



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

DETROIT EDISON COMPANY
WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.14
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 January 6, 1988, 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 the Facility Operating License No. NPF-43 is hereby amended to read as follows:

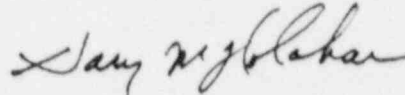
Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 14, 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.

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3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION



Gary M. Holahan, Assistant Director
for Regions III and V
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: January 12, 1988

ATTACHMENT TO LICENSE AMENDMENT NO.14

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

3/4 4-12

B 3/4 4-2

INSERT

3/4 4-12

B 3/4 4-2

B 3/4 4-2a

REACTOR COOLANT SYSTEM

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The reactor coolant system leakage shall be demonstrated to be within each of the above limits by:

- a. Monitoring the primary containment atmospheric gaseous radioactivity at least once per 4 hours,*
- b. Monitoring the primary containment sump flow rate at least once per 4 hours,
- c. Monitoring the drywell floor drain sump level at least once per 4 hours, and
- d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.*

4.4.3.2.2 Each reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 and verifying the leakage of each valve to be within the specified limit:

- a. At least once per 18 months, and
- b. Prior to returning the valve to service following maintenance, repair or replacement work on the valve which could affect its leakage rate.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

4.4.3.2.3 The high/low pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with alarm setpoints per Table 3.4.3.2-2 by performance of a:

- a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
- b. CHANNEL CALIBRATION at least once per 18 months.

*Not a means of quantifying leakage.

TABLE 3.4.3.2-1

REACTOR COOLANT SYSTEM PRESSURE ISOLATION VALVES

<u>VALVE NUMBER</u>	<u>VALVE DESCRIPTION</u>
1. RHR System	
E11-F015A	LPCI Loop A Injection Isolation Valve
E11-F015B	LPCI Loop B Injection Isolation Valve
E11-F050A	LPCI Loop A Injection Line Testable Check Valve
E11-F050B	LPCI Loop B Injection Line Testable Check Valve
E11-FU23	RPV Head Spray Outboard Isolation Valve
E11-F022	RPV Head Spray Inboard Isolation Valve
E11-F008	Shutdown Cooling RPV Suction Outboard Isolation Valve
E11-F009	Shutdown Cooling RPV Suction Inboard Isolation Valve
E11-F608	Shutdown Cooling Suction Isolation Valve
2. Core Spray System	
E21-F005A	Loop A Inboard Isolation Valve
E21-F005B	Loop B Inboard Isolation Valve
E21-F006A	Loop A Containment Check Valve
E21-F006B	Loop B Containment Check Valve
3. High Pressure Coolant Injection System	
E41-F007	Pump Discharge Outboard Isolation Valve
E41-F006	Pump Discharge Inboard Isolation Valve
4. Reactor Core Isolation Cooling System	
E51-F012	Pump Discharge Isolation Valve
E51-F013	Pump Discharge to Feedwater Header Isolation Valve

TABLE 3.4.3.2-2

REACTOR COOLANT SYSTEM INTERFACE VALVES

LEAKAGE PRESSURE MONITORS

<u>VALVE NUMBER</u>	<u>SYSTEM</u>	<u>ALARM SETPOINT (psig)</u>
E11-F015A & B, E11-F022, F023, E11-F050A & B	RHR LPCI	≤ 449
E11-F008, F009, F608	RHR Shutdown Cooling	≤ 135
E21-F005A & B, E21-F006A & B	Core Spray	≤ 452
E41-F006, F007	HPCI	≤ 71
E51-F012, F013	RCIC	≤ 71

3/4.4 REACTOR COOLANT SYSTEM

BASES

3/4.4.1 RECIRCULATION SYSTEM

Operation with one reactor core coolant recirculation loop inoperable is prohibited until an evaluation of the performance of the ECCS during one loop operation has been performed, evaluated, and determined to be acceptable.

An inoperable jet pump is not, in itself, a sufficient reason to declare a recirculation loop inoperable, but it does, in case of a design-basis-accident, increase the blowdown area and reduce the capability of reflooding the core; thus, the requirement for shutdown of the facility with a jet pump inoperable. Jet pump failure can be detected by monitoring jet pump performance on a prescribed schedule for significant degradation.

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles. Since the coolant in the bottom of the vessel is at a lower temperature than the coolant in the upper regions of the core, undue stress on the vessel would result if the temperature difference was greater than 145°F.

3/4.4.2 SAFETY/RELIEF VALVES

The safety valve function of the safety/relief valves operate to prevent the reactor coolant system from being pressurized above the Safety Limit of 1325 psig in accordance with the ASME Code. A total of 11 OPERABLE safety/relief valves is required to limit reactor pressure to within ASME III allowable values for the worst case upset transient.

Demonstration of the safety/relief valve lift settings will occur only during shutdown and will be performed in accordance with the provisions of Specification 4.0.5.

The low-low set system ensures that a potentially high thrust load (designated as load case C.3.3) on the SRV discharge lines is eliminated during subsequent actuations. This is achieved by automatically lowering the closing setpoint of two valves and lowering the opening setpoint of two valves following the initial opening. Sufficient redundancy is provided for the low-low set system such that failure of any one valve to open or close at its reduced setpoint does not violate the design basis.

REACTOR COOLANT SYSTEM

BASES

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of Regulatory Guide 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", May 1973.

3/4.4.3.2 OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining system leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shutdown to allow further investigation and corrective action. Service sensitive reactor coolant system Type 304 and 316 austenitic stainless steel piping; i.e., those that are subject to high stress or that contain relatively stagnant, intermittent, or low flow fluids, requires additional surveillance and leakage limits.

The purpose of the RCS interface valves leakage pressure monitors (LPMs) is to provide assurance of the integrity of the Reactor Coolant System pressure isolation valves which form a high/low pressure boundary. The LPM is designed to alarm on increasing pressure on the low pressure side of the high/low pressure interface to provide indication to the operator of abnormal interface valve leakage.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity thereby reducing the probability of gross valve failure and consequent intersystem LOCA. Leakage from the RCS pressure isolation valves is IDENTIFIED LEAKAGE and will be considered as a portion of the allowed limit.

3/4.4.4 CHEMISTRY

The water chemistry limits of the reactor coolant system are established to prevent damage to the reactor materials in contact with the coolant. Chloride limits are specified to prevent stress corrosion cracking of the stainless steel. The effect of chloride is not as great when the oxygen concentration in the coolant is low, thus the 0.2 ppm limit on chlorides is permitted during POWER OPERATION. During shutdown and refueling operations, the temperature necessary for stress corrosion to occur is not present so a 0.5 ppm concentration of chlorides is not considered harmful during these periods.

REACTOR COOLANT SYSTEM

BASES

CHEMISTRY (Continued)

Conductivity measurements are required on a continuous basis since changes in this parameter are an indication of abnormal conditions. When the conductivity is within limits, the pH, chlorides and other impurities affecting conductivity must also be within their acceptable limits. With the conductivity meter inoperable, additional samples must be analyzed to ensure that the chlorides are not exceeding the limits.

The surveillance requirements provide adequate assurance that concentrations in excess of the limits will be detected in sufficient time to take corrective action.