December 27, 1991

Docket No. 50-341

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Mr. William S. Orser
Senior Vice President - Nuclear
Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

Dear Mr. Orser:

SUBJECT: FERMI-2 - AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. NPF-43 (TAC NO. M81232)

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the Plant Technical Specifications (TS) in response to your letter dated February 21, 1991.

The amendment revises the Pressure Temperature Curves in the TS in accordance with Regulatory Guide 1.99, Revision 2.

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,

/s/

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 77 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

OFC	:LA:PDIII-1	:PM:PDIII-1	:OGC	:D:PDIII-1	:
NAME	:PShuttleworth	:JStang:jkd	:M ZOBLER	:LMarsh	:
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Mr. William Orser
Detroit Edison Company

Fermi-2 Facility

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

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. NPF-43

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

   A. The application for amendment by the Detroit Edison Company (the licensee) dated February 21, 1991, 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:

   Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 77, 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.

FOR THE NUCLEAR REGULATORY COMMISSION

Timothy G. Colburn

L. B. Marsh, Director
Project Directorate III-1
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 27, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change.

REMOVE	INSERT
xxi	xxi
xxv	xxv
3/4 4-19	3/4 4-19
3/4 4-20	3/4 4-20
3/4 4-21	3/4 4-21
3/4 4-22	3/4 4-22
B 3/4 4-3*	B 3/4 4-3*
B 3/4 4-4	B 3/4 4-4
B 3/4 4-5	B 3/4 4-5
B 3/4 4-6	B 3/4 4-6
B 3/4 4-7	B 3/4 4-7

* Overleaf page provided to maintain document completeness.
No changes contained on this page.

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LIST OF FIGURES

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LIST OF TABLES (Continued)

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

REACTOR COOLANT SYSTEM

LIMITING CONDITION FOR OPERATION

3.4.6.1 The reactor coolant system temperature and pressure shall be limited in accordance with the limit lines shown on Figure 3.4.6.1-1 (1) Curve A for hydrostatic or leak testing; (2) Curve B for heatup by non-nuclear means, cooldown following a nuclear shutdown and low power PHYSICS TESTS; and (3) Curve C for operations with a critical core other than low power PHYSICS TESTS, with:

a. A maximum heatup of 100°F in any 1-hour period,

b. A maximum cooldown of 100°F in any 1-hour period,

c. A maximum temperature change of less than or equal to 20°F in any 1-hour period during inservice hydrostatic and leak testing operations above the heatup and cooldown limit curves, and

d. The reactor vessel flange and head flange temperature greater than or equal to 71°F when reactor vessel head bolting studs are under tension.

APPLICABILITY: At all times.

ACTION:

With any of the above limits exceeded, restore the temperature and/or pressure to within the limits within 30 minutes; perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system; determine that the reactor coolant system remains acceptable for continued operations or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.1.1 During system heatup, cooldown and inservice leak and hydrostatic testing operations, the reactor coolant system temperature and pressure shall be determined to be within the above required heatup and cooldown limits and to the right of the limit lines of Figure 3.4.6.1-1 Curves A, B, or C, as applicable, at least once per 30 minutes.

SURVEILLANCE REQUIREMENTS (Continued)

4.4.6.1.2 The reactor coolant system temperature and pressure shall be determined to be to the right of the criticality limit line of Figure 3.4.6.1-1 Curve C within 15 minutes prior to the withdrawal of control rods to bring the reactor to criticality and at least once per 30 minutes during system heatup.

4.4.6.1.3 The reactor vessel material surveillance specimens shall be removed and examined, to determine changes in reactor pressure vessel material properties, as required by 10 CFR Part 50, Appendix H in accordance with the schedule in Table 4.4.6.1.3-1. The results of these examinations shall be used to update the curves of Figure 3.4.6.1-1.

4.4.6.1.4 The reactor vessel flange and head flange temperature shall be verified to be greater than or equal to 71°F:

a. In OPERATIONAL CONDITION 4 when reactor coolant system temperature is:

   1. $\leq$ 100°F, at least once per 12 hours.

   2. $\leq$ 80°F, at least once per 30 minutes.

b. Within 30 minutes prior to and at least once per 30 minutes during tensioning of the reactor vessel head bolting studs.

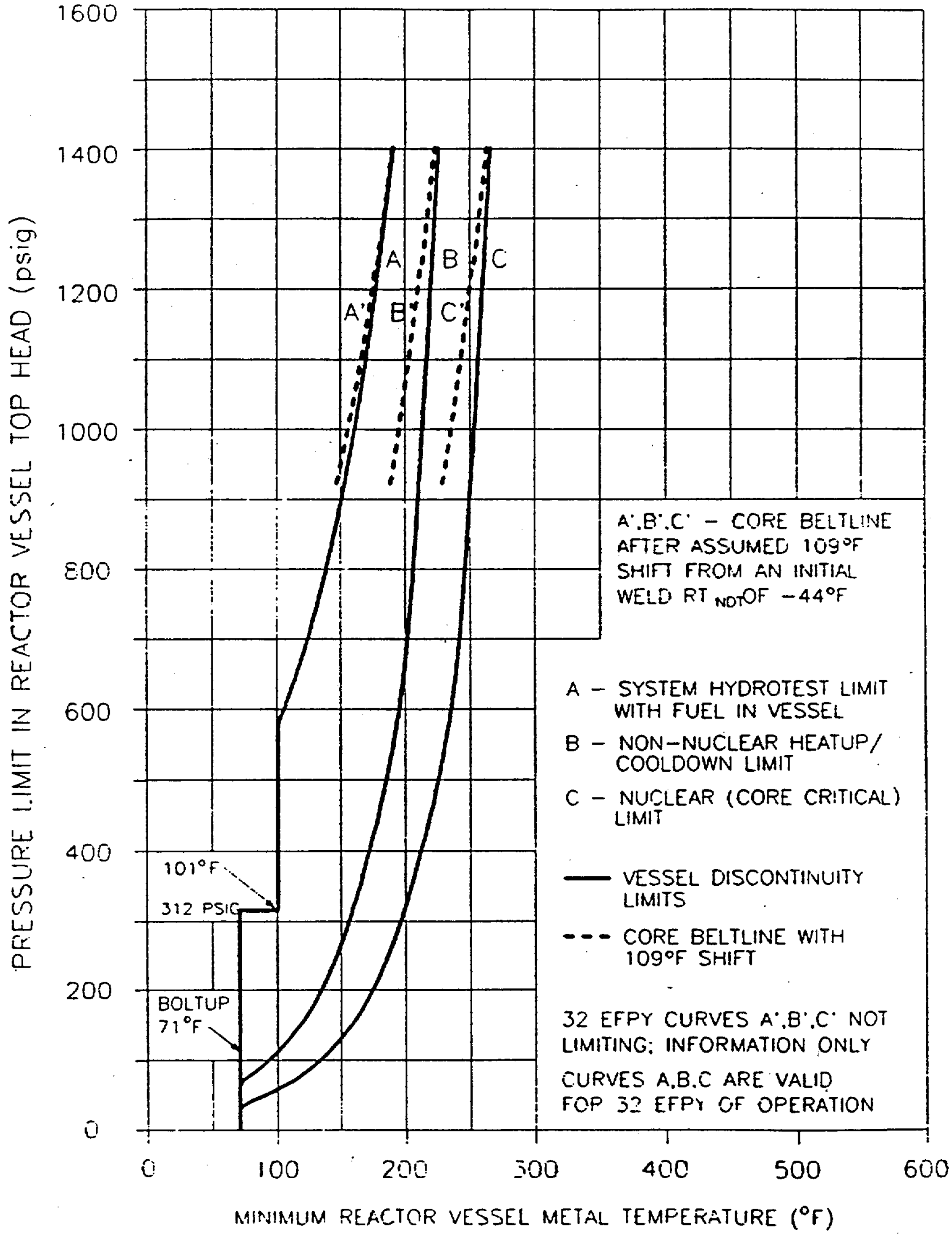

FIGURE 3.4.6.1-1

MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE
VERSUS REACTOR VESSEL PRESSURE

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

CAPSULE NUMBER	VESSEL LOCATION	LEAD FACTOR	WITHDRAWAL TIME (EFPY)
1	Azimuth 300	0.94	8
2	Azimuth 120	0.94	24
3	Azimuth 30	0.94	Standby

3/4.4.5 SPECIFIC ACTIVITY

The limitations on the specific activity of the primary coolant ensure that the 2 hour thyroid and whole body doses resulting from a main steam line failure outside the containment during steady state operation will not exceed small fractions of the dose guidelines of 10 CFR 100. The values for the limits on specific activity represent interim limits based upon a parametric evaluation by the NRC of typical site locations. These values are conservative in that specific site parameters, such as site boundary location and meteorological conditions, were not considered in this evaluation.

The ACTION statement permitting POWER OPERATION to continue for limited time periods with the primary coolant's specific activity greater than 0.2 microcuries per gram DOSE EQUIVALENT I-131, but less than or equal to 4.0 microcuries per gram DOSE EQUIVALENT I-131, accommodates possible iodine spiking phenomenon which may occur following changes in THERMAL POWER. In accordance with Generic Letter 85-19, the results of specific activity analyses in which primary coolant exceeds the limits of Specification 3.4.5 will be included in the Annual Report (Specification 6.9.1.5.d).

Information obtained on iodine spiking will be used to assess the parameters associated with spiking phenomena. A reduction in frequency of isotopic analysis following power changes may be permissible if justified by the data obtained.

Closing the main steam line isolation valves prevents the release of activity to the environs should a steam line rupture occur outside containment. The surveillance requirements provide adequate assurance that excessive specific activity levels in the reactor coolant will be detected in sufficient time to take corrective action.

3/4.4.6 PRESSURE/TEMPERATURE LIMITS

All components in the reactor coolant system are designed to withstand the effects of cyclic loads due to system temperature and pressure changes. These cyclic loads are introduced by normal load transients, reactor trips, and startup and shutdown operations. The various categories of load cycles used for design purposes are provided in Section 5.2 of the UFSAR. During startup and shutdown, the rates of temperature and pressure changes are limited so that the maximum specified heatup and cooldown rates are consistent with the design assumptions and satisfy the stress limits for cyclic operation.

During heatup, the thermal gradients in the reactor vessel wall produce thermal stresses which vary from compressive at the inner wall to tensile at the outer wall. These thermal induced compressive stresses tend to alleviate the tensile stresses induced by the internal pressure. Therefore, a pressure-temperature curve based on steady state conditions, i.e., no thermal stresses, represents a lower bound of all similar curves for finite heatup rates when the inner wall of the vessel is treated as the governing location.

The heatup analysis also covers the determination of pressure-temperature limitations for the case in which the outer wall of the vessel becomes the controlling location. The thermal gradients established during heatup produce tensile stresses which add to the pressure stresses already present. The thermal induced stresses at the outer wall of the vessel are tensile and are dependent on both the rate of heatup and the time along the heatup ramp; therefore, a lower bound curve similar to that described for the heatup of the inner wall cannot be defined. Subsequently, for the cases in which the outer wall of the vessel becomes the stress controlling location, each heatup rate of interest must be analyzed on an individual basis. Figure 3.4.6.1-1 was developed based on a heatup rate limit of 100°F/HR.

The reactor vessel materials have been tested to determine their initial RT_{NDT}. The results of these tests are shown in Section 5.2 of the UFSAR. Reactor operation and resultant fast neutron, E greater than 1 MeV, irradiation will cause an increase in the RT_{NDT}. Therefore, an adjusted reference temperature, based upon the fluence, nickel and copper content of the material in question, can be predicted using the recommendations of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement Of Reactor Vessel Materials." The pressure-temperature limit curve, Figure 3.4.6.1-1, Curves A′, B′ and C′, includes predicted adjustments for this shift in RT_{NDT} for the end of life fluence. However, Curves A, B, and C are more limiting than Curves A′, B′, and C′.

The actual shift in RT_{NDT} of the vessel material will be established periodically during operation by removing and evaluating, in accordance with ASTM E185-73 and 10 CFR 50, Appendix H, irradiated reactor vessel material specimens installed near the inside wall of the reactor vessel in the core area. Since the neutron spectra at the material specimens and vessel inside radius are essentially identical, the irradiated specimens can be used with confidence in predicting reactor vessel material transition temperature shift. The operating limit curves of Figure 3.4.6.1-1 shall be adjusted, as required, on the basis of the specimen data and recommendations of Regulatory Guide 1.99, Revision 2.

PRESSURE/TEMPERATURE LIMITS (Continued)

The pressure-temperature limit lines shown in Figure 3.4.6.1-1, Curves C and A, for reactor criticality and for inservice leak and hydrostatic testing have been provided to assure compliance with the minimum temperature requirements of Appendix G to 10 CFR Part 50 for reactor criticality and for inservice leak and hydrostatic testing.

The number of reactor vessel irradiation surveillance capsules and the frequencies for removing and testing the specimens in these capsules are provided in Table 4.4.6.1.3-1 to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

Double isolation valves are provided on each of the main steam lines to minimize the potential leakage paths from the containment in case of a line break. Only one valve in each line is required to maintain the integrity of the containment, however, single failure considerations require that two valves be OPERABLE. The surveillance requirements are based on the operating history of this type valve. The maximum closure time has been selected to contain fission products and to ensure the core is not uncovered following line breaks. The minimum closure time is consistent with the assumptions of the safety analyses to prevent pressure surges.

3/4.4.8 STRUCTURAL INTEGRITY

The inspection programs for ASME Code Class 1, 2, and 3 components ensure that the structural integrity of these components will be maintained at an acceptable level throughout the life of the plant.

Components of the reactor coolant system were designed to provide access to permit inservice inspections in accordance with Section XI of the ASME Boiler and Pressure Vessel Code 1974 Edition and Addenda through Summer, 1975.

The inservice inspection program for ASME Code Class 1, 2, and 3 components will be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the NRC pursuant to 10 CFR 50.55a(g)(6)(i).

3/4.4.9 RESIDUAL HEAT REMOVAL

A single shutdown cooling mode loop provides sufficient heat removal capability for removing core decay heat and mixing to assure accurate temperature indication, however, single failure considerations require that two loops be OPERABLE or that alternate methods capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation.

BASES TABLE B 3/4.4.6-1

THIS TABLE HAS BEEN DELETED

THIS FIGURE HAS BEEN DELETED

BASES FIGURE B 3/4.4.6-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated February 21, 1991, the Detroit Edison Company (DECo or the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment change provides pressure-temperature (P/T) limits for the reactor coolant system and the reactor pressure vessel for the Fermi-2 Technical Specifications, Section 3.4.6.1. The proposed P/T limits were requested for 32 effective full power years (EFPY) and were developed using Regulatory Guide (RG) 1.99, Revision 2. Generic Letter 88-11, "NRC Position on Radiation Embrittlement of Reactor Vessel Materials and Its Impact on Plant Operations," recommends RG 1.99, Rev. 2, be used in calculating P/T limits, unless the use of different methods can be justified. The P/T limits provide for the operation of the reactor coolant system during heatup, cooldown, criticality, and hydrotest.

To evaluate the P/T limits, the staff uses the following NRC regulations and guidance: Appendices G and H of 10 CFR Part 50; the ASTM Standards and the ASME Code, which are referenced in Appendices G and H; 10 CFR 50.36(c)(2); RG 1.99, Rev. 2; Standard Review Plan (SRP) Section 5.3.2; and Generic Letter 88-11.

Each licensee authorized to operate a nuclear power reactor is required by 10 CFR 50.36 to provide Technical Specifications for the operation of the plant. In particular, 10 CFR 50.36(c)(2) requires that limiting conditions of operation be included in the Technical Specifications. The P/T limits are among the limiting conditions of operation in the Technical Specifications for all commercial nuclear plants in the U.S. Appendices G and H of 10 CFR Part 50 describe specific requirements for fracture toughness and reactor vessel material surveillance that must be considered in setting P/T limits. An acceptable method for constructing the P/T limits is described in SRP Section 5.3.2.

Appendix G of 10 CFR Part 50 specifies fracture toughness and testing requirements for reactor vessel materials in accordance with the ASME Code and, in particular, that the beltline materials in the surveillance capsules be tested in accordance with Appendix H of 10 CFR Part 50. Appendix H, in turn, refers to ASTM Standards. These tests define the extent of vessel embrittlement at the time of capsule withdrawal in terms of the increase in reference temperature. Appendix G also requires the licensee to predict the effects of neutron irradiation on vessel embrittlement by calculating the adjusted reference temperature (ART) and

Charpy upper shelf energy (USE). Generic Letter 88-11 requested that licensees and permittees use the methods in RG 1.99, Rev. 2, to predict the effect of neutron irradiation on reactor vessel materials. This guide defines the ART as the sum of unirradiated reference temperature, the increase in reference temperature resulting from neutron irradiation, and a margin to account for uncertainties in the prediction method.

Appendix H of 10 CFR Part 50 requires the licensee to establish a surveillance program to periodically withdraw surveillance capsules from the reactor vessel. Appendix H refers to the ASTM Standards which, in turn, require that the capsules be installed in the vessel before startup and that they contain test specimens made from plate, weld, and heat-affected-zone (HAZ) materials of the reactor beltline.

2.0 EVALUATION

The staff evaluated the effect of neutron irradiation embrittlement on each beltline material in the Fermi-2 reactor vessel. The amount of irradiation embrittlement was calculated in accordance with RG 1.99, Rev. 2. The staff has determined that the material with the highest ART at 32 EFPY was the lower shell longitudinal welds 2-307A, B, and C with 0.26% copper (Cu), 0.87% nickel (Ni), and an initial RT_{ndt} of -10°F.

The licensee has not yet removed any surveillance capsule from Fermi-2 except that the flux wire dosimeter attached to Capsule 3 was removed. There are three specimen capsules in the surveillance program and the first capsule will be removed at 8 EFPY. The plant is at 2.5 EFPY in October 1991. All surveillance capsules contain Charpy impact specimens and tensile specimens made from base metal, weld metal, and HAZ metal.

For the limiting beltline material, weld 2-307A, B, and C, the staff calculated the ART to be 65.2°F at 1/4T (T = reactor vessel beltline thickness) and 26.2°F for 3/4T at 32 EFPY. The staff used a neutron fluence of 3.6E17 n/cm^2 at 1/4T and 1.7E17 n/cm^2 at 3/4T. The ART was determined by Section 1 of RG 1.99, Rev. 2, because no surveillance capsules have been removed from the Fermi-2 pressure vessel.

The licensee used the method in RG 1.99, Rev. 2, to calculate an ART of 65.3°F at 32 EFPY at 1/4T for the same limiting weld metal. The staff judges that the licensee's ART 65.3°F is more conservative than the staff's ART of 65.2°F, and it is acceptable. Substituting the ART of 65.2°F into equations in SRP 5.3.2, the staff verified that the proposed P/T limits for heatup, cooldown, and hydrotest meet the beltline material requirements in Appendix G of 10 CFR Part 50.

In addition to beltline materials, Appendix G of 10 CFR Part 50 also imposes P/T limits based on the reference temperature for the reactor vessel closure flange materials. Section IV.2 of Appendix G states that when the pressure exceeds 20% 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°F for normal operation and by 90°F for hydrostatic pressure tests and leak tests. Paragraph IV.A.3 of Appendix G states "an exception may be made for boiling water reactor vessels when water level is within the normal range for power operation and the pressure is less than 20 percent of the preservice system hydrostatic test pressure. In this case the minimum permissible temperature is 60°F (33°C) above the reference temperature of the closure flange regions that are highly stressed by the bolt preload." Based on the flange reference temperature of 11°F, the staff has determined that the proposed P/T limits satisfy Section IV.2 of Appendix G.

Section IV.B of Appendix G requires that the predicted Charpy USE at end of life be above 50 ft-lb. The material with the lowest unirradiated USE is weld 2-307A, B, and C with 67 ft-lb. The staff calculates that at 32 EFPY and at 1/4T, weld 2-307A, B, and C will have a USE of 50.4 ft-lb. This is greater than 50 ft-lb and, therefore, is acceptable.

Table 4.4.6.1.3-1 has been modified to reflect the lead factor used for the updated P-T curves. The lead factor better reflects the variation between the peak fluence location and capsule location. The staff has reveiwed the change and finds the change is acceptable since the new lead factor specified in consistent with those previously found to be acceptable in vessel configurations similar to Fermi 2 (e.g., Susquehanna 1 and Peach Bottom 2).

The licensee also proposes to change the TS Bases section B 3/4.4.6, Pressure-Temperature Limits. These changes are necessary to reflect the updated RG 1.99 Revision 2 methodology and the conclusions of this evaluation. In addition, editorial changes are made to correct references to the UFSAR and to eliminate detailed information typically located in the USFAR. These Bases changes are acceptable since they are either consistent with the revised methodology or editorial in nature.

The staff concludes that the proposed P/T limits for the reactor coolant system for heatup, cooldown, leak test, and criticality are valid through 32 EFPY because the limits conform to the requirements of Appendices G and H of 10 CFR Part 50. The proposed P/T limits also satisfy Generic Letter 88-11 because the method in RG 1.99, Rev. 2 was used to calculate the ART. Therefore, the proposed P/T limits may be incorporated into the Fermi-2 Technical Specifications.

3.0 REFERENCES

1. Regulatory Guide 1.99, Radiation Embrittlement of Reactor Vessel Materials, Revision 2, May 1988

2. NUREG-0800, Standard Review Plan, Section 5.3.2: Pressure-Temperature Limits

3. February 21, 1991, Letter from W. S. Orser (DE) to USNRC Document Control Desk, Subject: Proposed Technical Specification Change (License Amendment) - Pressure/Temperature Limits

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents which may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (56 FR 57694). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

6.0 CONCLUSION

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

Principal Contributor:
J. Tsao

Date: December 27, 1991

DATED: December 27, 1991

AMENDMENT NO. 77 TO FACILITY OPERATING LICENSE NO. NPF-43-FERMI-2

Docket File
NRC & Local PDRs
PDIII-1 Reading
Fermi Plant File
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