



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

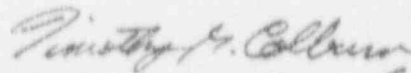
1. The Nuclear Regulatory Commission (the Commission) has found that:
 - A. The application for amendment by the Detroit Edison Company (the licensee) dated February 21, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules 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



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

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

INSERT

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

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

INDEX

LIST OF FIGURES

<u>FIGURE</u>		<u>PAGE</u>
3.1.5-1	SODIUM PENTABORATE VOLUME/ CONCENTRATION REQUIREMENTS.....	3/4 1-21
3.4.1.4-1	THERMAL POWER VS. CORE FLOW.....	3/4 4-5a
3.4.6.1-1	MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE.....	3/4 4-21
3.4.10-1	THERMAL POWER VS. CORE FLOW.....	3/4 4-31
4.7.5-1	SAMPLE PLAN 2) FOR SNUBBER FUNCTIONAL TEST.....	3/4 7-21
B 3/4 3-1	REACTOR VESSEL WATER LEVEL.....	B 3/4 3-7
B 3/4.6.2-1	LOCAL POOL TEMPERATURE LIMIT.....	B 3/4 6-5
B 3/4.7.3-1	ARRANGEMENT OF SHORE BARRIER SURVEY POINTS.....	B 3/4 7-6
5.1.1-1	EXCLUSION AREA.....	5-2
5.1.2-1	LOW POPULATION ZONE.....	5-3
5.1.3-1	MAP DEFINING UNRESTRICTED AREAS AND SITE BOUNDARY FOR RADIOACTIVE GASEOUS AND LIQUID EFFLUENTS.....	5-4

INDEX

LIST OF TABLES (Continued)

<u>TABLE</u>		<u>PAGE</u>
4.8.2.1-1	BATTERY SURVEILLANCE REQUIREMENTS.....	3/4 8-12
3.8.4.2-1	PRIMARY CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES.....	3/4 8-19
3.8.4.3-1	MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION.....	3/4 8-21
3.8.4.5-1	STANDBY LIQUID CONTROL SYSTEM ASSOCIATED ISOLATION DEVICES 480 V MOTOR CONTROL CENTERS	3/4 8-27
4.11.1.1.1-1	RADIOACTIVE LIQUID WASTE SAMPLING AND ANALYSIS PROGRAM.....	3/4 11-2
4.11.2.1.2-1	RADIOACTIVE GASEOUS WASTE SAMPLING AND ANALYSIS PROGRAM.....	3/4 11-9
3.12.1-1	RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM	3/4 12-3
3.12.1-2	REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES.....	3/4 12-9
4.12.1-1	DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS.....	3/4 12-10
5.7.1-1	COMPONENT CYCLIC OR TRANSIENT LIMITS.....	5-7
6.2.2-1	MINIMUM SHIFT CREW COMPOSITION.....	6-5

REACTOR COOLANT SYSTEM

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.

REACTOR COOLANT SYSTEM

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. $\leq 100^{\circ}\text{F}$, at least once per 12 hours.
 2. $\leq 80^{\circ}\text{F}$, 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.

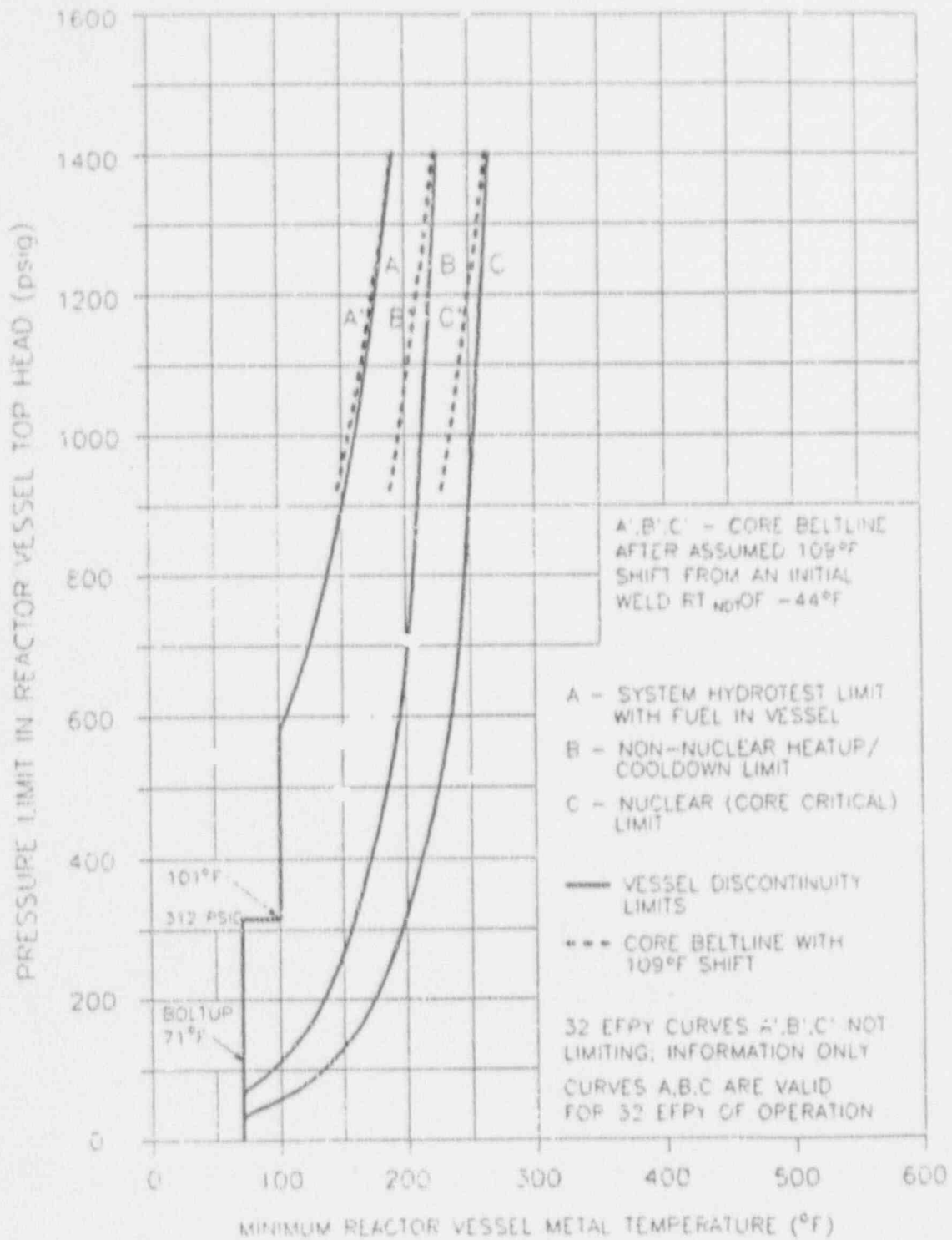


FIGURE 3.4.6.1-1

MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE
VERSUS REACTOR VESSEL PRESSURE

TABLE 4.4.6.1.3-1

REACTOR VESSEL MATERIAL SURVEILLANCE PROGRAM-WITHDRAWAL SCHEDULE

<u>CAPSULE NUMBER</u>	<u>VESSEL LOCATION</u>	<u>LEAD FACTORS</u>	<u>WITHDRAWAL TIME (DAYS)</u>
1	Azimuth 300	0.94	9
2	Azimuth 120	0.94	24
3	Azimuth 30	0.94	Standby

REACTOR COOLANT SYSTEM

BASES

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.

REACTOR COOLANT SYSTEM

BASES

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.

REACTOR COOLANT SYSTEM

BASES

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

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BASES FIGURE B 3/4.4.6-1