August 6, 1992

MEMORANDUM FOR: L. B. Marsh, Project Directorate (13D-18)
Project Directorate III-1
Division of Reactor Projects III, IV, V

FROM: Robert C. Jones, Chief
Reactor Systems Branch
Division of Systems Technology

SUBJECT: FERMI-2 PROPOSED LICENSE AMENDMENT POWER UPRATE REVIEW (TAC NO. M82102

Enclosed is the Reactor Systems Branch input to the Safety Evaluation Report being prepared by your Project Directorate for the subject power uprate license amendment. It is my understanding that you will use this and other technical branch inputs for developing the overall staff safety evaluation for this license amendment.

/s/

Robert C. Jones, Chief
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Division of Systems Technology

Enclosure:
As stated

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Introduction

Detroit Edison, the licensee for Fermi Unit 2, submitted a request by letter on September 24, 1991 to uprate the licensed power level from 3293 MWt to 3430 MWt. This represents approximately a 4.2% increase in thermal power with a 5% increase in rated steam flow. The planned approach to achieving the higher power level consists of (1) an increase in the core thermal power to create an increased steam flow, (2) a corresponding increase in feedwater flow, (3) no increase in maximum core flow, and (4) reactor operation primarily along extensions of current rod/flow control lines. This approach is consistent with the BWR generic power uprate guidelines presented in General Electric report NEDC 31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate," June 1991. The operating pressure will be increased approximately 25 psi to assure satisfactory pressure control and pressure drop characteristics for the increased steam flow. The increased core power will be achieved by utilizing a slightly flatter radial power distribution while still maintaining limiting fuel bundles within their constraints.

2.3.1 Power/Flow Operating Map

Power uprate raises the top portion of the operating map (power versus core flow) along the current rod/flow control lines. These lines have not changed, but have been renamed to reflect the redefinition of rated thermal power. Full power operation under Maximum Extended Operating Domain (MEOD) which was originally achieved at a minimum value of approximately 75% core flow will now be achieved at approximately 81% core flow along the same rod line. The absolute power at that point will be higher since full power is redefined.

2.4 Stability

The BWR Owner's Group and the NRC are addressing ways to minimize the occurrence and potential effects of power oscillations that have been observed for certain BWR operating conditions. Until long-term corrective actions are developed, the licensee has implemented the interim stability recommendations of General Electric in accordance with NRC Bulletin 88-07 and Supplement 1 to that bulletin, which restrict plant operation in the high power, low core flow region of the power/flow operating map.

2.5 Reactivity Control

2.5.1 Control Rod Drives and Scram Performance

The control rod drive (CRD) system was evaluated at the uprated steam flow and system pressure. Reactor pressure has little effect on scram insertion speed. The licensee evaluated the CRD system for insertion, withdrawal, and CRD cooling, and concluded that the CRD system will continue to carry out all its functions at uprated power. The license will continue to monitor, by various surveillance requirements, the scram time performance as required in the plant Technical Specifications to ensure that the original licensing basis for the scram system is preserved.

3.0 Reactor Coolant System and Connected Systems

3.1 Nuclear System Pressure Relief

The purpose of the nuclear system pressure relief is to prevent overpressurization of the nuclear system during abnormal operational transients. The plant safety/relief valves (SRV) provide this protection.

The only change in the nuclear system pressure relief for power uprate is an increase in SRV setpoints as described below. The nominal operating dome pressure will be increased by approximately 25 psi, therefore the SRV setpoints will be increased by a similar amount to provide adequate simmer margin.

3.2 Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor coolant pressure boundary is 1250 psig. The ASME code allowable pressure limit for pressurization events is 1375 psig. The licensee analyzed the limiting pressurization event which is an MSIV closure with failure of valve position scram. Four SRVs were assumed out of service and an initial operating pressure of 1045 psig was used in the analysis. The analysis also assumed 102% of 3420 MWt, 105% core flow, and a high flux scram. As expected, at the proposed uprated power level a higher peak pressure results than at the currenly licensed power level; but the peak pressure remains below the ASME code allowable limit. The calculated peak reactor coolant pressure boundary pressure is 1339 psig which is acceptable.

3.4 Reactor Recirculation System

The uprated power condition will be accomplished by operation along extensions of current rod lines on the power/flow map with no increase in the maximum core flow. It is expected that a small increase in flow resistance due to an increase in core average void fraction will occur when operating at maximum core flow. The licensee has committed to performing periodic surveillance tests to assure that the recirculation system will accommodate any changes in operating conditions due to operation at the maximum power uprate conditions. These tests will also assure that no undue vibration will occur at uprated power conditions.

3.7 Main Steamline Isolation Valves (MSIVs)

The performance of the MSIVs with regard to reactor coolant pressure boundary requirements such as closure time and leakage could be impacted by the increased operating pressure. However, the pressure increase is relatively small (less than 3%) and performance will be monitored by surveillance requirements in the plant Technical Specifications to ensure the original licensing basis for the MSIVs is preserved.

3.8 Reactor Core Isolation Cooling (RCIC)

The RCIC system provides core cooling when the reactor pressure vessel (RPV) is isolated from the main condenser, and the RPV pressure is greater than the maximum allowable for initiation of a low pressure cooling system. The licensee stated that they have assessed the RCIC system consistent with the bases and conclusions of generic evaluations of Section 4.2 in NEDC-31984P (LTR2). The licensee also committed to implement the recommendation of GE SIL 377, to add a small bypass around the steam admission valve to reduce the chances of turbine overspeed trips. The staff requires that licensees provide assurance that their RCIC system is capable of injecting its design flow at the conditions associated with power uprate and that the operability of the RCIC system will not be decreased because of the higher loads placed on the system, or because of any modifications made to the system. In response to a staff request, the licensee has committed to performance tests to ensure that the RCIC will function as designed at the uprated conditions by letter dated April 23, 1992. Successful completion of these tests should provide reasonable assurance that the performance of the RCIC system will not be decreased because of the higher loads placed on the system or because of any modifications made to the system to compensate for the increased loads.

3.9 Residual Heat Removal (RHR) System

The RHR system is design to restore and maintain the coolant inventory in the reactor vessel and to provide decay heat removal following reactor shutdown for both normal and post-accident conditions. The RHR is designed to operate in the low pressure injection (LPCI) mode, shutdown cooling mode, suppression pool cooling mode, and containment spray cooling mode. The LPCI mode is discussed elsewhere in this report.

The effect of power uprate on the shutdown cooling mode is to lengthen the time to reach the shutdown temperature (125°F) for the primary coolant. The licensee estimates that the time to reduce the coolant temperature to 125°F after steady state operation at uprated power is less than 14 hours. This is still within the design objective of the RHR to reach 125°F in approximately 20 hours.

The design bases for the suppression pool cooling mode is to ensure that the pool does not exceed 198°F immediately after a reactor blowdown. The licensee performed the analysis for a reactor blowdown at uprate power conditions to confirm that the suppression pool temperature will be equal or less than 198°F.

4.0 E[illegible]ered Safety Features

Emergency Core Cooling Systems (ECCS)

With the suppression pool temperature remaining below 198°F, the NPSH ECCS pump requirements are still satisfied for the limiting conditions of 0 psig containment pressure and the maximum expected temperature of pumped fluids will not change from the USAR licensing ba[illegible]

4.2.1 High Pre[illegible] [illegible] Injection (HPCI)

The HPCI system design basis is to provide reactor vessel inventory makeup during small and intermediate break loss-of-coolant accidents (LOCA) and isolation events. The HPCI is designed to provide its rated flow over a reactor pressure range of 150 psig to a maximum pressure based on the lowest SRV safety setpoint. The SRV opening setpoints will be increased for power uprate to maintain adequate simmer margin. Increasing the SRV setpoint pressure has a potential impact on the maximum operating pressure for the HPCI system.

The required flow rate remains unchanged. However, the HPCI pump and turbine operational requirements at uprated conditions are increased. The pump total dynamic head is increased by approximately three percent due to SRV setpoint increase. The speed and power requirements of the steam turbine are also increased.

The licensee adopted the assessment of turbine overspeeding as described in the generic topical report and has implemented GE SIL 480 for the HPCI system.

In response to a staff request, the licensee, by letter dated April 23, 1992 has committed to conducting performance tests to ensure HPCI can start and run, as designed, at uprated conditions. Successful completion of these tests should provide reasonable assurance that the operability of the HPCI system will not decrease because of higher loads placed on the system, or because of any modification made to the system to compensate for these increased loads.

4.2.2 RHR System (Low Pressure Coolant Injection)

The licensee has adopted the generic evaluation provided in the generic topical report for the LPCI mode of the RHR system. There are no changes associated with power uprate for the LPCI system.

4.2.3 Low Pressure Core Spray (LPCS) System

The licensee has adopted the bounding generic evaluation provided in the GE topical report for the LPCS system. The licensing and design flow rates plus the operating pressure will not be changed. Therefore, there is no impact on the LPCS system from power uprate.

4.3 ECCS Performance Evaluation

The ECCS performance under all LOCA conditions and their analysis models must satisfy the acceptance criteria and requirements of 10 CFR 50.46 and 10 CFR 50, Appendix K. The results of the ECCS/LOCA analysis using NRC approved methods are provided in later sections of this Safety Evaluation Report (SER).

9.0 Reactor Safety Performance Evaluatio

9.1 Reactor Transients

The limiting USAR transients were reevaluated using the GEMINI transient analysis methods with uprated power input parameters. The transients were analyzed at the uprated power and maximum allowed core flow point on the power/flow operating map for uprated operational conditions.

The current safety limit minimum critical power ratio (SLMCPR) was shown to be applicable for uprated conditions and then used to calculate the minimum critical power ratio (MCPR) operating limits. The limiting transient, Feedwater Controller Failure-Maximum Demand with Bypass failure and Moisture Separator Reheater Failure yielded the greatest change in CPR. This delta CPR added to the SLMCPR gives the operating limit minimum critical power ratio (OLMCPR).

9.3 Special Events

9.3.1 Anticipated Transients Without Scram (ATWS)

The licensee has committed to meeting the generic bounding analysis for ATWS events being performed generically by General Electric. The generic analysis is still under review by the staff. However, it will be acceptable if Fermi-2 meets the bounding generic ATWS analyses when they are approved by the staff.

9.3.2 Station Blackout (SBO)

Plant response and coping capabilities for a SBO event are impacted slightly by operation at uprated power level due to the increase in the operating temperature of the primary coolant system, increase in decay heat, and increase in main steam safety/relief valve setpoints. There are no changes to the systems and equipment used to respond to the SBO, nor is the coping time changed. The condensate water requirement also increases. However, the current condensate storage tank ensures that adequate water volume is available. The licensee evaluated the impact of power uprate on SBO for areas that contain equipement needed to mitigate the SBO event. The licensee concluded that the equipment needed for event mitigation is qualified for these higher temperatures. The licensee also determined that the systems needed to respond after power is restored are designed for the peak suppression pool temperature anticipated.

ECCS Performance Evaluation

The licensee used the staff approved SAFER/GESTR methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria.

A plant specific analysis was performed for Fermi-2 with the Cycle 3 fuel types.

The results of the break spectrum calculations show that the DBA recirculation line suction break with Division II battery failure is the limiting case. The nominal PCT is calculated to be 1002 F with a corresponding Appendix K PCT of 1397 F. The licensing basis PCT is caluclated to be 1602°F. The UBPCT is calculated to be 1351°F. The licensing basis PCT is less than 2200°F and the UBPCT is 250 F lower than the licensing basis PCT, therefore, the requirements of Appendix K are satisfied.

The licensee also reevaluated the ECCS performance for single loop operation (SLO) using the SAFER/GESTR LOCA methodology. The DBA size break is also limiting for SLO. Using the same assumptions in the SAFER/GESTR-LOCA calculation with no MAPLHGR reduction, yields a calculated nominal and Appendix K PCT of 1194°F and 1718°F, respectively. Since the PCT is below the 10 CFR 50.46 limit of 2200°F, no MAPLHGR reduction is required for SLO.

The Maximum Extended Operating Domain (MEOD) analysis and Maximum Extended Load Line Limit Analysis (MELLLA) provide an expanded operating rod line and an increased core flow range power-flow operating domain for Fermi-2. These analyses require a more restrictive initial MCPR and MAPLHGR/PLHGR and require MCPR and MAPLHGR multiplier factors to be imposed. These required power- and flow-dependent MCPR and MAPLHGR limits (with multipliers) bound the SLO power-flow condition to ensure that SLO PCTs during a postulated LOCA are below the normal two-loop operation calculated licensing basis PCTs. Additional clarifying information presented in a telephone call on July 15, 1992 provided assurance that the SLO uncertainties as applied in the SAFER/GESTAR methodology will also be less than the uncertainties for two loop operation. This is acceptable to the staff.