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DTE Energy

10 CFR 50.90

December 21, 2012
NRC-12-0037

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington D C 20555-0001

Reference: Fermi 2
NRC Docket No. 50-341
NRC License No. NPF-43

Subject: Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

In accordance with 10 CFR 50.90, Detroit Edison proposes to amend the Fermi 2 Plant Operating License, Technical Specification (TS) Section 1.1, "Definitions", Section 3.4.10 "RCS Pressure and Temperature (P/T) Limits", and Section 5.6, "Reporting Requirements" by replacing the existing reactor vessel heatup and cooldown rate limits and the pressure and temperature (P/T) limit curves with references to the Pressure and Temperature Limits Report (PTLR) at Fermi 2.

The conditions of U.S. Nuclear Regulatory Commission (NRC) Generic Letter 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," were applied during development of the proposed P/T curves. The TS changes were developed in accordance with the NRC approved guidance of Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in Improved Standard Technical Specification 5.6.6, Reactor Coolant System PTLR." The analytical methodology used in development of the proposed P/T curves and the PTLR, is the NRC approved, General Electric-Hitachi (GEH) Boiling Water Reactor Owners Group (BWROG) Licensing Topical Report NEDC-33178P-A, Revision 1, "General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves."

Enclosure 6 contains Proprietary Information – Withhold Under 10 CFR 2.390.
When separated from Enclosure 6, this document is decontrolled.

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NRR

The effect of lower water level instrumentation (WLI) nozzle material properties was not properly considered by GEH in the development of the currently licensed P/T curves. As discussed in Enclosure 1, this issue is being addressed in accordance with NRC Administrative Letter 98-10. The PTLR incorporates the corrections needed to address the WLI nozzle material properties not previously considered.

Enclosure 1 provides a description of the proposed changes and includes the technical evaluation and associated no significant hazards determination and environmental evaluation. Enclosure 2 provides a marked-up copy of the TS pages showing the proposed changes. Enclosure 3 provides the revised TS pages. Enclosure 4 provides a marked-up copy of the TS Bases pages, for information only, indicating the proposed changes. Enclosure 5 provides the Fermi 2 PTLR report. Enclosure 6 contains proprietary responses to requests for additional information provided to Grand Gulf Nuclear Station regarding a similar request, customized for Fermi 2. This enclosure includes affidavits from GEH and the Electric Power Research Institute (EPRI) requesting withholding the proprietary information from public disclosure. Enclosure 7 is a non-proprietary version of Enclosure 6.

Detroit Edison has reviewed the proposed change against the criteria of 10 CFR 51.22 and has concluded that it meets the criteria provided in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

Approval of this license amendment is requested by November **XX**, 2013 with a 60 day implementation period.

No new commitments are being made in this submittal.

In accordance with 10 CFR 50.91, a copy of this application, with attachments, is being provided to the designated Michigan State Official.

Should you have any questions or require additional information, please contact Mr. Zackary W. Rad of my staff at (734) 586-5076.

Sincerely,

Enclosures:

1. Evaluation of Proposed License Amendment
2. Markup of Existing TS Pages
3. Revised TS Pages
4. Markup of Existing TS Bases Pages (For Information Only)
5. Fermi 2 Pressure Temperature Limits Report
6. Responses to Requests for Additional Information – Proprietary
7. Responses to Requests for Additional Information – Non-Proprietary

cc: NRC Project Manager
NRC Resident Office
Reactor Projects Chief, Branch 4, Region III
Regional Administrator, Region III
Supervisor, Electric Operators,
Michigan Public Service Commission

I, J. Todd Conner, do hereby affirm that the foregoing statements are based on facts and circumstances which are true and accurate to the best of my knowledge and belief.

J. Todd Conner
Site Vice President, Nuclear Generation

On this 21 day of December, 2012 before me personally appeared J. Todd Conner, being first duly sworn and says that he executed the foregoing as his free act and deed.

Sharon S. Marshall
Notary Public

SHARON S. MARSHALL
NOTARY PUBLIC, STATE OF MI
COUNTY OF MONROE
MY COMMISSION EXPIRES Jun 14, 2013
ACTING IN COUNTY OF Monroe

Enclosure 1 to
NRC-12-0037

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

Evaluation of Proposed License Amendment

Evaluation of the Proposed License Amendment

1.0 Description

The proposed amendment would modify Fermi 2 Plant Operating License, Appendix A, Technical Specifications (TS) by replacing the pressure and temperature (P/T) limit curves with references to the Pressure and Temperature Limits Report (PTLR). Relocation of the P/T limit curves to the PTLR is consistent with the guidance provided in U.S. Nuclear Regulatory Commission (NRC) approved GE Hitachi (GEH) Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, "GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves" (Reference 1). This topical report uses the guidelines provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits" (Reference 2). Additionally, the TS changes in this license amendment request are consistent with the guidance provided in GL 96-03 as supplemented by Technical Specification Task Force (TSTF) traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 3), and the guidance contained in the August 4, 2011 NRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6, "Reporting Requirements", as discussed below.

2.0 Proposed Change

The proposed change includes the following TS revisions:

1. TS Section 1.1, "Definitions" - A new definition, "Pressure and Temperature Limits Report (PTLR)," is added. The wording for this definition is consistent with that in TSTF-419-A.

2. TS Section 3.4.10, "RCS Pressure and Temperature (P/T) Limits" - The P/T curves and the associated TS wording have been deleted and replaced with references to the PTLR.

3. TS Section 5.6, "Reporting Requirements" - A new Section 5.6.8, "Reactor Coolant System (RCS) Pressure and Temperature Limits Report (PTLR)" has been added. The format and content of new Section 5.6.8 are consistent with that in TSTF-419-A (Reference 2), and the guidance contained in the August 4, 2011 NRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6, "Reporting Requirements". This new section: (1) identifies the individual TS that address reactor coolant system P/T limits; (2) references the NRC-approved topical report that documents PTLR methodologies in a complete citation; and (3) requires that the PTLR and any revision or supplement thereto be submitted to the NRC.

Enclosure 2 provides the existing TS pages marked-up to show the proposed changes. Enclosure 3 provides the revised TS pages. Marked-up pages showing corresponding changes to the TS Bases are provided in Enclosure 4 for information only. The TS Bases changes will be processed in accordance with the Fermi 2 TS Bases Control Program (TS 5.5.10).

Enclosure 5, the Fermi 2 PTLR, provides the P/T curves developed to represent steam dome pressure versus minimum vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline region. Fermi 2 is currently licensed to P/T curves for up to 24 and 32 effective full power years (EFPY); the analysis performed in this report establishes revised curves for up to 24 and 32 EFPY. The 1998 Edition of the ASME Boiler and Pressure Vessel Code including 2000 Addenda was used in this evaluation.

As documented in Section 4.0 of the Safety Evaluation Report for GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, licensees who choose to implement NEDC-33178P-A, Revision 1 as their facility's PTLR methodology must address one plant-specific action item:

- The licensee must identify the report used to calculate the neutron fluence and document that the plant-specific neutron fluence calculation will be performed using an approved neutron fluence calculation methodology.

Accordingly, the PTLR incorporates a fluence calculated in accordance with the GEH Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the NRC (Reference 5), and is in compliance with Regulatory Guide 1.190. The latest information from the Boiling Water Reactor Vessel and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) that is applicable to Fermi 2 has been utilized.

The P/T curves in the PTLR represent 24 and 32 EFPY, where 32 EFPY represents the end of the 40 year license, and 24 EFPY is provided as an intermediate point between the current EFPY and 32 EFPY. The fluence used in developing the P/T curves is conservatively based upon operation at Current Licensed Thermal Power (CLTP) of 3430 MWt for 12.04 EFPY and extended power uprate (EPU) of 3952 MWt for 19.96 EFPY. This is conservative, since the plant did not implement EPU. The fluence was determined for the Water Level Instrumentation (WLI) nozzle, excluding extended power uprate but including Measurement Uncertainty Recapture (MUR) uprate, and is based on operation at Original Licensed Thermal Power of 3293 MWt for 3.228 EFPY, Current Licensed Thermal Power (CLTP) of 3430 MWt for approximately 16.35 EFPY, and at MUR power of 3486 MWt for approximately 10.59 EFPY or through the remainder of the 40 year license. MUR is planned to be implemented after RF16 or beginning in Cycle 17.

As stated previously, relocation of the P/T limit curves to the PTLR is consistent with the guidance provided in NRC Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits"

(Reference 2), TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR" (Reference 3), and the guidance contained in the August 4, 2011 NRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6, "Reporting Requirements".

3.0 Background

10 CFR 50, Appendix G, requires the establishment of P/T limits for material fracture toughness requirements of the Reactor Coolant Pressure Boundary materials. 10 CFR 50, Appendix G requires an adequate margin to brittle failure during normal operation, abnormal operational transients, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G.

Historically, the P/T limit curves for BWRs have been contained in the TS, necessitating the submittal of license amendment requests to update the curves. This caused both the NRC and licensees to expend resources that could otherwise be devoted to other activities.

Generic Letter 96-03 allows plants to relocate their P/T curves and associated numerical limits (such as heatup and cooldown rates) from the plant TS to a PTLR, which is a licensee-controlled document. As stated in Generic Letter 96-03, during the development of the improved Standard Technical Specifications (ISTS), a change was proposed to relocate the P/T limits currently contained in the plant TS to a PTLR. As one of the improvements to the STS, the NRC staff agreed with the industry that the curves may be relocated outside the plant TS to a PTLR so that the licensee could efficiently maintain these limits. One of the prerequisites for having the PTLR option is that the P/T curves and limits be derived using methodologies approved by the NRC, and that the associated licensing topical reports describing the approved methodologies be referenced in the plant TS.

The purpose of GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, is to provide the methodology developed by GEH for the determination of reactor pressure vessel P/T curves. The adequacy of the GEH methodology is demonstrated through a detailed description of the calculation procedures and examples showing agreement between GEH practices and the standards and Code requirements set forth in 10 CFR 50, Appendix G. NEDC-33178P-A, Revision 1, does not include development or licensing of vessel fluence methods. The fluence methods are provided in GEH Licensing Topical Report NEDC-32983P-A, Revision 2.

In order to implement the PTLR, the analytical methods used to develop the P/T limits must be consistent with those previously reviewed and approved by the NRC and must be referenced in the Administrative Controls section of the plant TS. GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, provides the current methodology for developing reactor coolant system P/T limit curves and other associated numerical limits for BWRs. The Fermi 2 P/T curves have been developed in accordance with the NEDC-33178P-A, Revision 1, methodology as documented in the PTLR provided in Enclosure 5.

The P/T curves included in the PTLR have been developed to present steam dome pressure versus minimum reactor vessel metal temperature incorporating appropriate non-beltline limits and irradiation embrittlement effects in the beltline region. Complete P/T curves were developed for 24 and 32 EFPY. These P/T curves and a tabulation of the curves are provided in the PTLR. This report incorporates a fluence (E > 1 MeV) calculated in accordance with the GEH Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the NRC, and is in compliance with Regulatory Guide 1.190. The latest information from the BWRVIP Integrated Surveillance Program that is applicable to Fermi 2 has been utilized.

The methodology used to generate the P/T curves in this report is presented in Section 3.0 of the PTLR. The 1998 Edition of the ASME Boiler and Pressure Vessel Code including 2000 Addenda was used in this evaluation.

The operating limits for pressure and temperature are required for three categories of operation:

1. Hydrostatic pressure tests and leak tests, referred to as Curve A,

2. Non-nuclear heatup/cooldown and low-level physics tests, referred to as core not critical operation or Curve B, and

3. Core critical operation, referred to as Curve C.

There are four vessel regions that should be monitored against the P/T curve operating limits; these regions are defined on the Reactor Pressure Vessel (RPV) thermal cycle diagram:

- Closure flange region (Region A)
- Core beltline region (Region B)
- Upper vessel (Regions A & B)
- Lower vessel (Regions B &C)

For the core not critical and the core critical curves, the P/T curves specify a coolant heatup and cooldown temperature rate of 100°F/hr or less for which the curves are applicable. However, the core not critical and the core critical curves were also developed to bound transients defined on the RPV thermal cycle diagram and the nozzle thermal cycle diagrams. The bounding transients used to develop the curves are described in NEDC-33178P-A, Revision 1. For the hydrostatic pressure and leak test curve, a coolant heatup and cooldown temperature rate of 20°F/hr or less must be maintained at all times. The P/T curves apply for both heatup and cooldown and for both the *1/4T* and *3/4T* locations because the maximum tensile stress for either heatup or cooldown is applied at the *1/4T* location. For beltline curves, this approach has added conservatism because irradiation effects cause the allowable toughness, K1r, at *1/4T* to be less than that at *3/4T* for a given metal temperature. Curves A and B provide separate bottom head as well as composite upper vessel and beltline requirements.

Separate P/T curves were developed for the upper vessel, beltline (at various intermediate and end of license EFPYs), bottom head for the Pressure Test and Core Not Critical conditions. Composite P/T curves were generated for each of the Pressure Test, Core Not Critical and Core Critical conditions at intermediate and end of license EFPYs. The composite curves were generated by enveloping the most restrictive P/T limits from the separate bottom head, beltline, upper vessel and closure assembly P/T limits.

The proposed TS revisions associated with relocation of the P/T limits to a PTLR are consistent with the guidance provided in GL 96-03 as supplemented by TSTF-419-A, and Reference 4.

During an internal review of design basis documentation, questions were raised by Fermi 2 personnel regarding the potential impact of a generic issue identified by GEH for WLI nozzle material on the P/T curves. GEH previously determined that the effect of the lower WLI nozzle material properties had not been properly considered in the development and review of P/T curves for some plants operating to the P/T curves developed by GEH. Some of these plants required revision to their P/T curves to bound the WLI requirements. GEH incorrectly grouped Fermi 2 with the plants not impacted by this issue. However, based on further review by GEH, it was determined that Fermi 2 should have been grouped with the plants requiring revision to their P/T curves. Investigation by GEH determined that the currently licensed 24 EFPY P/T curves (B, core not critical and C, core critical) bound the requirements of the WLI nozzle up to 21 EFPY (Curve A is not impacted). The plant will not reach 21 EFPY levels before the sixteenth refueling outage (RF16), currently scheduled for the first quarter of 2014.

In accordance with NRC Administrative Letter 98-10, "Dispositioning of Technical Specifications That Are Insufficient To Assure Plant Safety," Fermi 2 entered the condition into the corrective action program. The operating procedures that monitor reactor pressure and temperature during heatup, cooldown, and leakage testing were updated with corrected information provided by GEH in a revised Fermi 2 P/T curves report. These operating procedures contain enclosures listing reactor vessel metal temperatures as a function of pressure for all P/T curves, for both 24 EFPY and 32 EFPY. This license amendment request implements the long term corrective action to revise the Fermi 2 P/T curves.

In Reference 6, the U.S. Nuclear Regulatory Commission requested additional information concerning the Grand Gulf Nuclear Station, Unit 1 license amendment request pertaining to the implementation of a PTLR. Enclosures 6 and 7 provide proprietary and nonproprietary versions of responses to these questions for Fermi 2.

4.0 Technical Analysis

Generic Letter 96-03 provides regulatory guidance regarding relocation of P/T curves and associated numerical limits (such as heatup and cooldown rates) from plant TS to a PTLR (a licensee-controlled document). As stated in GL 96-03, a licensee requesting such a change must satisfy the following three criteria:

1. Have NRC-approved methodologies to reference in the TS,

2. Develop a PTLR to contain the P/T limit curves, associated numerical limits, and any necessary explanation, and

3. Modify applicable sections of the TS accordingly.

The NRC-approved methodology of GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, has been adopted for preparation of the Fermi 2 P/T limit curves. Section 5.0, of the NRC Safety Evaluation Report in Reference 1 concluded:

- "The NRC staff concludes that BWROG LTR NEDC-33178P, Revision 1, satisfies the criteria in Attachment 1 in GL 96-03 and provides adequate methodology for BWR licensees to calculate P-T limit curves, given that licensees referencing this LTR comply with the conditions listed in Section 4.0 of this SE. Using this methodology and following the PTLR guidance in GL 96-03, as amended by NRC TSTF-419, BWR licensees will be able to relocate the P/T limit curves from TS to a PTLR, a licensee-controlled document."

As discussed previously, the PTLR incorporates a fluence calculated in accordance with the GE Licensing Topical Report NEDC-32983P-A, Revision 2, which has been approved by the NRC (Reference 5), and is in compliance with Regulatory Guide 1.190, thus satisfying the requirement contained in Section 5.0 of the NRC Safety Evaluation Report.

Proposed revisions to applicable sections of the TS have been prepared and are provided in Enclosure 2 to this submittal. These proposed TS changes are consistent with the guidance provided in GL 96-03, as supplemented by TSTF-419-A, and the guidance contained in the August 4, 2011 NRC letter (Reference 4) which requires the full methodology citation in TS Section 5.6, "Reporting Requirements".

In 2011, it was determined that Fermi 2 should have been grouped with the plants requiring revision to their P/T curves based on WLI material properties. Investigation by GEH determined that the currently licensed 24 EFPY P/T curves bound the requirements of the WLI nozzle, but only up to 21 EFPY. Revised P/T curves have been generated for Fermi 2. The PTLR incorporates the corrections needed to address the WLI nozzle material properties.

The NRC has approved similar license amendments utilizing the guidance provided in GL 96-03 and TSTF419-A to relocate P/T limit curves to a PTLR. Recent examples for boiling water reactor plants include:

1. Grand Gulf Nuclear Station, Unit 1, (License Amendment No. 191 issued by NRC letter dated July 18, 2012 – ADAMS Accession No. ML121210020).

2. Pilgrim Nuclear Power Station (License Amendment No. 234 issued by NRC letter dated January 26, 2011 - ADAMS Accession No. ML110050298).

3. Nine Mile Point Nuclear Station, Unit 1 (License Amendment No. 204 issued by NRC letter dated January 1, 2010 - ADAMS Accession No. ML093370002).

5.0 Regulatory Safety Analysis

5.1 <u>No Significant Hazards Consideration</u>

In accordance with 10 CFR 50.92, Detroit Edison has made a determination that the proposed amendment involves no significant hazards consideration.

1. The proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

   The proposed changes modify the TS by replacing references to existing reactor vessel heatup and cooldown rate limits and P/T limit curves with references to the PTLR. The proposed amendment also adopts the NRC approved methodology of the GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, for the preparation of the Fermi 2 P/T limit curves. In 10 CFR 50, Appendix G, requirements are established to protect the integrity of the Reactor Coolant Pressure Boundary in nuclear power plants. Implementing the NRC-approved methodology for calculating P/T limit curves and relocating those curves to the PTLR provides an equivalent level of assurance that Reactor Coolant Pressure Boundary integrity will be maintained, as specified in 10 CFR 50, Appendix G.

   The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration of the plant or the manner in which the plant is operated and maintained. The ability of structures, systems, and components to perform their intended safety functions is not altered or prevented by the proposed changes, and the assumptions used in determining the radiological consequences of previously evaluated accidents are not affected.

   Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. The proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

   The change in methodology for calculating P/T limits and the relocation of those limits to the PTLR does not alter or involve any design basis accident initiators. Reactor Coolant Pressure Boundary integrity will continue to be maintained in accordance with 10 CFR 50, Appendix G, and the assumed accident performance of plant structures, systems and components will not be affected. These changes do not involve any physical alteration of the plant (i.e., no new or different type of equipment will be installed), and installed equipment is not being operated in a new or different manner. Thus, no new failure modes are introduced.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. The proposed change does not involve a significant reduction in the margin of safety. The proposed changes do not affect the function of the Reactor Coolant Pressure Boundary or its response during plant transients. By calculating the P/T limits using NRC-approved methodology, adequate margins of safety relating to Reactor Coolant Pressure Boundary integrity are maintained. The proposed changes do not alter the manner in which safety limits, limiting safety system settings, or limiting conditions for operation are determined. There are no changes to setpoints at which protective actions are initiated, and the operability requirements for equipment assumed to operate for accident mitigation are not affected.

   Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

Based on the above evaluation, Detroit Edison concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and accordingly, a finding of no significant hazards consideration is justified.

In conclusion, based on the considerations discussed above, (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) 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 the health and safety of the public.

5.2 Applicable Regulatory Requirements

As discussed in the Safety Evaluation Report for GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, the NRC has established requirements in 10 CFR 50, Appendix G in order to protect the integrity of the Reactor Coolant Pressure Boundary in nuclear power plants. Appendix G requires that the P/T limits for an operating light-water nuclear reactor be at least as conservative as those that would be generated if the methods of Appendix G to Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code were used to generate the P/T limits. 10 CFR Part 50, Appendix G also requires that applicable surveillance data from RPV material surveillance programs be incorporated into the calculations of plant specific P/T limits, and that the P/T limits for operating reactors be generated using a method that accounts for the effects of neutron irradiation on the material properties of the RPV beltline materials. NRC regulatory guidance related to P/T limit curves is found in Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Materials" and Standard Review Plan, NUREG-0800, Section 5.3.2, "Pressure-Temperature Limits, Upper-Shelf Energy, and

Pressurized Thermal Shock".

Adoption of the NRC-approved methodology described in the GEH Nuclear Energy Licensing Topical Report NEDC-33178P-A, Revision 1, for the preparation of the P/T limit curves ensures that the requirements of 10 CFR 50, Appendix G will be satisfied. 10 CFR Part 50, Appendix H provides criteria for the design and implementation of reactor pressure vessel material surveillance programs for operating light water reactors. Fermi 2 demonstrates its compliance with the requirements of 10 CFR 50, Appendix H through participation in the BWRVIP ISP and the latest material information was utilized in preparation of the report.

The PTLR incorporates the corrections needed to address the WLI nozzle material properties not being properly considered. In 2011 it was determined that Fermi 2 should have been grouped with the plants requiring revision to their P/T curves based on WLI material properties. Investigation by GEH determined that the currently licensed 24 EFPY P/T curves bound the requirements of the WLI nozzle, but only up to 21 EFPY. The PTLR is based on revised Fermi 2 P/T curves.

6.0 Environmental Considerations

Detroit Edison has reviewed the proposed change against the criteria of 10 CFR 51.22 for environmental considerations. The proposed change does not involve a significant hazards consideration, nor does it significantly change the types or significantly increase the amounts of effluents that may be released offsite. The proposed change does not significantly increase individual or cumulative occupational radiation exposures. Based on the foregoing, Detroit Edison concludes that the proposed change meets the criteria provided in 10 CFR 51.22(c)(9) for a categorical exclusion from the requirements for an Environmental Impact Statement or an Environmental Assessment.

7.0 References

1. Letter from D. Coleman (BWR Owners' Group) to U.S. Nuclear Regulatory Commission, "Submittal of GE BWROG Topical Report NEDC-33178P-A, Revision 1, 'General Electric Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves'," ML092370486, dated July 29, 2009

2. Generic Letter (GL) 96-03, "Relocation of the Pressure Temperature Limit Curves and Low Temperature Overpressure Protection System Limits," dated January 31, 1996

3. Technical Specification Task Force (TSTF) Traveler TSTF-419-A, "Revise PTLR Definition and References in ISTS 5.6.6, RCS PTLR," dated August 4, 2003

4. Letter from J. Jolicoeur (U.S. Nuclear Regulatory Commission) to Technical Specification Task Force, "Implementation of Travelers TSTF-363, Revision 0, 'Revise Topical Report References in ITS 5.6.5, COLR (Core Operating Limits Report),' TSTF-408, Revision 1, 'Relocation of LTOP (Low-Temperature Overpressure Protection) Enable Temperature and PORV (Power-Operated Relief Valve) Lift Setting to the PTLR (Pressure-Temperature Limits Report),' and TSTF-419, Revision 0, 'Revise PTLR Definition and References in ISTS (Improved Standard Technical Specification) 5.6.6, RCS (Reactor Coolant System) PTLR'," ML 110660285, dated August 4, 2011

5. Letter from G. Stramback (GE) to U.S. Nuclear Regulatory Commission, "Accepted Version of GE Licensing Topical Report NEDC-32983P-A, Revision 2 (TAC No. MC3788)," ML072480116, dated February 1, 2006

6. Letter from M. A. Krupa (Entergy Operations, Inc.) to U.S. Nuclear Regulatory Commission, "Request for Additional Information Regarding Extended Power Uprate," ML110540540, dated February 23, 2011

Enclosure 2 to
NRC-12-0037

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

Markup of Existing TS Pages

1.1-5
3.4-23 to 3.4-28
3.4-28a to 3.4-28e
5.0-22

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.

Insert

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 ~~limits.~~

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.	30 minutes
	<u>AND</u>		
	A.2	Determine RCS is acceptable for continued operation.	72 hours
B. Required Action and associated Completion Time of Condition A not met.	B.1	Be in MODE 3.	12 hours
	<u>AND</u>		
	B.2	Be in MODE 4.	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. <u>AND</u> C.2 Determine RCS is acceptable for operation.	Immediately 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:~~ Verify RCS pressure, RCS temperature, and RCS heatup and cooldown rates are within the limits specified in the PTLR. ~~a. RCS pressure and RCS temperature are to the right of the applicable limits specified in Figures 3.4.10-1 through 3.4.10-6~~; and ~~b. RCS heatup and cooldown rates are limited to:~~ ~~1. ≤ 100°F in any 1 hour period; and~~ ~~2. ≤ 20°F in any 1 hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves.~~	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 ~~Figures 3.4.10-3 or 3.4.10-6, as applicable.~~ 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 ~~≤ 145°F.~~ 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 ~~≤ 50°F.~~ 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 ~~≤ 145°F.~~ 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 ~~≤ 50°F.~~ 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 ~~≥ 72°F~~ 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 ~~≥ 72°F.~~ 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 ~~≥ 72°F.~~ within the limits specified in the PTLR.	12 hours

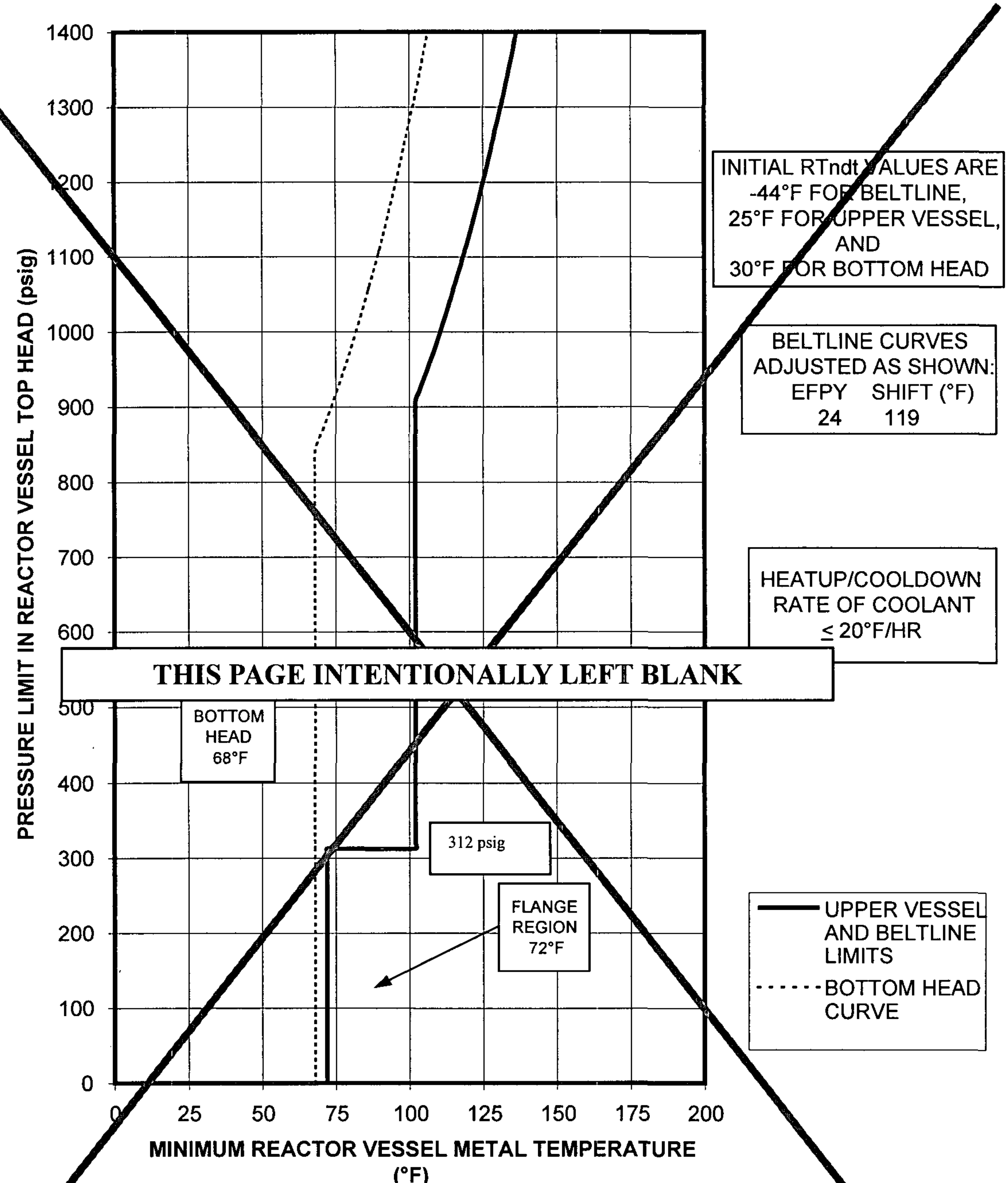

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FIGURE 3.4.10-1
SYSTEM PRESSURE TEST P/T CURVE (CURVE A) FOR UP TO 24 EFPY

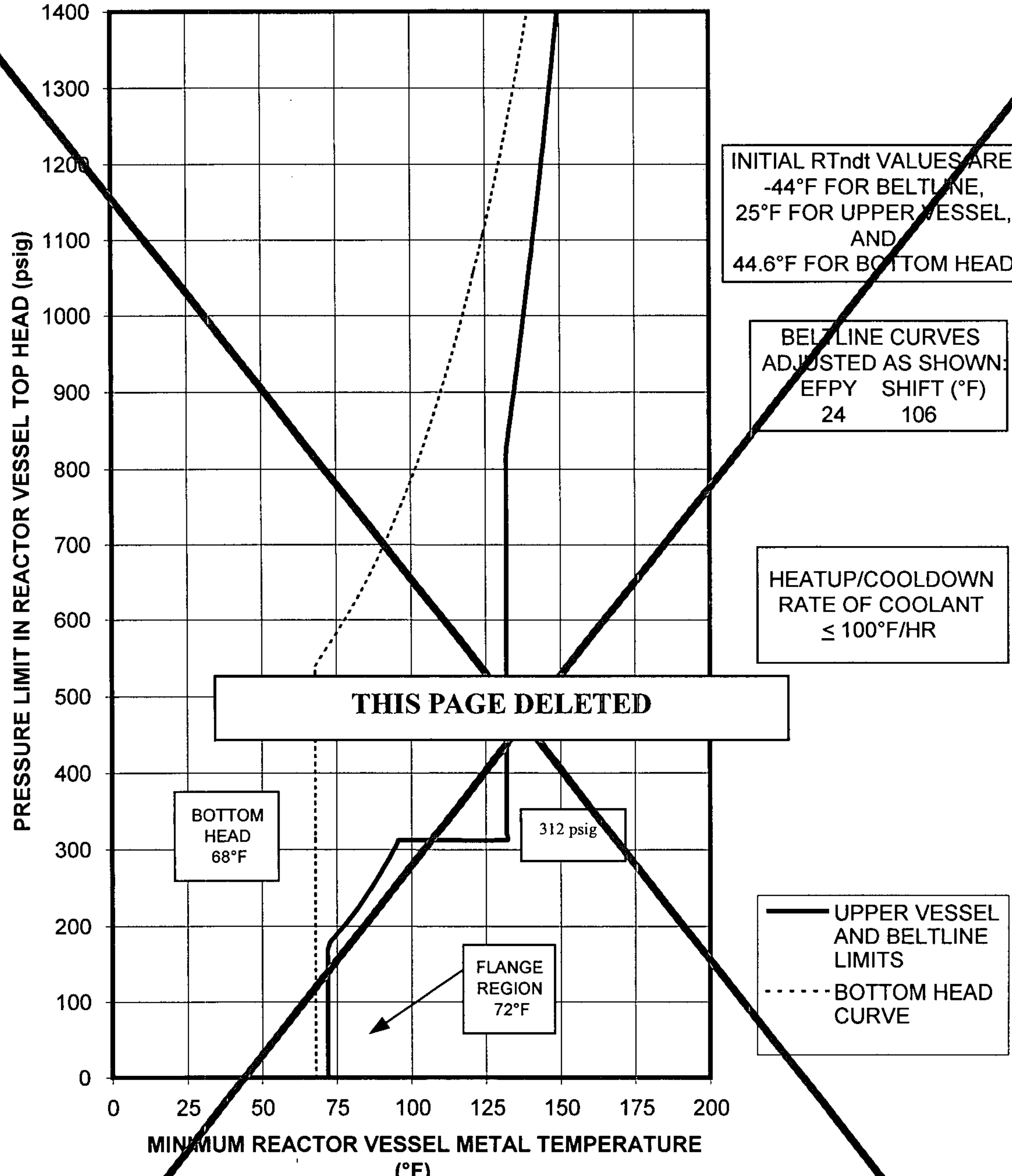

FIGURE 3.4.10-2
CORE NOT CRITICAL P/T CURVE (CURVE B) FOR UP TO 24 EFPY

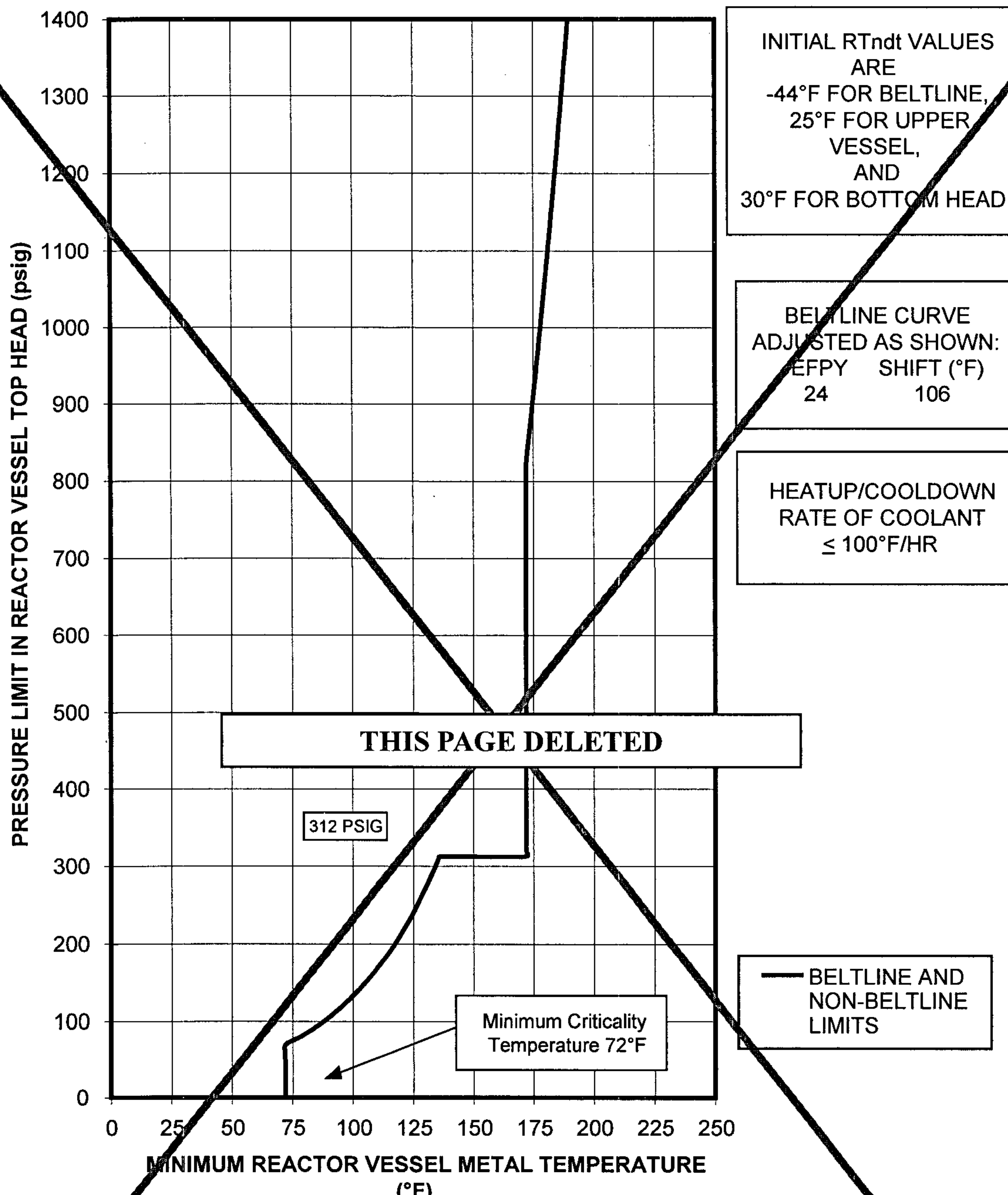

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FIGURE 3.4.10-3
CORE CRITICAL P/T CURVE (CURVE C) FOR UP TO 24 EFPY

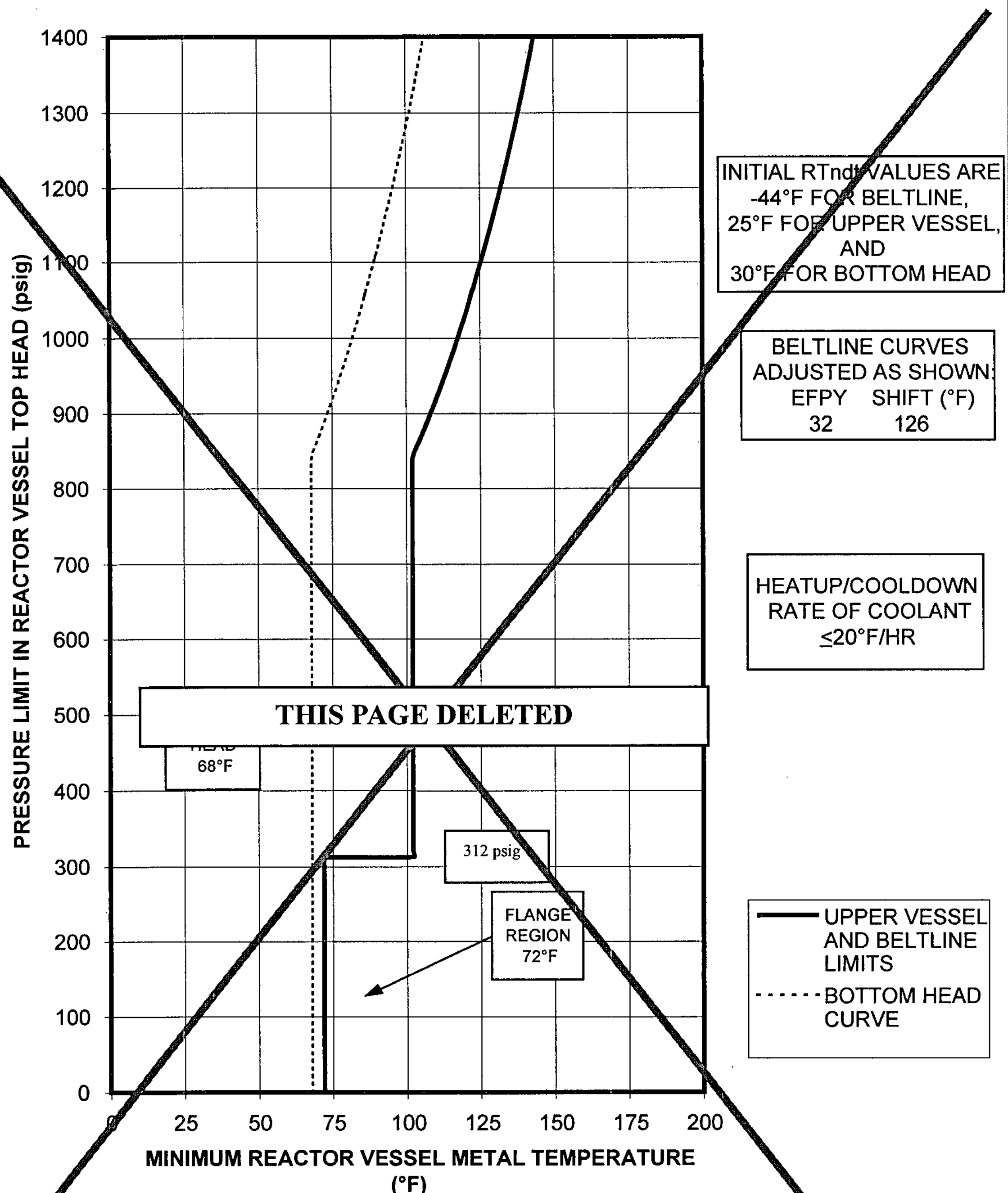

FIGURE 3.4.10-4
SYSTEM PRESSURE TEST P/T CURVE (CURVE A) FOR UP TO 32 EFPY

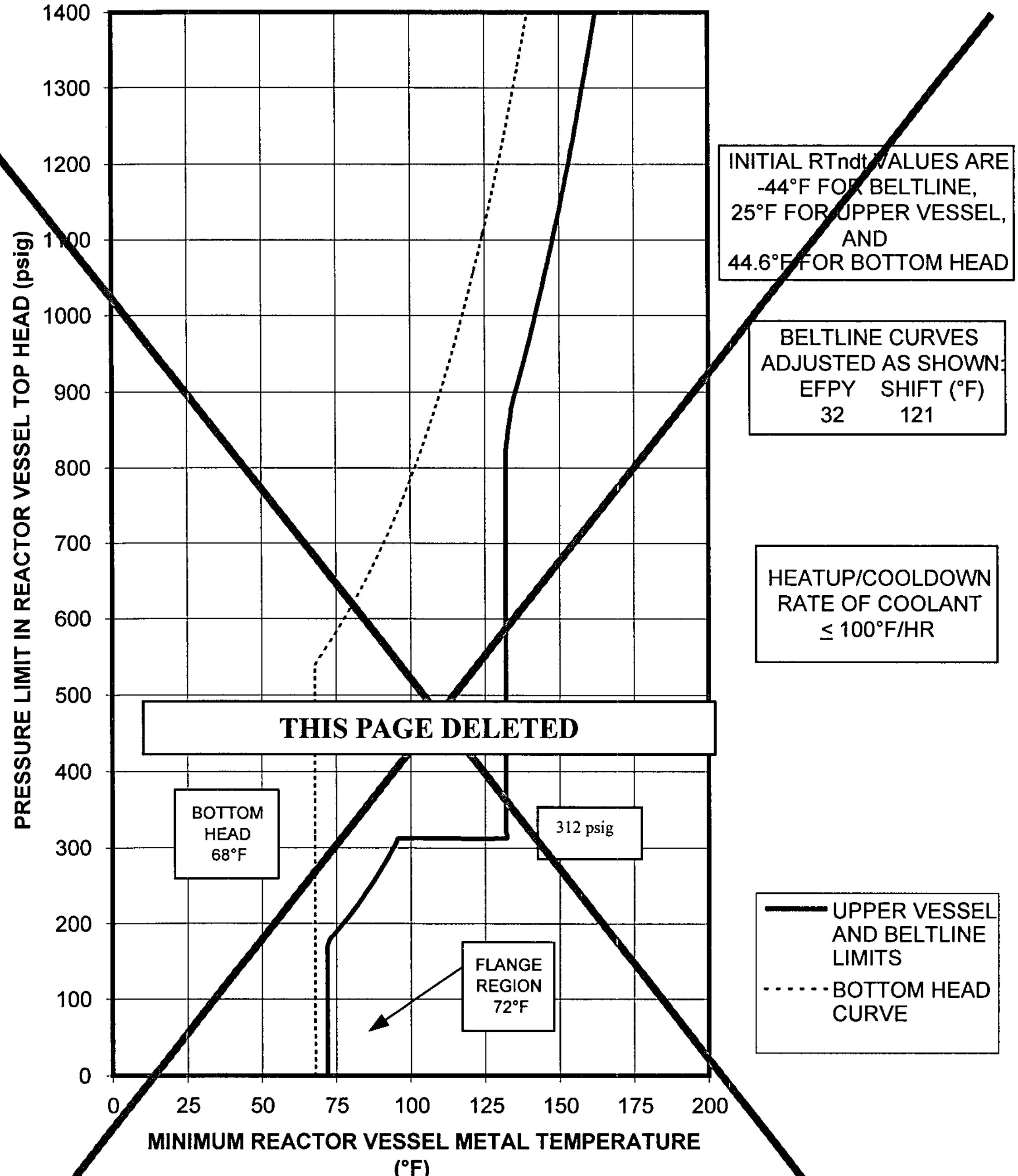

FIGURE 3.4.10-5
CORE NOT CRITICAL P/T CURVE (CURVE B) FOR UP TO 32 EFPY

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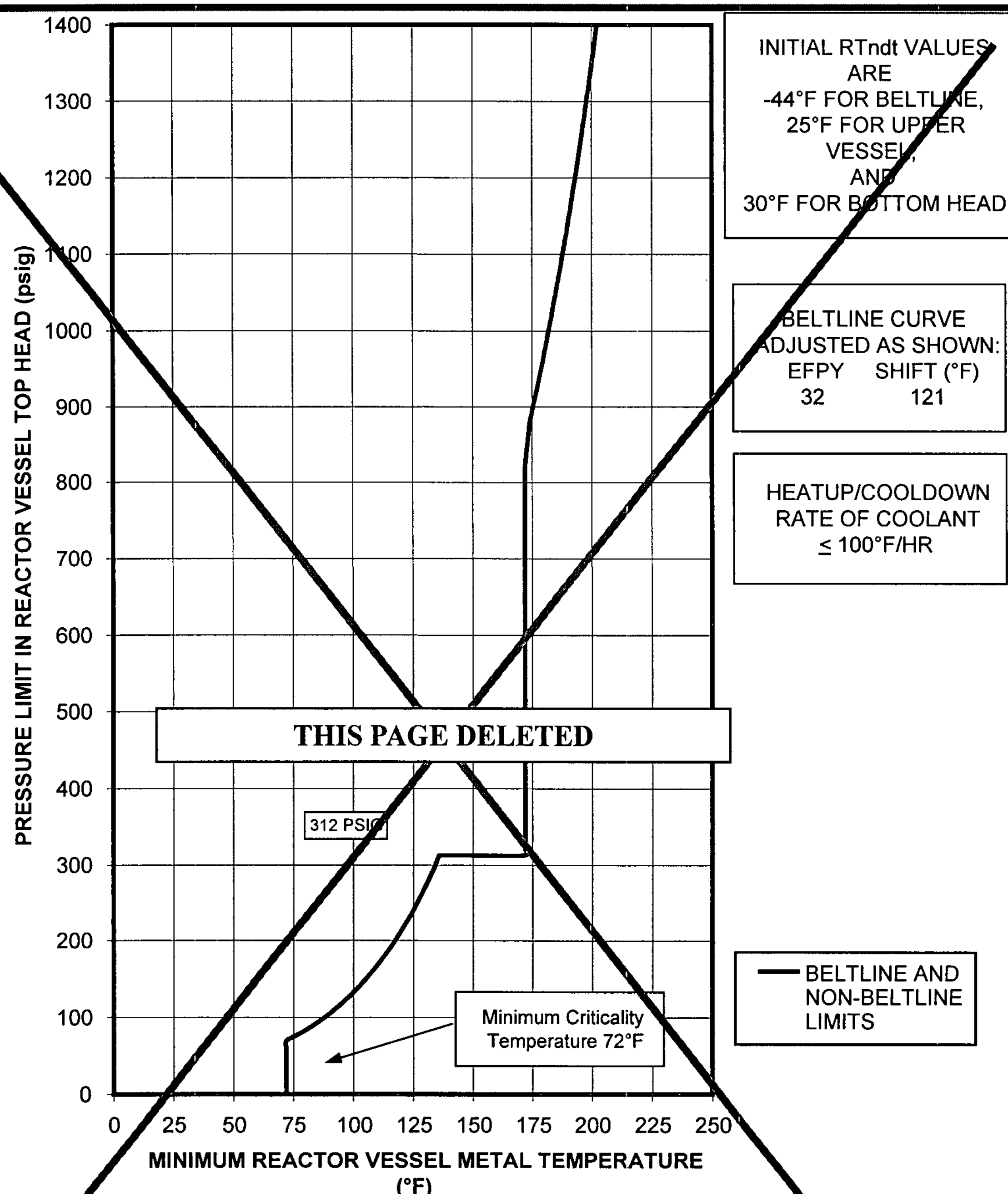

FIGURE 3.4.10-6
CORE CRITICAL P/T CURVE (CURVE C) FOR UP TO 32 EFPY

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.

Insert

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.

Enclosure 3 to
NRC-12-0037

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

Revised TS Pages

1.1-5
3.4-23 to 3.4-28
5.0-22

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. <u>AND</u> 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. <u>AND</u> 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
	AND		
	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.

Enclosure 4 to
NRC-12-0037

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

Markup of Existing TS Bases Pages
(For Information Only)

B 3.4-10-1
B 3.4-10-3
B 3.4-10-6
B 3.4-10-7
B 3.4-10-8
B 3.4-10-9

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.10 RCS Pressure and Temperature (P/T) Limits

BASES

BACKGROUND

All components of the RCS are designed to withstand effects of cyclic loads due to system pressure and temperature changes. These loads are introduced by startup (heatup) and shutdown (cooldown) operations, power transients, and reactor trips. This LCO limits the pressure and temperature changes during RCS heatup and cooldown, within the design assumptions and the stress limits for cyclic operation.

The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR) (Ref 7)

~~Figures 3.4.10-1 through 3.4.10-6~~ contain P/T limit curves for hydrostatic or leak testing (Curve A); for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power physics tests (Curve B); and for operations with a critical core other than low power physics tests (Curve C). Two sets of curves are provided; the first set is applicable provided reactor generated energy did not exceed 24 Effective Full Power Years (EFPY), and the second set is applicable for 32 EFPY and bounds operation throughout the 40-year plant license. Other related P/T limits are provided in the ~~SRs.~~

PTLR.

Each P/T limit curve defines an acceptable region for normal operation. The usual use of the curves is operational guidance during heatup or cooldown maneuvering, when pressure and temperature indications are monitored and compared to the applicable curve to determine that operation is within the allowable region.

The LCO establishes operating limits that provide a margin to brittle failure of the reactor vessel and piping of the reactor coolant pressure boundary (RCPB). The vessel is the component most subject to brittle failure. Therefore, the LCO limits apply mainly to the vessel.

10 CFR 50, Appendix G (Ref. 1), requires the establishment of P/T limits for material fracture toughness requirements of the RCPB materials. Reference 1 requires an adequate margin to brittle failure during normal operation, anticipated operational occurrences, and system hydrostatic tests. It mandates the use of the ASME Code, Section III, Appendix G (Ref. 2).

BASES

APPLICABLE SAFETY ANALYSES

The P/T limits are not derived from Design Basis Accident (DBA) analyses. They are prescribed during normal operation to avoid encountering pressure, temperature, and temperature rate of change conditions that might cause undetected flaws to propagate and cause nonductile failure of the RCPB, a condition that is unanalyzed. Since the P/T limits are not derived from any DBA, there are no acceptance limits related to the P/T limits. Rather, the P/T limits are acceptance limits themselves since they preclude operation in an unanalyzed condition.

RCS P/T limits satisfy Criterion 2 of 10 CFR 50.36(c)(2)(ii).

the limits specified in the PTLR

LCO

The elements of this LCO are:

a. RCS pressure, temperature, and heatup or cooldown rate are within ~~limits~~ during RCS heatup, cooldown, and inservice leak and hydrostatic testing;

the limit of the PTLR

b. The temperature difference between the reactor vessel bottom head coolant and the reactor pressure vessel (RPV) steam space coolant is within ~~limit~~ during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow;

the limits of the PTLR

c. The temperature difference between the reactor coolant in the respective recirculation loop and in the reactor vessel meets ~~limit~~ during recirculation pump startup, and during increases in THERMAL POWER or loop flow while operating at low THERMAL POWER or loop flow with an idle recirculation loop;

the criticality limits specified in the PTLR

d. RCS pressure and temperature are within ~~criticality limits,~~ prior to achieving criticality; and

e. The reactor vessel flange and the head flange temperatures are within ~~limits~~ when the reactor vessel head bolting studs are under tension.

These limits define allowable operating regions and permit a large number of operating cycles while also providing a wide margin to nonductile failure.

BASES

ACTIONS (continued)

C.1 and C.2

Operation outside the P/T limits in other than MODES 1, 2, and 3 (including defueled conditions) must be corrected so that the RCPB is returned to a condition that has been verified by stress analyses. The Required Action must be initiated without delay and continued until the limits are restored.

Besides restoring the P/T limit parameters to within limits, an evaluation is required to determine if RCS operation is allowed. This evaluation must verify that the RCPB integrity is acceptable and must be completed before approaching criticality or heating up to > 200°F. Several methods may be used, including comparison with pre-analyzed transients, new analyses, or inspection of the components. ASME Code, Section XI, Appendix E (Ref. 6), may be used to support the evaluation; however, its use is restricted to evaluation of the beltline.

Condition C is modified by a Note requiring Required Action A.2 be completed whenever the Condition is entered. The Note emphasizes the need to perform the evaluation of the effects of the excursion outside the allowable limits. Restoration alone per Required Action C.1 is insufficient because higher than analyzed stresses may have occurred and may have affected the RCPB integrity.

SURVEILLANCE REQUIREMENTS

SR 3.4.10.1

Verification that operation is within ~~the applicable limits of Figures 3.4.10-1 through 3.4.10-6~~ PTLR limits is required every 30 minutes when RCS pressure and temperature conditions are undergoing planned changes. This Frequency is considered reasonable in view of the control room indication available to monitor RCS status. Also, since temperature rate of change limits are specified in hourly increments, 30 minutes permits a reasonable time for assessment and correction of minor deviations.

Surveillance for heatup, cooldown, or inservice leakage and hydrostatic testing may be discontinued when the criteria given in the relevant plant procedure for ending the activity are satisfied.

BASES

SURVEILLANCE REQUIREMENTS (continued)

This SR has been modified with a Note that requires this Surveillance to be performed as applicable only during system heatup and cooldown operations and inservice leakage and hydrostatic testing.

SR 3.4.10.2

A separate limit is used when the reactor is approaching criticality. Consequently, the RCS pressure and temperature must be verified within the appropriate limits before withdrawing control rods that will make the reactor critical.

PTLR

Performing the Surveillance within 15 minutes before control rod withdrawal for the purpose of achieving criticality provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the control rod withdrawal.

SR 3.4.10.3, SR 3.4.10.4, SR 3.4.10.5, and SR 3.4.10.6

Differential temperatures within the applicable limits ensure that thermal stresses resulting from the startup of an idle recirculation pump will not exceed design allowances. Limiting the temperature differential between and idle recirculation loop and the core inlet temperature ~~to ≤ 50°F~~ prior to startup of the idle loop maintains consistency with the assumptions of the reactor vessel nozzle and reactor recirculation system ASME Upset category stress analysis and partial power fuel thermal limit analyses.

Limiting differential temperatures within the applicable limits, during a THERMAL POWER increase or recirculation flow increase in single loop operation, while THERMAL POWER ≤ 30% RTP or operating loop flow ≤ 50% of rated loop flow, ensure that thermal stresses will not exceed design allowances.

Performing the Surveillance within 15 minutes before starting the idle recirculation pump, THERMAL POWER increase during single loop operation, or recirculation flow increase during single loop operation, provides adequate assurance that the limits will not be exceeded between the time of the Surveillance and the time of the idle pump start, power

SURVEILLANCE REQUIREMENTS (continued)

increase, or flow increase.

An acceptable means of demonstrating compliance with the temperature differential requirement in SR 3.4.10.4 and SR 3.4.10.6 is to compare the temperatures of the operating recirculation loop and the idle loop.

These SRs have been modified by Notes that require the Surveillance to be performed only in certain MODES. In MODE 5, the overall stress on limiting components is lower. Therefore, ΔT limits are not required for SRs 3.4.10.3 and 3.4.10.4 in MODE 5. In MODES 3, 4, and 5, THERMAL POWER increases are not possible, and recirculation flow increases will not result in additional stresses. Therefore ΔT limits are only required for SRs 3.4.10.5 and 3.4.10.6 in MODES 1 and 2. The Notes also state that the SR is only required to be met during the event of concern (e.g., pump startup, power increase or flow increase) since this is when the stresses occur.

<u>SR 3.4.10.7, SR 3.4.10.8, and SR 3.4.10.9</u>

Limits on the reactor vessel flange and head flange temperatures are generally bounded by the other P/T limits during system heatup and cooldown. However, operations approaching MODE 4 from MODE 5 and in MODE 4 with RCS temperature less than or equal to certain specified values require assurance that these temperatures meet the LCO limits.

the limits specified in the PTLR.

The flange temperatures must be verified to be above the limits 30 minutes before and while tensioning the vessel head bolting studs to ensure that once the head is tensioned the limits are satisfied. When in MODE 4 with RCS temperature ≤ 80° F, 30 minute checks of the flange temperatures are required because of the reduced margin to the limits. When in MODE 4 with RCS temperature ≤ 100°F, monitoring of the flange temperature is required every 12 hours to ensure the temperature is within ~~limits~~.

The 30 minute Frequency reflects the urgency of maintaining the temperatures within limits, and also limits the time that the temperature limits could be exceeded. The 12 hour Frequency is reasonable based on the rate of temperature change possible at these temperatures.

BASES

REFERENCES

1. 10 CFR 50, Appendix G.

2. ASME, Boiler and Pressure Vessel Code, Section III, Appendix G.

3. 10 CFR 50, Appendix H.

4. BWRVIP-86-A, October 2002

5. Regulatory Guide 1.99, Revision 2, May 1988.

6. ASME, Boiler and Pressure Vessel Code, Section XI, Appendix E.

7. The PRESSURE AND TEMPERATURE LIMITS REPORT (PTLR).

Enclosure 7 to
NRC-12-0037

Fermi 2 NRC Docket No. 50-341
Operating License No. NPF-43

Proposed License Amendment to Relocate Pressure and Temperature Curves to a Pressure Temperature Limits Report

GE Hitachi Non-Proprietary Notice

Responses to Requests for Additional Information – Non-Proprietary

Enclosure 2

1-2RXPKK-13

GEH Responses to GGNS PT Curve RAIs

Non-Proprietary Information-Class I (Public)

Non-Proprietary Notice

This is a non-proprietary version of Enclosure 1 of 1-2RXPKK-13 which has the proprietary information removed. Portions of the document that have been removed are indicated by an open and closed bracket as shown here [[ ]].

NRC RAI 4

Do the PT limit curves include a hydrostatic pressure adjustment for the column of water in a full RPV? If so, provide the pressure head used in the PT limit curve analysis.

GEH Response

Yes, the PT limit curves include a hydrostatic pressure adjustment for the column of water in a full RPV.

The pressure head for the beltline hydrostatic test curve (Curve A) for Fermi 2 is 23.3 psig. This is determined using the height of the vessel and the elevation of the bottom of active fuel.

The full vessel pressure head is 31.1 psig. This pressure is indirectly used in the PT curve analysis. It is considered in the determination of K_I for the bottom head curve as discussed in the PT curve licensing topical report (LTR), *GE Hitachi Nuclear Energy Methodology for Development of Reactor Pressure Vessel Pressure-Temperature Curves*, NEDC-33178P-A, Revision 1. Sections 4.3.2.1.1 and 4.3.2.2.2 of that LTR include additional discussion regarding the use of the pressure head.

NRC RAI 5

Address inconsistencies between the statement that "the PT curves are beltline (A1224-1 plate) limited above 1330 psig for Curve A for 35 EFPY..." and the NRC staff determination that the PT curves are beltline (A1224-1 plate) limited ~1360 psig from data in Table 1 of GNRO-2010/00056.

GEH Response

This RAI is not applicable to Fermi 2 as it addresses a plant-specific GGNS statement.

NRC RAI 6

Provide the surveillance data and the analysis of the surveillance data used to determine ART from Reference 6.3 (BWRVIP-135, Revision 1 "BWR Vessel and Internals Project Integrated Surveillance Program (ISP) Data Source Book and Plant Evaluations"), as required by NEDC-33178P-A.

GEH Response

BWRVIP-135, Revision 2 provides the surveillance data considered in determining the chemistry and any adjusted Chemistry Factors (CF) for the beltline materials.

Excerpt from BWRVIP-135, Revision 2:

[[

]]

For Fermi 2, the Integrated Surveillance Program (ISP) representative plate, heat [[ ]], is not the target plate. This heat was contained in two Hatch Unit 1 capsules that have been tested and analyzed. The resultant chemistry is [[ ]] Cu and [[ ]] Ni. The CF from Regulatory Guide 1.99, Revision 2 (RG1.99) is [[ ]]. The fitted CF is [[ ]]; however, as the ISP material is not the target material, the ART table evaluated the ISP plate material using the RG1.99 CF. This material was not considered in determining the limiting ART for the PT curves; i.e., this is not the limiting material.

Excerpt from BWRVIP-135, Revision 2:

[[

]]

For Fermi 2, the ISP representative weld, heat [[ ]], which is also known as [[ ]], is the same as the target weld. This heat was contained in Supplemental Surveillance capsules that have been tested and analyzed. The resultant chemistry is [[ ]] Cu and [[ ]] Ni. The CF from RG1.99 is [[ ]]. The fitted CF is [[ ]]. As the ISP weld heat is the identical heat to the target vessel weld, the ART table evaluated the ISP weld material using an adjusted CF. The adjusted CF is [[ ]], calculated in accordance with RG1.99. This material was considered in determining the limiting ART for the PT curves.

Excerpt from BWRVIP-135, Revision 2:

[[

]]

NRC RAI 7

Provide additional detail for the non-beltline analysis conducted in the following areas in order for the NRC staff to complete independent verification of the proposed PT limits:

a. *Identify limiting materials for the Reference Temperature for Nil Ductility Transition (RT_{NDT}) values used to shift the generic Bottom Head and Upper Vessel PT curves when applying NEDC-33178P-A.*

b. *The NRC staff identified a limiting RT_{NDT} of 10°F for the Bottom Head Torus Plates, while GGNS assumed a RT_{NDT} of 24.6°F for the Bottom Head Curve B. Support all RT_{NDT} values reported by providing details of any plant-specific analysis conducted.*

c. *Explain minor differences in assumed RT_{NDT} values for the Bottom Head. Specifically Curves A and C assume a limiting RT_{NDT} of 19°F, while Curve B assumes a limiting RT_{NDT} of 24.6°F.*

d. *Which region of the RPV is limiting for Curve C < 312 psig.*

GEH Response

Item (a)

In order to determine how much to shift the Pressure-Temperature (PT) curves, an evaluation is performed using Tables 4-4a and 4-5a from NEDC-33178P-A. These tables define the required Temperature minus Reference Temperature of Nil Ductility Transition ($T-RT_{NDT}$) that is used to adjust the non-shifted curves. Each component listed in these tables is evaluated using the plant-specific initial RT_{NDT} for each component. The required temperature is then determined by adding the $T-RT_{NDT}$ to the plant-specific RT_{NDT}, resulting in the required T for the curve. As the upper vessel curve is initially based on the non-shifted feedwater (FW) nozzle $T-RT_{NDT}$, all resulting T values are compared to the FW nozzle T. The difference between the maximum T and the FW nozzle $T-RT_{NDT}$ is used to shift the upper vessel curve. The same method is applied for the Control Rod Drive (CRD) curve. In this manner, it is assured that each curve bounds the maximum discontinuity that is represented.

For the Fermi 2 upper vessel curve, the maximum T value from the method described above is [[ ]]. The initial required $T-RT_{NDT}$ for the [[ ]]. This is then adjusted by the Fermi 2-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the other components bounded by the upper vessel curve, the limiting value is for the [[ ]]. The required $T-RT_{NDT}$ for the [[ ]], which is added to the [[ ]]. It is seen that the resulting T required for the [[ ]]. Similarly, the required $T-RT_{NDT}$ for the [[ ]] and the [[ ]]. This also results in a required T of [[ ]]. As [[ ]] is limiting, the Fermi 2 upper vessel curve is based on an RT_{NDT} of [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-4a of NEDC-33178P-A; only the limiting cases are presented here.

Items (b) and (c)

It is noted that the original RAIs were prepared during the NRC review of the Grand Gulf Nuclear Station PT curve report. The values cited in the RAI are not applicable to Fermi 2; however, responses are provided as applicable to the Fermi 2 PT curve calculations.

For the Fermi 2 bottom head or CRD [[ ]], respectively), the maximum T value from the method described above is [[ ]]. The required T-RT_{NDT} for the [[ ]]; this is adjusted by the Fermi 2-specific maximum [[ ]], resulting in [[ ]]. Comparing this to the next limiting value, the required T-RT_{NDT} is [[ ]], which is added to the [[ ]]. It is seen that the resulting T required for the top head nozzles is [[ ]]. As [[ ]] than [[ ]], the Fermi 2 bottom head (CRD) curve is based on an [[ ]]. As noted above, this calculation was performed for each component shown in Table 4-5a of NEDC-33178P-A; only the limiting case is presented here.

Appendix H of NEDC-33178P-A contains the details of an analysis performed to determine the baseline requirement (non-shifted) for the [[ ]]. It can be seen in Section H.5 of Appendix H that the stresses developed in this finite element analysis demonstrated that the [[ ]], resulting in a baseline non-shifted required T-RT_{NDT} of [[ ]]. Therefore, considering the determination of the required shift from the paragraph above for [[ ]], calculations for all components listed in Table 4-5a of NEDC-33178P-A were compared to the CRD T, which is [[ ]] (where [[ ]] materials). Therefore, the shift for the bottom head [[ ]].

Item (d)

For Curve C, the upper vessel and beltline regions are bounding at pressures up to 60 psig. For pressures between 60 psig and 312.5 psig, the upper vessel is bounding; this is true for both 24 and 32 EFPY.

NRC RAI 8

Attachment 7 identifies nozzle N12 as a beltline water level instrument nozzle and notes that an evaluation was conducted using the limiting material properties for the adjoining shell ring, which appears to be appropriate as nozzle N12 is identified as austenitic. Provide details of this evaluation which demonstrates that the drill hole for the beltline water level instrument nozzle is not limiting.

GEH Response

The Fermi 2 N16 water level instrumentation nozzle is fabricated from SA508 Class 1 carbon steel material. Therefore, the N16 nozzle was evaluated for ART. Appendix J of NEDC-33178P-A provides detailed results of an analysis performed for the water level instrumentation nozzle that provides the required stresses for the drill hole in the shell plate. These stresses were used to generate a specific curve applicable for the water level instrumentation nozzle to assure that this location is bounded in the PT curves.

The fluence utilized for the N16 WLI nozzle is based on a less conservative power history that excludes Extended Power Uprate (EPU) and considers Thermal Power Optimization (TPO) after RFO16. The peak surface fluence of 1.65e17 n/cm^2 considers operation at 3293 MWt for 3.228 EFPY, 3430 MWt for 16.35 EFPY, and 3486 MWt (TPO) for 10.59 EFPY.

For Fermi 2, the water level instrumentation nozzles are [[ ]] for 24 EFPY. For Curve B at 24 EFPY, the water level instrumentation nozzle is [[ ]]. For 32 EFPY, the water level instrumentation nozzles are [[ ]]. For Curve B, the water level instrumentation nozzle is [[ ]].

Similarly, for Curve C, the water level instrumentation nozzle is [[ ]].

NRC RAI 9

Provide details on any plant-specific feedwater nozzle evaluation conducted in support of the proposed PT limits or explain why plant-specific evaluation was not required.

GEH Response

An evaluation was performed for the feedwater nozzle as described in Section 4.3.2.1.3 of NEDC-33178P-A. The first part of the evaluation is as described in the response to RAI #7, where it is assured that the limiting component that is represented by the upper vessel nozzle curve is bounded by the [[ ]]. A second evaluation was performed using the Fermi 2-specific feedwater nozzle dimensions; this evaluation is shown below to demonstrate that the [[ ]] curve is applicable to Fermi 2:

Vessel radius to base metal, R_v	[[
Vessel thickness, t_v	
Vessel pressure, P_v	
Pressure stress = PR/t = [[ ]]	
Dead Weight + Thermal Restricted Free End stress	
Total Stress = [[ ]]	]]

The factor F (a/r_n) from Figure A5-1 of "PVRC Recommendations on Toughness Requirements for Ferritic Materials," Welding Research Council Bulletin 175, August 1972 (WRC-175) is determined where:

$a = \frac{1}{4}(t_n^2 + t_v^2)^{1/2}$	[[
t_n = thickness of nozzle	
t_v = thickness of vessel	
r_n = apparent radius of nozzle = $r_i + 0.29*r_c$	
r_i = actual inner radius of nozzle	
r_c = nozzle radius (nozzle corner radius)	]]

Therefore, a/r_n = [[ ]]. The value F (a/r_n), taken from Figure A5-1 of WRC-175 for an [[ ]]. Including the safety factor of 1.5, the stress intensity factor, K_I, is $1.5\ \sigma\ (\pi a)^{1/2} * F(a/r_n)$:

Fermi 2 Plant-Specific Nominal K_I = 1.5 * [[ ]]

A detailed upper vessel example calculation for core not critical conditions is provided in Section 4.3.2.1.4 of NEDC-33178P-A. Section 4.3.2.1.3 of NEDC-33178P-A, presents the [[ ]] for the FW nozzle evaluation upon which the baseline non-shifted upper vessel PT curve is based. It can be seen that the nominal K_I from this evaluation is [[ ]]. Therefore, it has been shown that the nominal K_I for the Fermi 2-specific FW nozzle is less than the [[ ]] K_I, demonstrating applicability of the FW nozzle curve for Fermi 2.