV wall thickness (1/4T) location for 24 and 32 EFPY are shown in Enclosure 5 of the licensee's application. Corresponding parameters at the three-quarter of the RPV wall thickness (3/4T) were not provided in the attachments. Instead, the licensee applied the maximum tensile stress for both heatup and cooldown at the 1/4T location. The licensee stated that this approach is conservative because the 1/4T material toughness is lower than that in the 3/4T locations.

P/T limit Curves A, B, and C were provided in Enclosure 5 of the application, and are based on application of the GE methodology. The licensee noted that the P/T limit curves were limited by beltline materials for portions of the curves detailing precisely which portions these were. The licensee provided data from the Integrated Surveillance Program BWRVIP-135, "BWR Vessel and Internals Project Integrated Surveillance Program (BWRVIP ISP) Data Source Book and Plant Evaluations" in compliance with a requirement in the GE methodology. However, since the target plate material did not match the representative material, the data from BWRVIP-135 was not used. The BWRVIP-135 source book is used by the industry in compliance with BWRVIP-86, Rev. 1, "BWR Vessel and Internals Project Updated BWR Integrated Surveillance Program Implementation Plan" (ADAMS Accession No. ML090300556). Information was also included detailing the determination process for evaluating non-beltline but potentially limiting components.

3.2 <u>NRC Staff Evaluation</u>

The NRC staff reviewed the licensee's submittal to determine: (1) whether the proposed TS changes meet the criteria for implementation of a PTLR, as defined in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," and NRC-approved Technical Specifications Task Force (TSTF)-419-A, "Revise PTLR Definition and References in Improved Standard Technical Specification 5.6.6, Reactor Coolant System PTLR," (2) whether the proposed PTLR meets the seven technical criteria for an acceptable PTLR, as defined in GL 96-03, and (3) whether the proposed 24 and 32 EFPY P/T limits contained in the PTLR were correctly generated in accordance with the NRC-approved methodology of GEH Topical Report NEDC-33178-P-A, and are incompliance with the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Appendix G, as required by GL 96-03 in order to implement the PTLR.

3.2.1 PTLR Implementation

The licensee utilized the GE methodology to generate their P/T limits. As documented in the SE for topical report NEDO-33178-A, the NRC staff has approved the GE methodology for use in generating PTLRs.

As mentioned in Section 2.0 of this safety evaluation (SE), GL 96-03 requires the licensee to evaluate several technical criteria to demonstrate the acceptability of its PTLR. The NRC staff examined the proposed PTLR and determined that it was developed from the Template PTLR found in the GE methodology report and meets the seven technical criteria:

(1) The PTLR methodology describes the transport calculation methods including computer codes and formula used to calculate neutron fluences.

The submittal documents that the neutron fluence was calculated in accordance with the NRC-approved methodology NEDC-32983-A, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation" (ADAMS Accession No. ML072480121). This approved report documents the transport calculation methods including computer codes and formula, used to calculate neutron fluencies. Hence, the first criterion is met.

(2) The PTLR methodology describes the surveillance program.

The submittal documents that Fermi 2 participates in the approved BWRVIP Integrated Surveillance Program as documented in the NRC staff-approved BWRVIP Report, BWRVIP-86NP, Rev. 1-A, "Updated BWR Integrated Surveillance Program (ISP) Implementation Plan," dated May 2013 (ADAMS Accession No. ML13176A097), which meets the requirements of 10 CFR Part 50, Appendix H. Hence, the second criterion is met.

(3) The PTLR methodology describes how the low temperature overpressure protection system limits are calculated applying system/thermal hydraulics and fracture mechanics.

This is not applicable to BWRs. Therefore, this criterion does not apply to Fermi 2.

(4) The PTLR methodology describes the method for calculating the ART values using RG 1.99, Rev. 2.

The submittal indicated that RG 1.99, Rev. 2 provides the methods for determining the ARTs for the beltline materials, with their chemistry factors determined by surveillance data information from the BWRVIP ISP, as appropriate. The NRC staff confirmed that RG 1.99, Rev. 2, was used for calculating ARTs for the beltline materials. Hence, the fourth criterion is met.

(5) The PTLR methodology describes the application of fracture mechanics in the construction of P/T limits based on ASME Code, Section XI, Appendix G, and the SRP.

On page 3 of the PTLR it is stated that the P/T limits were calculated in accordance with

the GE methodology. This description is sufficient as the GE methodology report was reviewed and found to meet the fifth criterion. Hence, the fifth criterion is met.

(6) The PTLR methodology describes how the minimum temperature requirements in Appendix G to 10 CFR Part 50 are applied to P/T limits for boltup temperature and hydrotest temperature.

Again, referencing the GE methodology is sufficient because the report contains detailed information regarding the minimum temperature requirements for boltup temperature and hydrotest temperature. The NRC staff reviewed and approved the GE methodology under the sixth criterion. Hence, the sixth criterion is met.

(7) The PTLR methodology describes how the data from multiple surveillance capsules are used in the ART calculation.

Again, referencing the GE methodology is sufficient because the report contains detailed information regarding this criterion in the GE methodology Appendix I. The NRC staff reviewed and approved the GE methodology under the seventh criterion. Hence, the seventh criterion is met.

In summary, the implementation of Fermi 2 PTLR is acceptable.

3.2.2 P/T Limits

The proposed P/T limits are a composite of the RPV beltline, the bottom head, and the upper vessel curves. Independent P/T curves generated by the NRC staff are consistent with P/T curves provided by the licensee. The staff's curves were generated using the GE methodology and ASME Code, Section XI, Appendix G.

To evaluate the proposed Fermi 2 RPV beltline P/T limits the NRC staff first confirmed the licensee's selection of limiting materials. For Fermi 2 beltline materials, the staff found that the initial reference temperature (RT_{NDT}), copper (Cu), and nickel (Ni) values were largely in agreement with the information in the NRC's Reactor Vessel Integrity Database (RVID). The licensee reported best estimate chemistry and ISP data from BWRVIP-86, Rev. 1, to confirm the collection of credible chemistry and surveillance data. Best estimate chemistries from BWRVIP-86 do not significantly differ from RVID and, therefore, the inclusion of best estimate chemistry does not change the limiting beltline material previously identified by the staff. The licensee only calculated ART values for the RPV 1/4T location. The NRC staff concurs that this is reasonable as the licensee's approach of using the maximum tensile stress for either heatup or cooldown and applying it at the 1/4T location is equivalent to using the maximum thermal stress intensity factor (K_{IT}) and the minimum fracture toughness (K_{Ic}) in the heatup and cooldown analysis, making the proposed P/T limits bound both the heatup and cooldown curves.

As previously noted, the licensee made use of the GE methodology in generating P/T limits, with composite and limiting P/T limit Curves A, B, and C provided by the licensee. The composite curves reported by the licensee are consistent with composite curves generated by the NRC staff applying the GE methodology, shifting the approved generic GE bottom curves by

the ART for the limiting material identified. For Curve C, below 20 percent of the hydro test pressure (312 pounds per square inch – gauge (psig)), the NRC staff found the upper vessel curve generated using the GE methodology limiting, consistent with the composite P/T curve provided by the licensee. For all other conditions, the Appendix G to 10 CFR Part 50 requirements for the minimum metal temperature of the closure head flange and vessel flange regions produce limiting "notches," serving to explain the distinct vertical lines at constant temperature above approximately 312 psig in the licensee's proposed P/T limits. For all Fermi 2 curves, a minimum temperature of 68 °F for the bottom head and 72 °F for the flange region was verified to be ASME Code compliant with the stipulation that these regions must be at least RT_{NDT} + 60 °F (where RT_{NDT} represents that property of the limiting material in the relevant region). When P > 312 psig, the minimum temperature of 102 ºF for the pressure test curve, 132 ºF for the normal operation/core not critical curve, and 172 ºF for the normal operation/core critical curve are derived from adding 90 ºF, 120 ºF, and 160 ºF to the RT_{NDT} of 12 ºF for the limiting flange material as specified in Appendix G to 10 CFR Part 50 for the three operating conditions.

The licensee noted that the N16 nozzle, a beltline WLI nozzle, was evaluated. The NRC staff reviewed the dispositioning of this and other relevant nozzles and determined that they were adequately addressed in the implementation of the PTLR.

In its October 17, 2013, RAI response, the licensee clarified that the Fermi 2 N16 WLI nozzle is a partial penetration design nozzle and that Appendix J of the methodology was used to calculate the P/T limits for the N16 WLI nozzle. The licensee provided a sample calculation for Curve B, wherein the N16 WLI nozzle was found to be bounding between 580 psig and 1220 psig. In this calculation the licensee applied a correction factor, R, to the thermal stress for the limiting transient and then fed the resulting stress into the method of Appendix J to the methodology. Although the use of R is part of the methodology for full penetration nozzles, it is not clear that the methodology allows the R factor to be used for a partial penetration nozzle in combination with Appendix J of the methodology. However, the NRC staff determined that the thermal stress value from Appendix J of the methodology is extremely conservative compared to that which would result from the normal and upset transients required to be addressed by the ASME Code, Section XI, Appendix G, as it is derived from an emergency transient. Since the resulting applied stress intensity determined for the N16 WLI nozzle is very conservative even with the R factor, the NRC staff finds the use of the R adjustment to be acceptable.

The NRC staff also reviewed the licensee's analysis of non-beltline components and materials. The licensee documented its evaluation of this in Enclosure 5 of the application. In many plant designs the material properties of the beltline have been controlled such that geometric and non-beltline materials may in fact be the limiting factors in portions of P/T limits. The staff requested that the licensee further clarify how the P/T limit curves in the submittal bounded all RPV materials and the lowest permissible service temperatures of all ferritic reactor coolant pressure boundary (RCPB) materials. In its October 17, 2013, RAI response, the licensee provided a discussion of the GE methodology. The RAI response clarified precisely how each non-beltline portion of the RPV was analyzed and how the associated P/T limit curves were developed. This analysis resulted in three curves, one for the upper vessel, one for the vessel head, and one for the vessel bottom head. Discontinuities were addressed, as well, and found to be bounded by the aforementioned curves. The licensee also addressed RCPB piping and

welds, citing compliance with ASME Code Section XI, Appendix G, Article G-3000, paragraph G-3100:

> [For materials] used for piping, pumps, and valves for which impact tests are required (NB-2311), the tests and acceptance standards of Section III, Division 1 are considered to be adequate to prevent non-ductile failure under the loadings and with the defect sizes encountered under normal, upset, and testing conditions. Level C and Level D Service Limits should be evaluated on an individual case basis (G-2300).

Pertinently, Section 5.2.4.2.1 of the Fermi 2 updated final safety analysis report (UFSAR), Revision 18, states:

> Versions of 10 CFR 50, Appendix G, prior to the 1983 Edition had specific requirements for the preparation and testing of all reactor coolant pressure boundary materials. In lieu of these specific requirements, the present version of Appendix G requires that for a reactor vessel which was constructed in conformance with an ASME Code Section III earlier than the Summer 1972 Addenda of the 1971 Edition, the fracture toughness data and data analyses must be supplemented in a manner approved by the Director, Office of Nuclear Reactor Regulation, to demonstrate equivalence with the present fracture requirements of Appendix G. The Fermi 2 reactor vessel was constructed in compliance with an ASME Code earlier than the summer 1972 Addenda of the 1971 Edition. The NRC has stated in Supplement 1 to NUREG 0798, the Fermi 2 Safety Evaluation Report, that the alternative methods proposed by Fermi 2 to demonstrate compliance with Appendix G has been reviewed, evaluated, and found to provide the safety margin required by Appendix G. Accordingly, Fermi 2 has supplied sufficient information to demonstrate equivalency with the fracture toughness requirements of the present version of 10 CFR 50, Appendix G, (1983 as amended November 1986 and October 1988)."

In addition, the licensee cited UFSAR information related to compliance with GDC 14, 30, 31, and 32. The NRC staff reviewed this information and concluded that compliance with the Fermi 2 UFSAR would provide reasonable assurance of adequacy for non-beltline RCPB ferritic components. Consequently, the NRC staff finds that the non-beltline RPV components and ferritic RCPB materials have been adequately addressed by Fermi 2 for the purposes of implementing a PTLR.

As part of the application, the licensee also included responses to NRC staff RAIs issued for a similar review. In these responses the licensee confirmed that the hydrostatic pressure adjustment for the water column in a full RPV was included in its analysis. The licensee also provided material chemistries noted in BWRVIP-135. Also included was a detailed discussion of the Fermi 2 N16 WLI nozzle as noted above. Finally, the licensee provided details regarding plant-specific feedwater nozzle evaluation. The NRC staff reviewed this information and found it to be acceptable.

Based on the above evaluation, the NRC staff determined that the licensee's proposed P/T limits are in accordance with the GE methodology and satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50. Hence, the licensee's proposed P/T limit curves are acceptable and support the implementation of a PTLR for the Fermi 2 RPV.

3.2.3 Fluence Calculations

The NRC staff evaluated the neutron fluence calculations to determine whether they are acceptable for generating the P/T limit curves. The guidance provided in RG 1.190 indicates that the following attributes comprise an acceptable fluence calculation:

- A fluence calculation performed using an acceptable methodology
- Analytic uncertainty analysis identifying possible sources of uncertainty
- Benchmark comparison to approved results of a test facility
- Plant-specific qualification by comparison to measured fluence values

The fast neutron exposure parameters were determined for the licensee by GE-Hitachi, using the methods discussed in NEDO-32983-A Revision 2, "General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations" (ADAMS Accession No. ML072480121). As noted by the safety evaluation enclosed in the approved topical report, the NRC determined that this methodology is generically acceptable for the reference in licensing actions.

As described in NEDO-32983-A, Revision 2, the neutron fluence was calculated using the following methods. A solution to the Boltzmann transport equation is approximated using the two-dimensional discrete ordinates transport code. The licensee uses a cross-section library that the NRC staff has found generically acceptable in light of the guidance contained in RG 1.190 (refer to Section 3.1 of the safety evaluation approving NEDO-32983-A, Revision 1, enclosed in ML072480121). Approximations include a P_3 Legrendre expansion for anisotropic scattering and a S_8 order of angular quadrature (refer to Page 4 of the NRC staff SE for NEDO-32983, Rev. 2, Section 3.1, "Pressure Vessel Fluence Calculation Methodology"). These approximations are in line with the minimum P_3 expansion and S_8 quadrature suggested in RG 1.190. As described in NEDO-32983-A, Revision 2, the neutron source distribution has spatial distribution of the neutron source density assumed as proportional to the relative cycle-averaged energy production per each fuel node and each bundle location. Typical core-averaged relative power density variation for (r,θ) calculation and typical core-zone averaged variation in the axial direction for (r, z) calculation are described in NEDO-32983-A, Revision 2 for reference. The neutron calculations, as described above, are performed in a manner consistent with the guidance set forth in RG 1.190.

Also described in NEDO-32983-A, Revision 2, an analytic uncertainty analysis was performed by combining the uncertainties associated with the individual components of the transport calculations in square-root-of-the-sum-of-the squares. The calculations were compared with the benchmark measurements from the vessel fluence benchmark problems provided in NUREG/CR-6115, "PWR and BWR Pressure Vessel Fluence Calculation Benchmark Problems and Solutions," provided by Brookhaven National Laboratory (BNL). This constitutes acceptable test facilities.

Finally, NEDO-32983-A, Revision 2 is acceptably benchmarked for Fermi 2, as it contains a database of BWR dosimetry benchmarking, and Fermi 2's unit geometry (BWR/4 Reactor Vessel) is well represented within the database. Source term uncertainty came to be 13 percent for fast responses seen in NEDO-32893-A, Revision 2. Therefore, reaction rates were

calculated within 20 percent of measured values, as suggested in RG 1.190. Based on this, the NRC staff determined that these uncertainties are acceptable.

Additionally, the fluence was calculated assuming an extended power uprate (EPU) as a bounding case, even though an EPU has not yet been implemented at Fermi 2. Consequently, the fluence was calculated to be higher than it would have otherwise been. The fluence for the N16 WLI nozzle did not include EPU conditions but did include a measurement uncertainty recapture uprate. As the fluence calculations were performed in accordance with an NRC-approved methodology and using the guidance in RG 1.190, and a conservative fluence case was chosen by using EPU conditions, the NRC staff finds the fluence calculations to be acceptable insofar as they support the requested PTLR implementation.

3.2.4 Summary

Based on the NRC staff's review of the information provided in the licensee's submittals dated December 21, 2012, July 9, 2013, and October 17, 2013, the NRC staff concludes that the proposed Fermi 2 PTLR meets GL 96-03 requirements for implementation and, therefore, is approved as part of Fermi 2 licensing basis.

The Fermi 2 RPV P/T limits are based on an acceptable methodology documented in the NEDC-33178P-A report. The NRC staff performed independent evaluations and verified that the P/T limits were developed appropriately using the NEDC-33178P-A methodology, and the proposed P/T limits, valid for 24 and 32 EFPYs, satisfy the requirements of Appendix G to Section XI of the ASME Code and Appendix G to 10 CFR Part 50. The TS revision to reflect the use of this acceptable methodology is also appropriate.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Michigan State official was notified of the proposed issuance of the amendment. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes the surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (April 9, 2013 (78 FR 21167)). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) there is reasonable assurance that such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: D. Widrevitz
M. Hardgrove

Date: February 24, 2014

February 4, 2014

Mr. Joseph H. Plona
Senior Vice President
and Chief Nuclear Officer
DTE Electric Company
Fermi 2 - 210 NOC
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: RELOCATION OF PRESSURE AND TEMPERATURE CURVES TO A PRESSURE TEMPERATURE LIMITS REPORT (TAC NO. MF0446)

Dear Mr. Plona:

The U.S. Nuclear Regulatory Commission has issued the enclosed Amendment No. 195 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment consists of changes to the Technical Specifications in response to your application dated December 21, 2012, as supplemented by letters dated July 9, 2013, and October 17, 2013.

The amendment revises the Fermi 2 Technical Specification (TS) Section 1.1, Definitions, TS Section 3.4.10, [Reactor Coolant System] Pressure and Temperature (P/T) Limits, and TS Section 5.6, Reporting Requirements, by replacing the existing reactor vessel heatup and cooldown rates limits and the pressure and temperature (P/T) limit curves with references to the Pressure and Temperature Limits Report (PTLR) at Fermi 2. P/T limit curves and supporting information representing operation to 24 and 32 effective full power years were included.

A copy of our safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,
/RA/

Thomas J. Wengert, Senior Project Manager
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-341

Enclosures:
1. Amendment No. 195 to NPF-43
2. Safety Evaluation

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