January 25, 2006

Mr. Donald K. Cobb
Assistant Vice President - Nuclear Generation
Detroit Edison Company
6400 North Dixie Highway
Newport, MI 48166

SUBJECT: FERMI 2 - ISSUANCE OF AMENDMENT RE: REACTOR COOLANT SYSTEM PRESSURE AND TEMPERATURE CURVES (TAC NO. MC6468)

Dear Mr. Cobb:

The Commission has issued the enclosed Amendment No. 168 to Facility Operating License No. NPF-43 for the Fermi 2 facility. The amendment revises the Technical Specifications (TSs) in response to your application dated March 17, 2005, as supplemented by letter dated April 15, 2005.

The amendment revises TS 3.4.10, "RCS Pressure and Temperature (P/T) Limits." Specifically, the amendment revises the P/T curves for the hydrostatic pressure test, non-nuclear heatup and cooldown, and nuclear (core critical) limits illustrated in TS Figure 3.4.10-1 with six recalculated separate curves for 24 and 32 effective full power years of reactor operation. In addition, the amendment also revises associated surveillance requirements.

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/

David H. Jaffe, Sr. 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. 168 to NPF-43
2. Safety Evaluation

cc w/encls: See next page

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cc:

David G. Pettinari
Legal Department
688 WCB
Detroit Edison Company
2000 2nd Avenue
Detroit, MI 48226-1279

Michigan Department of Environmental Quality
Waste and Hazardous Materials Division
Radiological Protection and Medical Waste Section
Nuclear Facilities Unit
Constitution Hall, Lower-Level North
525 West Allegan Street
P.O. Box 30241
Lansing, MI 48909-7741

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
6400 N Dixie Highway
Newport, MI 48166

Mr. M. V. Yudasz, Jr., Director
Monroe County Emergency Management Division
965 South Raisinville Road
Monroe, MI 48161

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
2443 Warrenville Road, Suite 210
Lisle, IL 60532-4352

Ronald W. Gaston
Manager, Nuclear Licensing
Detroit Edison Company
Fermi 2 - 200 TAC
6400 North Dixie Highway
Newport, MI 48166

Supervisor - Electric Operators
Michigan Public Service Commission
6545 Merchantile Way
P.O. Box 30221
Lansing, MI 48909

Wayne County Emergency Management Division
10250 Middlebelt Road
Detroit, MI 48242

DETROIT EDISON COMPANY

DOCKET NO. 50-341

FERMI 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 168
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 (DECo) dated March 17, 2005, as supplemented by letter dated April 15, 2005, 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:

   <u>Technical Specifications and Environmental Protection Plan</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 168, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo 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 of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Timothy Kobetz, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: January 25, 2006

ATTACHMENT TO LICENSE AMENDMENT NO. 168

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the 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
3.4-24	3.4-24
3.4-25	3.4-25
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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 168 FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI 2

DOCKET NO. 50-341

1.0 INTRODUCTION

By application to the U.S. Nuclear Regulatory Commission (NRC, Commission) dated March 17, 2005 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML050830480), as supplemented by letter dated April 15, 2005 (ADAMS Accession No. ML051230442), the Detroit Edison Company (the licensee) requested changes to its Technical Specifications (TSs) for Fermi 2. The supplement dated April 15, 2005, provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the NRC staff's original proposed no significant hazards determination as published in the *Federal Register* on April 26, 2005 (70 FR 21453).

The proposed amendment would revise TS 3.4.10, "RCS [Reactor Coolant System] Pressure and Temperature (P/T) Limits." Specifically, the proposed amendment would revise the P/T curves for the hydrostatic pressure test, non-nuclear heatup and cooldown, and nuclear (core critical) limits illustrated in TS Figure 3.4.10-1 with six recalculated separate curves for 24 and 32 effective full power years (EFPY) of reactor operation. In addition, the amendment would revise associated surveillance requirements (SRs).

2.0 REGULATORY EVALUATION

The NRC staff evaluated the acceptability of the licensee's proposed P/T limit curves based on the following regulations and guidance:

Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.60(a) states:

> Except as provided in paragraph (b) of this section, all light-water nuclear power reactors, other than reactor facilities for which the certifications required under §50.82(a)(1) have been submitted, must meet the fracture toughness and material surveillance program requirements for the reactor coolant pressure boundary set forth in appendices G and H to this part.

Appendix G to 10 CFR Part 50, "Fracture Toughness Requirements," requires that facility P/T limit curves for the reactor pressure vessel (RPV) be at least as conservative as those obtained by applying the methodology of Appendix G to Section XI of the American Society of Mechanical Engineers *Boiler and Pressure Vessel Code* (ASME Code). 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 Appendix G to 10 CFR Part 50, is the 2001 Edition through the 2003 Addenda of the ASME Code. This edition of Appendix G to Section XI of the ASME Code incorporates the provisions of ASME Code Cases N-588, “Alternative to Reference Flaw Orientation of Appendix G for Circumferential Welds in Reactor Vessels, Section XI, Division 1,” and N-640, “Alternative Reference Fracture Toughness for Development of P-T Curves for ASME Section XI, Division 1.” In addition, Appendix G to 10 CFR Part 50 imposes minimum head flange temperatures when system pressure is above 20 percent of the preservice hydrostatic test pressure.

Appendix H to 10 CFR Part 50, “Reactor Vessel Material Surveillance Program Requirements,” establishes requirements related to facility RPV material surveillance programs. NRC Regulatory Guide (RG) 1.190, “Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence,” describes methods and assumptions acceptable to the NRC staff for determining the pressure vessel neutron fluence. RG 1.99, Revision 2, “Radiation Embrittlement of Reactor Vessel Materials,” contains methodologies for determining the increase in transition temperature resulting from neutron radiation.

Generic Letter (GL) 92-01, Revision 1, “Reactor Vessel Structural Integrity,” requested that licensees submit their plant’s RPV data for NRC staff review. GL 92-01, Revision 1, Supplement 1, requested that licensees provide and assess data from other licensees that could affect their RPV integrity evaluations.

NUREG-0800, “Standard Review Plan,” Section 5.3.2, “Pressure Temperature Limits and Pressurized Thermal Shock,” provides guidance on using these regulations and documents in the NRC staff’s review. In addition, Section 5.3.2 provides guidance to the NRC staff in performing check calculations of the licensee’s submittal.

Appendix A to 10 CFR Part 50, “General Design Criteria [GDC] for Nuclear Power Plants,” Criterion 30, “Quality of reactor coolant pressure boundary,” and Criterion 31, “Fracture prevention of reactor coolant pressure boundary,” establish requirements for pressure vessel fluence calculations. The NRC staff has approved vessel fluence calculation methodologies which satisfy the requirements of GDC 30 and 31 and adhere to RG 1.190 guidance. Fluence calculations are acceptable if they are done with approved methodologies or with methods which are shown to conform to RG 1.190 guidance.

3.0 TECHNICAL EVALUATION

The NRC staff has reviewed the licensee’s technical and regulatory analyses in support of its proposed license amendment.

3.1 Assessment of Neutron Fluence Levels

The NRC staff reviewed the licensee’s submittal to establish that the pressure vessel fluence used in the calculation of the vessel material properties for the estimation of the pressure vessel P/T limit curves is acceptable. The current Fermi 2 P/T limit curves were approved in Amendment No. 87 and their estimated applicability is to December 31, 2005. The licensee provided new vessel fluence calculations for both the 24 and 32 EFPY.

The licensee submitted General Electric Company (GE) Document No. GE-NE-0000-0031-6254-R1, “DTE Energy Fermi 2 Energy Center Neutron Flux Evaluation,” dated February 2005, which describes in detail the methodology, and justifies the input to the calculations in view of RG 1.190 guidance. The methodology is based on GE Document No. NEDC-32983-P-A, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluation,” that the NRC approved in a letter to GE dated September 14, 2001 (ADAMS Accession No. ML012400381). The method is based on a discrete ordinates method and cross sections derived from the ENDF/B-VI cross section file as recommended in RG 1.190. The transport approximations are P_3 for the inelastic scattering approximation and an acceptable value for the angular quadrature approximation.

Fermi 2 is currently in fuel cycle 11 with the next refueling outage scheduled for March 2006. Fermi 2 implemented a 3293 to 3430 megawatts thermal (MWt) power uprate at the end of the third fuel cycle. The Fermi 2 RPV flux for the first 10 cycles of operation was determined utilizing cycle 6 as a representative cycle with a thermal power limit of 3430 MWt, which is conservative due to the fact that the reactor operated at 3293 MWt for the first three cycles. An additional conservative factor of 10 percent was added to the cycle 6 RPV flux. The total energy generated by the Fermi 2 reactor for cycles 1 through 10 is equivalent to 12 EFPY at a 3430 MWt licensed thermal power. Regarding future plant operations, the fluence calculations conservatively assumed the reactor would operate at 3952 MWt (the planned extended power uprate (EPU)) for the remaining cycles until the end of the 40-year license in 2025. The fluence calculations assumed that the total energy at that time would be equivalent to 32 EFPY. This is conservative because the current licensed power level is limited to 3430 MWt. The NRC staff determined that the source computation is conservative and, therefore, it is acceptable.

The core operating conditions were retrieved from plant engineering data, including material compositions and core void fractions. The core, shroud, vessel, jet pump and surveillance capsule geometry and location were derived from design data. The fuel location and fuel type reflect actual design core loadings for the pre-EPU and EPU phase of the operation. The geometric representation of the core and its surroundings is in excellent agreement with the actual plant geometry and, therefore, it is acceptable.

The neutron source distribution was based on integrated energy production in conjunction with isotopic density distribution and corresponding fission energy and fission yield data. The calculated peak, inside vessel surface fluence for 24 EFPY, is 7.13×10^{17} neutrons/centimeter squared (n/cm^2), and for 32 EFPY is 9.68×10^{17} n/cm^2. Because these values were calculated in accordance with RG 1.190 guidance as discussed above, they are acceptable.

3.2 P/T Limit Curve Assessment

The NRC staff evaluated the acceptability of the licensee’s proposed Fermi 2 P/T limit curves that provide new limits that are valid for 24 and 32 EFPY of operation.

The methodology in Appendix G to Section XI of the ASME Code postulates the existence of a sharp surface flaw normal to the direction of the maximum applied stress for axial welds, plates, and forgings. A sharp surface flaw parallel to the weld is postulated for the evaluation of circumferential welds. For materials in the beltline, upper head, and lower head regions of the RPV, the maximum flaw size is postulated to have a depth that is equal to one-fourth of the thickness and a length equal to 1½ times the wall thickness. Thus, the critical locations in the

RPV beltline and head regions are the 1/4-thickness (1/4T) and 3/4-thickness (3/4T) locations, which correspond to the crack tips of the postulated inside and outside surface flaws of the vessel. For RPV nozzles, the surface flaw is postulated to be parallel to the axis of the nozzle's corner radius.

The basic parameter, in Appendix G to Section XI of the ASME Code, for calculating P/T limit curves is the stress intensity factor, K_I, which is a function of the stress state and flaw configuration. Beginning with the 1999 Addenda to the 1998 Edition, the static crack initiation fracture toughness curve, K_{IC}, has been provided in Figure G-2210-1 in Appendix G to Section XI of the ASME Code. The axes in Figure G-2210-1 are K_{IC} and T-RT_{NDT}, where T is temperature and RT_{NDT} is the reference temperature of the material. For beltline materials, the RT_{NDT} is increased due to neutron radiation embrittlement. This value is described as an adjusted reference temperature (ART).

The methodology in Appendix G to Section XI of the ASME Code requires that licensees determine the ART value at the maximum postulated flaw depth of beltline materials. The ART value is defined as the sum of the initial (unirradiated) reference temperature (initial RT_{NDT}), the mean value of the adjustment in reference temperature caused by irradiation ($^{a}RT_{NDT}$), and a margin (M) term. The $^{a}RT_{NDT}$ is the product of a chemistry factor (CF) and a fluence factor. The CF is dependent upon the amount of copper and nickel in the material and may be determined from tables in RG 1.99, Revision 2 or from surveillance data. The fluence factor is dependent upon the neutron fluence at the maximum postulated flaw depth. The M term is dependent upon whether the initial RT_{NDT} is a plant-specific or a generic value and whether the CF was determined using the tables in RG 1.99, Revision 2 or from surveillance data.

The M term is used to account for uncertainties in the values of initial RT_{NDT}, copper and nickel contents, fluence, and calculational procedures. RG 1.99, Revision 2 describes the methodology used in calculating the M term.

The methodology in Appendix G to Section XI of the ASME Code requires that P/T limit curves must satisfy a factor of 2.0 on stress intensity factors arising from primary membrane and bending stresses during normal plant operation (including heatups, cooldowns, and anticipated transients), and a factor of 1.5 on stress intensity factors arising from primary membrane and bending stresses when leak rate or hydrostatic pressure tests are performed on the RCS. Table 1 of Appendix G to 10 CFR Part 50 provides the NRC staff's criteria for meeting the P/T limit requirements and the minimum temperature requirements for bolting up the vessel during normal and pressure testing operations. Table 1 of Appendix G to 10 CFR Part 50 also identifies P/T limit requirements based on the RT_{NDT} of the materials in the closure flange region which is highly stressed by the bolt preload.

The proposed amendment would replace the following TS figures with P/T limit curves valid to 24 and 32 EFPY of operation:

Applicable to 24 EFPY

Figure 3.4.10-1	System Pressure Test P/T Curve (Curve A)
Figure 3.4.10-2	Core Not Critical P/T Curve (Curve B)
Figure 3.4.10-3	Core Critical P/T Curve (Curve C)

Applicable to 32 EFPY

Figure 3.4.10-4	System Pressure Test P/T Curve (Curve A)
Figure 3.4.10-5	Core Not Critical P/T Curve (Curve B)
Figure 3.4.10-6	Core Critical P/T Curve (Curve C)

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 licensee's ART calculation is in accordance with RG 1.99, Revision 2. For P/T calculations, the licensee used the 1998 Edition of the ASME Code, including 2000 Addenda, which is equivalent to the 2001 Edition through 2003 Addenda of the ASME Code. This version of the ASME Code has been endorsed in 10 CFR 50.55a. The P/T limit curve methodology used K_{IC} from Figure A-4200-1 of Section XI in Appendix A to determine the relationship between fracture toughness and T-RT_{NDT}. P/T limit curves were developed using the geometry of the RPV shells and discontinuities, the initial RT_{NDT} of the RPV materials, and the ART for the beltline materials. As a conservative simplification, the thermal gradient stress at the 1/4T location is assumed to be tensile for both heatup and cooldown, thereby resulting in the approach of applying the maximum tensile stress at the 1/4T location. The licensee stated that this approach is conservative because (1) the stresses at the 1/4T location during heatup are assumed to be tensile, and (2) radiation effects cause the allowable toughness at 1/4T to be less than that at 3/4T for a given metal temperature.

In addition to the beltline considerations, limits related to non-beltline discontinuities such as nozzles, penetrations, and flanges influence the construction of P/T limit curves. The non-beltline limits were included to allow monitoring of the vessel bottom head and upper vessel regions, separate from the beltline region, to help minimize heating requirements prior to pressure testing.

3.2.1 Bottom Head Curves

Bottom head curves are utilized because the water in the vessel lower head is separated from the water in contact with the vessel beltline and upper head regions by the reactor baffle plates. The water in the regions above the baffle plate is heated by decay heat from the reactor core, while the water in the lower head is at a lower temperature due to the injection of control rod drive (CRD) water for vessel pressurization. With little or no circulation through the recirculation pump loops, these regions are therefore maintained at different temperatures during inservice leak and hydrostatic testing and non-nuclear heatup/cooldown conditions.

The bottom head thermal transient and pressure stresses were developed using the limiting normal and upset conditions and generic boiling water reactor (BWR)/6 dimensions. Generic P/Ts for the bottom head curves were determined using the applied stress intensity factor (K_I) based on a revised finite element analysis that is described in Appendix H of the licensee's March 17, 2005, submittal. The generic bottom head curves have been adjusted to reflect the highest RT_{NDT} value of 30 EF for the bottom head plates and welds.

Evaluations were also performed to determine whether the CRD nozzle analyzed in the earlier finite element analysis was the limiting location. The licensee concluded that these analyses showed that the generic BWR/6 P/T limit curve, indexed to the RT_{NDT} of the limiting bottom head material, is conservative when applied to the Fermi 2 bottom head.

The NRC staff concluded that the licensee's evaluation for the bottom head curves is acceptable because the licensee showed that all discontinuities in the bottom head region were bounded by the CRD discontinuity, and that stresses in the generic BWR/6 vessel were higher than those in the Fermi 2 vessel.

3.2.2 Upper Vessel, Flange, and Beltline Region Curves

The P/T limit curves for inservice leak and hydrostatic testing and non-nuclear heatup/cooldown operations were developed from curves based on the material properties for the upper vessel (including feedwater nozzle), vessel flange, and vessel beltline regions. The P/T limit curve for core-critical operations, however, did not consider feedwater nozzles. Since the bottom head curves are less limiting than the upper vessel, vessel flange, and beltline region curves, the bottom head curves are not utilized for developing the core-critical operations curve. Using the highest RT_{NDT} for the materials in the beltline, upper vessel, and closure flange regions, the licensee developed P/T limit curves to meet the criteria in 10 CFR Part 50, Appendix G and ASME Code, Section XI, Appendix G.

The upper vessel region P/T limits were based on analysis of the feedwater nozzle and beltline regions. The K_I for the feedwater nozzle during pressure test conditions was computed using the methods from Weld Research Council (WRC) Bulletin 175 together with the geometry from a feedwater nozzle. Since Appendix G to Section XI of the ASME Code indicates that the methods used in WRC Bulletin 175 are acceptable for analyzing the inside corner of a nozzle and cylindrical shell for elastic stresses due to internal pressure stress, the method of analysis proposed by the licensee for the upper vessel and feedwater nozzle will satisfy the requirements of Appendix G to 10 CFR Part 50. The K_I for the upper vessel curve during normal operation was determined using the primary and secondary stresses from a feedwater nozzle finite element analysis and a membrane stress intensity factor, M_m, based on the values identified in Appendix G to Section XI of the ASME Code for a postulated surface flaw normal to the direction of maximum stress. The P/Ts for the upper vessel curve included the limiting feedwater transient for normal and upset conditions.

The beltline region P/T limits were based on the ART for the limiting materials in the beltline of the Fermi 2 RPV. The limiting beltline material is the lower shell axial weld (heat 13253 and heat 12008). Heat CE-2(WM), which has been determined to be heat 13253,12008 is the heat for the surveillance weld material as defined by the BWR integrated surveillance program (ISP). For this material, the CF was adjusted by multiplying the ISP least-squares fit CF developed in accordance with RG 1.99, Revision 2, as reported in BWRVIP-102, by the ratio of the CF for the vessel weld chemistry to the CF for the ISP surveillance chemistry. This results in an adjusted CF of 354.5 [326.96*(224/206.6)=354.5]. The key parameters for the licensee's ART determination for the 1/4T location are shown as follows:

Plate/Heat	EFPY	Adjusted CF	Init. RT_{NDT}	1/4T Fluence	$^{a}RT_{NDT}$	Margin	ART
CE-2(WM)(13253, 12008)	24	354.5	-44 EF	3.04×10^{17} n/cm^2	78 EF	28 EF	62 EF
CE-2(WM)(13253, 12008)	32	354.5	-44 EF	4.06×10^{17} n/cm^2	93 EF	28 EF	77 EF

The fluence values were calculated using a computer code that was based on an NRC-approved methodology. The fluence calculation was approved by the NRC staff in a memorandum dated May 2, 2005 (ADAMS Accession No. ML005123002).

The P/T limit curves apply to both heatup and cooldown and for both 1/4T and 3/4T locations because the maximum tensile stress of either heatup or cooldown is applied at the 1/4T location. The NRC staff's assessment also included an independent calculation of the ART value for the 1/4T location of Fermi 2 RPV beltline regions. For the evaluation of the limiting beltline materials, the NRC staff confirmed that the ARTs were based on the methodology of RG 1.99, Revision 2. The P/T limit curve methodology used the formula in Appendix G to Section XI to determine the K_I values.

The licensee stated that the feedwater nozzle, which experiences feedwater flow that is colder relative to the vessel coolant, was selected to represent all nozzles for determining fracture toughness in the upper shell region. The highest non-beltline RT_{NDT} for the feedwater nozzle at Fermi 2 is 12 EF. However, the licensee has increased the RT_{NDT} to 25 EF to consider the stresses in the recirculation inlet nozzle. The generic curve applied to the Fermi 2 upper vessel is modified for upper vessel by shifting the pressure versus (T-RT_{NDT}) values to reflect the RT_{NDT} value of 25 EF.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes a minimum temperature at the closure head flange based on the reference temperature for the flange material. Section IV.A.2 of Appendix G states that when the pressure exceeds 20 percent of the preservice system hydrostatic test pressure, the temperature of the closure flange regions (highly stressed by the bolt preload) must exceed the reference temperature of the material in those regions by at least 120 EF for normal operation and by 90 EF for hydrostatic pressure tests and leak tests. Based on the flange RT_{NDT} of 12 EF for Fermi 2, the NRC staff has determined that the proposed P/T limits have satisfied the requirement for the closure flange region during normal operation and inservice leak and hydrostatic testing.

3.2.3 Changes to Associated Surveillance Requirements

In addition to the introduction of six P/T curves, the license amendment request proposed P/T changes to surveillance requirements (SR) 3.4.10.1 and SR 3.4.10.2 for RCS limits. The requested changes to SR 3.4.10.1 and SR 3.4.10.2 would reflect the replacement of one P/T curve with six curves by changing the figure numbers to be used in performance of the surveillances. These changes would make the SRs reference the correct figures and are therefore acceptable.

The licensee requested changes to the reactor vessel flange and head flange temperatures stated in SR 3.4.10.7, SR 3.4.10.8, and SR 3.4.10.9. The changes would reflect the new minimum temperatures indicated on new Figures 3.4.10.1 through 6. As stated earlier in this SE, the methodology used to create the figures is acceptable to the NRC, therefore, it is acceptable to use the information in the figures as a new minimum value in the SRs.

3.3 Technical Evaluation Conclusion

The NRC staff has reviewed the licensee's submittal, and based on the review discussed above, the NRC staff finds the proposed changes to be acceptable.

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

This 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 with respect to surveillance requirements. The NRC staff has determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding published April 26, 2005 (70 FR 21453). Accordingly, this 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 this amendment.

6.0 CONCLUSION

The NRC staff 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) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors: N. Ray
L. Lois

Date: January 25, 2006