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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated September 24, 1991, and modified January 31, and April 30, 1992, the Detroit Edison Company (DECo or the licensee) requested an amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment would change the licensed thermal power level of the reactor from the current limit of 3293 megawatts thermal (MWt) to an increased limit of 3430 MWt. This request is in accordance with the generic boiling water reactor (BWR) power uprate program established by the General Electric Company (GE) and approved by the U.S. Nuclear Regulatory Commission (NRC) staff in a letter dated September 30, 1991. The licensee submitted additional information to supplement the application by letters dated February 24, March 23 and 26, April 23, May 11, and August 12 and 13, 1992 and by telephone calls on July 15 and 29, 1992. This information did not change the initial proposed no significant hazards consideration determination as noticed in the Federal Register on March 18, 1992 (57 FR 9442) and June 24, 1992 (57 FR 28198).

2.0 DISCUSSION

In late 1990, GE representatives submitted GE Licensing Topical Report (LTR) NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor Power Uprate" (Reference 1). In this LTR, GE proposed to create a generic program to increase the rated thermal power levels of the BWR/4, BWR/5 and BWR/6 product lines by approximately 5 percent. The LTR contained a proposed outline for individual license amendment submittals, as well as discussions of the scope and depth of reviews which would need to be performed and the methodologies which would be used in these reviews. By letter dated September 30, 1991, the NRC issued a staff position concerning the LTR (Reference 2), which approved the proposed program, provided that individual power uprate amendment requests meet certain requirements contained in the document.

The generic BWR power uprate program was created to provide a consistent means for individual licensees to recover additional generating capacity beyond their current licensed limit, up to the reactor power level used in the original design of the nuclear steam supply system (NSSS). The original licensed power level was generally based on the vendor guaranteed power level for the reactor. The difference between the guaranteed power level and the design power level is often referred to as "stretch power." Since the design

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power level is used in determining the specifications for all major NSSS equipment, including the emergency core cooling systems (ECCS), increasing the rated thermal power limits does not violate the design parameters of the NSSS equipment, nor does it significantly impact the reliability of this equipment.

The licensee's amendment request to uprate the current licensed power level of 3293 MWt to a new limit of 3430 MWt represents an approximate 4.2 percent increase in thermal power with a corresponding 5 percent increase in rated steam and feedwater flows. The planned approach to achieving the higher power level consists of (1) an increase in the core thermal power level to increase steam production in the reactor; (2) an increase in feedwater flow corresponding to the increase in steam flow; (3) no increase in maximum allowable core flow; and (4) operation of the reactor along extensions of current rod position/flow rate control lines. This approach is consistent with the generic BWR power uprate guidelines presented in Reference 1 and approved by the staff. The increased core power will be achieved by utilizing a slightly flatter radial power distribution while maintaining the most limiting fuel bundles within their operating constraints. The operating pressure of the reactor will be increased approximately 25 psi to assure satisfactory turbine pressure control and pressure drop characteristics with the increased steam flow.

3.0 EVALUATION

The staff's review of the Fermi-2 power uprate amendment request utilized applicable Rules, Regulatory Guides, SRP Sections, and NRC staff positions regarding the topics being evaluated. Additionally, the Fermi-2 submittal was evaluated for compliance with the generic BWR power uprate program as defined in Reference 1. Detailed discussions of individual review topics follows.

3.1 Reactor Core and Fuel Performance

The effect of power uprate was evaluated for potential impact on various areas related to reactor thermo-hydraulic and neutronic performance. These included changes to the power/flow operating map, core stability, reactivity control, fuel design, control rod drives, and scram performance. Additionally, the staff considered the impact of power uprate on reactor transients, anticipated transients without scram (ATWS), emergency core cooling system (ECCS) performance, and peak cladding temperature for design basis accident break spectra.

3.1.1 Fuel Design and Operation

The licensee has stated that no new fuel designs would be needed to achieve power uprate. This statement is consistent with information provided by GE in LTR NEDC-31984P (Reference 3). Fuel operating limits, such as the maximum average planar linear heat generation rate (MAPLHGR) and operating limit minimum critical power ratio (OLMCPR) for future fuel reloads will continue to be met after power uprate. The methods used for calculation of MAPLHGR and OLMCPR limits will not be changed as a result of power uprate, although the actual thermal limits may vary between cycles. Cycle-specific thermal limits will be included in the plant Core Operating Limits Report (COLR).

DECo has installed four ASEA Brown Boveri Atom (ABBA) type SVEA-96 fuel assemblies into the core for evaluation purposes. Although these fuel assemblies were not manufactured by GE, the design of the SVEA-96 fuel assemblies is sufficiently similar to the GE type GE9B fuel assembly that the applicable GE fuel performance correlations are applicable. The licensee has further committed to place these ABBA fuel assemblies in locations such that they will not be the most limiting assemblies on either a nodal or bundle power basis. Thus, the staff concludes that the use of the ABBA SVEA-96 fuel assemblies, as stated in the licensee's submittal, is acceptable for power uprate.

3.1.2 Fuel Enrichment and Burnup

In response to a staff question concerning uprated power operation, the licensee, in a February 24, 1992 letter, noted their plans to use fuels enriched to a maximum of 5.0 percent by weight of Uranium-235 (^{235}U), and fuel burnup levels not exceeding a maximum rod average burnup of 60,000 MegaWatt-days per metric ton of uranium (MWD/MTU). In their letter, the licensee stated that these values of fuel enrichment and burnup are bounded by an NRC Environmental Assessment (EA) published in the Federal Register (53 FR 6040) and that the conclusions made in the EA are also applicable to Fermi-2. The licensee later clarified that maximum fuel enrichment would be 4.8 percent ^{235}U and maximum rod average burnup would be 49,100 MWD/MTU.

The staff agrees with the licensee's statement that the conclusions of the EA published in the Federal Register (53 FR 6040) are applicable to Fermi-2, and that the use of extended burnup fuels within the limits specified above will have no significant adverse radiological or non-radiological impacts, and will not significantly affect the quality of the human environment.

The staff has reviewed the licensee's submittals as well as a report prepared for the NRC by Pacific Northwest Laboratory (PNL) entitled "Assessment of the Use of Extended Burnup Fuel in Light Water Cooled Power Reactors," NUREG/CR-5009, dated February 1985. In this report, PNL examined the changes that could result in the NRC design-basis accident (DBA) assumptions contained in various Standard Review Plan (SRP) Sections and Regulatory Guides (RGs) as a result of extended fuel burnup (up to 60,000 MWD/MTU). The staff agrees with the conclusions reached by PNL in the report; namely, that the only DBA which could be affected by the extended fuel burnup would be the potential thyroid doses that could result from a fuel handling accident. The PNL report estimated that the calculated iodine gap-release fraction is 20 percent greater for some high power fuel designs than the assumed value of 0.10 stated in RG 1.25. Thus, the calculated thyroid doses resulting from a fuel handling accident with extended burnup fuel could be 20 percent higher than those estimated using RG 1.25.

The staff has reevaluated the fuel handling accident for Fermi-2 using the uprated power level. The calculated 2-hour thyroid dose at the exclusion area boundary would remain less than 1 rem. Similarly, the low population zone (LPZ) thyroid and whole-body doses would be expected to remain less than 0.1 rem for the fuel handling accident. The staff concludes that the potential increased doses resulting from DBA with continued extended burnup levels of up to 60,000 MWD/MTU meet the acceptance criteria provided in SRP Section 15.7.4, and remain well within the dose guidelines described in 10 CFR Part 100. Consequently, the staff finds that the changes proposed by the licensee with respect to the use of fuel enrichments up to 5 percent ^{235}U and for fuel burnup not exceeding 60,000 MWD/MTU to be acceptable.

3.1.3 Power/Flow Operating Map

Power uprate raises the upper portion of the core operating map (reactor 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 the Maximum Extended Operating Domain (MEOD) which was previously achieved at a minimum value of approximately 75 percent of maximum core flow will now be achieved at approximately 81 percent of maximum core flow along the same rod lines. The absolute power MWt at that point on the operating map will be higher since the rated thermal power limit will be redefined.

3.1.4 Stability

The BWR Owners' Group (BWROG) and the NRC are currently addressing methods to minimize the occurrence and potential effects of core power oscillations which have occasionally been observed for certain BWR operating conditions. Until this issue is resolved, the licensee has adopted the generic interim operating constraints proposed by GE. Existing plant procedures have been incorporated in accordance with NRC Bulletin 88-07 and Supplement 1 to that Bulletin which restrict plant operation in the high power/low flow region of the power/flow operating map. Since plant operation after power uprate will simply extend the power/flow map to a higher power level (with corresponding higher flow), the current restricted operation regions of the power/flow map will remain unchanged, and operator actions upon entry into these regions will likewise remain the same. This is consistent with information presented in the generic evaluations provided by GE in Reference 3.

3.1.5 Control Rod Drives and Scram Performance

The control rod drive (CRD) system was evaluated using the uprated steam flow and system pressure. The increased reactor pressure has little effect on scram insertion speed. The licensee has evaluated the CRD system for control rod insertion and withdrawal functions, as well as CRD cooling, and concluded that the CRD system will continue to perform all of its functions at uprated conditions. The licensee will continue to monitor, through various plant TS surveillance requirements, the scram time performance in order to ensure that the original licensing bases for the CRD system are maintained. This approach is consistent with that proposed by GE in Reference 3.

The Fermi-2 power uprate conditions with the increase of reactor dome pressure, temperature and steam flow rate are within the range of values specified in GE generic guidelines for the BWR/4 power uprate. The CRD system was evaluated for a normal maximum reactor dome pressure of 1060 psig, which is higher than the nominal power uprate operating pressure of 1030 psig for Fermi-2. Based on the review of the Fermi-2 power uprate amendment and the GE generic guidelines, the staff concludes that the CRD mechanism will continue to meet its design basis and the CRD will continue to perform its safety function at uprated power.

3.2 Reactor Coolant System and Connected Systems

The staff's review of the mechanical engineering portions of the Fermi-2 power uprate amendment request centered on the effects of power uprate on the structural and pressure boundary integrity of the piping systems and components, their supports, and reactor vessel and internal components.

3.2.1 Nuclear Steam Pressure Relief

The purpose of the nuclear steam pressure relief system is to prevent overpressurization of the NSSS during abnormal operational transients. In BWRs, the main steam line safety/relief valves (SRVs) provide this protection. In Reference 3, GE evaluated the impact of uprated conditions; namely, increased temperatures, pressures, and flow rates on the SRVs. GE concluded that the function and structural integrity of the SRVs would not be compromised by power uprate. The only change to the SRVs which would result from a power uprate would be an increase in the setpoints of the SRVs to accommodate an approximate 25 psi increase in reactor vessel upper head pressure. These setpoints would be increased to maintain an adequate simmer margin during reactor operation.

3.2.2 Reactor Overpressure Protection

The design pressure of the reactor vessel and reactor coolant pressure boundary will remain at 1250 psig after power uprate. The American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME) Code's allowable pressure limit for pressurization events is 1375 psig. The licensee has analyzed the limiting pressurization event, which is a main steam isolation valve (MSIV) closure with failure of the reactor to automatically scram on MSIV position. Four SRVs were assumed to be out of service and an initial operating pressure of 1045 psig was used in the analysis. The analysis also assumed operation at 102 percent of 3430 MWt, 105 percent of rated core flow, and an automatic scram on high neutron flux during the event. At the uprated power, a peak pressure of 1339 psig results, which is higher than the current peak pressure but below the ASME Code's allowable limit. Therefore, the staff concludes that reactor overpressure protection will remain adequate after power uprate.

3.2.3 Reactor Vessel and Internals

The licensee evaluated the reactor vessel and internal components, considering load combinations that include reactor internal pressure difference (RIPD), loss-of-coolant accident (LOCA), safety relief valve (SRV), seismic, annulus pressurization (AP), jet reaction (JR), and fuel lift loads.

The licensing basis LOCA loads such as suppression pool swell, condensation oscillation (CO), and chugging remain unchanged because Fermi-2 dynamic loads were defined based upon the Mark I long-term test conditions, which bound the power uprate conditions with respect to the drywell pressurization rate, vent mass and energy flow rates, and suppression pool water temperature.

With respect to SRV loads, the highest SRV analytical setpoint for Fermi-2 will be 1190 psig after uprate, which is 1 percent (11 psig) higher than the setpoint defined for the original SRV dynamic loads (1179 psig). Since SRV loads are proportional to the SRV pressure setpoint, the 1 percent increase in SRV loads is considered to be negligible with respect to structural response of the reactor vessel and internal components.

The loads that contribute to potential fuel lift are the scram uplift force and reactor building upward motion due to seismic, AP, and JR loads. The seismic loads are unaffected by power uprate. The AP and JR loads increase slightly (about 1 percent) due to a reactor dome pressure increase from 1016 psig to 1030 psig as a result of power uprate. Therefore, the changes to current fuel bundle lift loads are considered to be minimal. The RIPD loads are also increased by approximately 5 percent due to the uprated power conditions. However, this increase in RIPD loading is not significant.

The stresses and fatigue usage factors for reactor vessel components were evaluated by the licensee in accordance with the requirements of the 1968 Edition of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1968 Edition with Summer 1969 Addenda (Reference 7), to assure compliance with the original Code of record for Fermi-2. The load combinations for normal, upset, and faulted conditions were considered in the evaluation. A limiting fatigue usage factor of 0.985 was calculated for the low pressure core spray nozzle safe end for 40 years of operation based upon the uprated power level. There were no new assumptions used in the analysis for the power uprate conditions from those utilized by the licensee in previous evaluations. Based on the staff's review, the maximum stresses and fatigue usage factor, as provided by the licensee, are within the Code's allowable limits and are, therefore, acceptable.

3.2.4 Reactor Recirculation System

The increase in reactor power will be accomplished by operation along extensions of current rod lines on the power/flow map with no increase in the maximum rated core flow. A small increase in flow resistance is expected to occur when operating at maximum core flow, due to an increase in the core average void fraction and a corresponding increase in two-phase flow resistance. 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 increased maximum power conditions. The reactor recirculation pumps will be monitored to assure that no undue vibration will occur at uprated power conditions.

3.2.5 Reactor Coolant Piping

The piping systems which will experience increased piping loads due to uprated power conditions are the main steam lines, associated extraction steam and drain lines, recirculation, low pressure core spray (LPCS), condensate, feedwater, standby liquid control (SLCS), reactor water cleanup (RWCU), and control rod drive (CRD) systems.

The staff's review of the licensee's submittals indicated that the main steam and recirculation piping systems were evaluated for the uprated power conditions, including higher flow rate, temperature and pressure for thermal expansion, dynamic loads, and vibration effects. The evaluations performed consisted of determining the percent increase in ASME Code (Reference 8), Subsection NB-3600, equations 9, 10, 12, 13, and 14 due to power uprate conditions. These percent increases were applied to the calculated stresses in each piping system at the highest stress locations. These revised stresses were then compared with the Code allowable limits for normal, upset, and faulted conditions for acceptability. The licensee stated that the design adequacy evaluations show that the Code requirements are satisfied for all evaluated piping systems and that power uprate will not have an adverse effect on the primary piping system design.

The licensee also stated that the Class 1 portions of the LPCS, feedwater, SLCS, RWCU (outside containment), main steam (outside containment), main steam line drain, reactor core isolation cooling (outside containment), high pressure coolant injection (outside containment), residual heat removal (outside containment), and reactor pressure vessel (RPV) head vent line were evaluated and shown to be adequate at the uprated conditions. Small bore reactor coolant pressure boundary (RCPB) piping, such as instrument lines, was also evaluated. For these lines, the licensee stated that the original Code of record, Code allowable limits, and analytical techniques were used, and that no new assumptions were introduced which were not in the original analyses.

In response to the staff's positions regarding the generic BWR power uprate program, General Electric stated that high energy line breaks and subsequent dynamic effects have been considered in the GE generic evaluation. The licensee also stated that postulation of pipe break locations is performed in accordance with Branch Technical Position MEB 3-1 of SRP Section 3.6.2. No new postulated pipe break locations were identified.

Pumps and valves (including SRVs) were originally designed and manufactured to design pressures of 1250 psig to 1650 psig. The ASME Code allows a peak pressure of 110 percent of the design value; that is, the allowable peak pressure for pumps and valves is 1375 psig to 1815 psig, in comparison to the maximum RCPB transient pressure of 1339 psig for the uprated power conditions. Accordingly, the staff concludes that the pressure integrity of pumps and valves will be assured for operation at uprated power.

The licensee stated that piping interface loads to the RPV nozzles, anchors, struts, penetrations, flanges, pumps, and valves were evaluated in a manner similar to that for piping. The effects of uprated power conditions on thermal and vibration displacement limits were also evaluated. The anchorage, base plates, and lugs were evaluated and qualified by applying conservative loads from GE generic enveloping design loads. The licensee concluded that interface loads on the system components do not exceed (original) component acceptance criteria. The pipe supports were evaluated based on the comparison of the difference between the original design stresses and the Code limits, and the stress increases due to power uprate. The licensee indicated that those pipe supports were determined to be acceptable.

3.2.6 Main Steam Isolation Valves (MSIVs)

The performance of the MSIVs with regard to reactor coolant pressure boundary requirements, such as closure time and leakage, could potentially be impacted by the increased reactor operating pressure. However, the pressure increase is relatively small (less than 5 percent) and MSIV performance will be monitored by surveillance requirements in the plant TSs to ensure that the original licensing basis for the MSIVs is preserved.

3.2.7 Reactor Core Isolation Cooling (RCIC) System

The RCIC system provides core cooling when the reactor pressure vessel is isolated from the main condenser, and RPV pressure is greater than the maximum allowable for initiation of a low pressure cooling system. The licensee has assessed the RCIC system in a manner consistent with the bases and conclusions of Section 4.2 of Reference 3. The licensee has committed to implement the recommendation of GE SIL 377; specifically, to add a small bypass around the steam admission valve of the RCIC turbine in order to reduce the probability of a turbine overspeed trip during system start-up. The staff has required that individual licensees provide assurance that the 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 other modifications made to the system. In response to a staff request, the licensee has committed to conduct performance tests to ensure that the RCIC system will continue to function as designed at the uprated conditions (See Section 3.8.3).

Successful completion of these tests should provide reasonable assurance that the performance of the RCIC system will not be compromised because of the higher loads placed on the system or because of any modifications made to the system to compensate for these increased loads.

3.2.8 Residual Heat Removal (RHR) System

The RHR system is designed 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 coolant 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 less than or equal to 198 °F

3.2.9 Reactor Water Cleanup (RWCU) System

The RWCU system operating pressure and temperature will increase slightly as a result of power uprate. The licensee has evaluated the impact of these increases and has concluded that uprate will not adversely affect RWCU system integrity. The cleanup effectiveness of the RWCU system may be slightly diminished as a result of increased feedwater flow to the reactor; however, current TS limits for reactor water chemistry will not be changed as a result of power uprate. Therefore, power uprate will not significantly impact the operatic or coolant boundary integrity of the RWCU system.

3.3 Engineered Safety Features

The staff's review of the impact of the Fermi-2 power uprate amendment request included the effect on containment system performance, the standby gas treatment system (due to increased iodine loading), post-LOCA combustible gas control, the main steam isolation valve leakage control system, the control room atmosphere control system and the emergency cooling water system. This review was performed to ensure that the ability of these systems to perform their safety function to respond to or mitigate the effects of design basis accidents was not impaired by the approval of power uprate. Additionally, the effects of power uprate on high energy line breaks, fire protection, and station blackout were considered.

3.3.1 Containment System Performance

Primary containment temperature and pressure response following a postulated LOCA is of great importance when determining the potential for offsite release of radioactive material, in determining ECCS pump net positive suction head (NPSH) requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment. In Reference 1, GE proposed to update the calculational methods used for determining peak containment temperatures and pressures following a postulated LOCA. In particular, GE proposed to utilize the SHEX computer code when calculating the peak suppression pool temperature during the long-term portion of containment post-LOCA response, in place of the previously used M3CPT/HXSIZ combination. The staff, in Reference 4, stated that although the NRC had not formally approved the SHEX code on a generic basis, use of SHEX in place of M3CPT/HXSIZ would be permitted on a plant-specific basis, provided adequate information was provided to justify its use.

3.3.1.1 Use of SHEX for Long-Term Suppression Pool Temperature Response

When evaluating containment post-LOCA response, the M3CPT code is used to calculate short-term containment temperature and pressure response following a postulated LOCA, while either SHEX or a combination of M3CPT and HXSIZ would be used to determine the long-term suppression pool temperature. The M3CPT code uses a mechanistic method to model the highly transient conditions in the containment immediately following a LOCA, and is capable of modelling containment long-term response, up to the initiation of containment cooling. M3CPT has been verified against experimental data and has been previously approved by the NRC staff.

During the 1970's, GE used the M3CPT/HXSIZ combination to model the long-term response of the containment to large-break LOCAs. The M3CPT code was used to model both the short-term and long-term response to the LOCA from the time of the breakup to the time of initiation of containment cooling. After initiation of containment cooling, the HXSIZ code was used to model the containment heat exchangers, using input values obtained from M3CPT. By modelling the containment heat exchangers, the suppression pool temperature could be calculated as a function of time.

The SHEX code utilizes more refined models than those used by M3CPT/HXSIZ to determine suppression pool temperature, and is capable of modelling containment responses to more accident scenarios than the HXSIZ code. Many of the models used in SHEX are the same as, or very similar to, those used in M3CPT. SHEX is also capable of modelling all containment auxiliary systems, permitting a more accurate analysis of actual containment conditions following a postulated LOCA.

The licensee believes that M3CPT/HXSIZ was used to perform the original plant licensing calculations, but is unable to provide documentation to support this claim. However, GE has stated that the M3CPT/HXSIZ combination was commonly used in containment evaluation during the time of licensing of Fermi-2. Additionally, several statements made in the plant Updated Final Safety Analysis Report (UFSAR) indicate the use of assumptions which are commonly used with HXSIZ. Thus, the staff agrees with the licensee's claim that the

HXSIZ code was most likely used in the long-term containment analysis documented in the plant UFSAR.

General Electric, on behalf of the licensee, evaluated the containment response to LOCA conditions, using the M3CPT computer calculation for short-term drywell pressure response and the SHEX computer code for long-term suppression pool temperature response (Reference 17). The results of this evaluation were compared to similar results obtained from the M3CPT/HXSIZ combination using identical input parameters in order to verify that the results obtained by SHEX were at least as conservative as those obtained by M3CPT/HXSIZ. Using assumptions consistent with power uprate, SHEX predicted a peak suppression pool temperature of 196.5 °F, while M3CPT/HXSIZ predicted 196.1 °F. Additionally, time/temperature plots obtained from both codes showed extreme similarity in predicted suppression pool temperatures as a function of time throughout the event. Since the codes predict essentially identical peak suppression pool temperatures (the SHEX result is slightly more conservative), use of SHEX for the analysis of long-term suppression pool response to power uprate is acceptable for Fermi-2.

3.3.1.2 <u>Containment System Performance Evaluation</u>

The licensee evaluated the effects of power uprate on the containment response to postulated LOCAs using the M3CPT/SHEX combination as described above. In addition to using a new code to model long-term response, the licensee revised a number of input parameters to the containment analysis in order to more accurately reflect actual plant operating conditions. In the short-term analysis, the licensee assumed a higher initial reactor power level, higher reactor dome pressure, higher initial drywell temperature, a larger initial suppression pool water volume, and a higher initial suppression pool temperature. The analysis, using the revised input parameters, predicted a peak drywell pressure of 49.9 psig, as compared to 48.3 psig calculated by the licensee at the current power level as part of the Mark I Long Term Program. The uprated peak pressure is bounded by a peak pressure of 56.5 psig which was calculated by the licensee and is documented in the UFSAR. Additionally, the peak drywell pressure remains below the containment pressure acceptance criteria of 62 psig.

In the long-term analysis, the licensee changed a number of assumptions which would tend to make the results more conservative. These included a lower suppression pool volume, higher initial suppression pool temperature, feedwater addition to the suppression pool, and a delayed heat exchanger initiation time. The licensee also made two assumptions which would tend to make the results less conservative. These assumptions were a lower initial service water temperature and a more realistic decay heat model. As discussed above, SHEX predicted a peak suppression pool temperature of 196.5 °F for uprated conditions, which is more conservative than the M3CPT/HXSIZ result of 196.1 °F, and the UFSAR value of 191 °F based on the current power level. The uprated peak suppression pool temperature of 196.5 °F remains below the acceptance criteria of 198 °F and is, therefore, acceptable.

The staff has concluded that the containment temperature and pressure response following a postulated LOCA will remain acceptable after uprate. The staff also concludes that the containment will continue to meet the requirements for sufficient margin from temperature and pressure limits as described in 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment design basis." The staff, therefore, considers the containment response following power uprate to be acceptable.

3.3.2 Emergency Core Cooling Systems (ECCS)

With the suppression pool temperature remaining below 198 °F, the ECCS NPSH requirements will still be satisfied after uprate for the limiting conditions of 0 psig containment pressure, and the maximum expected temperature of pumped fluids will not change from the UFSAR licensing basis.

3.3.2.1 High Pressure Coolant Injection (HPCI) System

The HPCI system design basis is to provide reactor vessel inventory make-up during small and intermediate break for loss-of-coolant accidents (LOCA) and reactor vessel isolation events. The HPCI system 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 effect of power uprate on HPCI system operability, including potential system modifications, was addressed by GE in Reference 3.

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 3 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 overspeed 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, committed to conducting performance tests to ensure HPCI can operate as designed at uprated conditions (See Section 3.8.3). 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.

3.3.2.2 RHR System (Low Pressure Coolant Injection, LPCI)

The licensee has adopted the generic evaluation provided in the generic topical report (Reference 3) for the LPCI mode of the RHR system. This analysis is applicable to Fermi-2 and there are no changes associated with power uprate for the LPCI mode of operation.

3.3.2.3 Low Pressure Core Spray (LPCS) System

The licensee has adopted the bounding generic evaluation provided in the GE topical report (Reference 3) for the LPCS system. That analysis is applicable to Fermi-2. 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.

3.3.3 Emergency Core Cooling System 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 Part 50, Appendix K. The results of the ECCS/LOCA analysis using NRC approved methods are presented below.

A plant-specific analysis was performed for Fermi-2 using the Cycle 3 fuel types. The licensee used the staff-approved SAFER/GESTR methodology to assess the ECCS capability for meeting the 10 CFR 50.46 criteria.

The results of the break spectra calculations show that the DBA recirculation line suction break with Division II battery failure is the limiting case. The nominal peak cladding temperature (PCT) is calculated to be 1002 °F with a corresponding Appendix K PCT of 1597 °F. The licensing basis PCT is calculated to be 1602 °F. The upper bound peak cladding temperature (UBPCT) is calculated to be 1[illegible]1 °F. The licensing basis PCT is less than 2200 °F and the UBPCT is 251 °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 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 more restrictive initial MCPR and MAPLHGR/PLHGR limits and require MCPR and MAPLHGR multiplier factors to be imposed. These required power-dependent 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. The staff finds this conclusion to be acceptable.

3.3.4 Standby Gas Treatment System (SGTS)

The standby gas treatment system (SGTS) consists of two 100 percent capacity filter trains each containing a full complement of needed components including a 6-inch charcoal adsorber and HEPA filters (one each) upstream and downstream of the charcoal adsorber (NUREG-0798, Fermi-2 SER, Subsection 6.5.2.1, July 1981). The system is designed to ensure controlled and filtered release of radioiodine and radioactive material in particulate form from the containment to the environment during accident or abnormal conditions to maintain offsite thyroid doses within the 10 CFR Part 100 limits (300 rem). The system accomplishes the above design objective, since each train is sized to change one secondary containment (SC) air volume per day and maintain the SC at a slight negative pressure of 1/4 inch water gauge with respect to the outside atmosphere, to prevent unfiltered release of radioactive material from the SC to the environment. The staff agrees with the licensee that the proposed slight uprate in power (4.2 percent) by itself will not have any adverse impact on the capability of the SGTS to meet the above design objective since it does not change the ventilation design aspect of the SGTS.

The staff recognizes that iodine loading in the filters will increase marginally (4.2 percent) due to the proposed power uprate. The staff had concluded earlier (Fermi-2 SER, Subsection 6.5.2.1, July 1981) that the SGTS design meets the intent of RG 1.52 guidelines with respect to the design, testing, and maintenance criteria of engineered safety feature (ESF) grade filters and is, therefore, acceptable. The staff notes that one of these criteria deals with the filter loading capability. The staff further notes that the licensee has determined (licensee's submittals dated September 24, 1991, Enclosure 3, Section 4.4) that although the iodine loading in the filters will increase slightly, it will remain well below the original design capacity of the filter. Further, in a telephone conversation with the staff on July 29, 1992, the licensee confirmed that its earlier calculation on SGTS filter loading of iodine showed an ample margin between the calculated value and RG 1.52 acceptance criterion (no more than 2.5 milligrams of iodine [both radioactive and stable isotopes] per gram of activated carbon) to accommodate the slight increase in iodine loading that can be expected from the 4.2 percent increase in the proposed power uprate. Based on the above, the staff concludes that its earlier conclusion regarding the system's ability to meet the guidelines of RG 1.52 continues to be valid for the proposed uprated power situation. The staff also notes that even with a slight increase (4.2 percent) in the previously calculated limiting offsite thyroid dose (150 rem as given in Table 15.1, Fermi-2 SER, 1981) due to the uprated power, the thyroid dose will still remain well below the 10 CFR Part 100 limit of 300 rem.

Based on the above findings, the staff concludes that the uprated power level operation will not have any impact on the ability of the SGTS to meet its design objectives.

3.3.5 Other ESF Systems

3.3.5.1 Main Steam Isolation Valve Leakage Control System

The licensee's containment analysis calculated that the peak post-LOCA pressures at uprated power conditions do not increase beyond the original design basis. Based on the staff's review of those calculations (see Section 3.3.1), the staff agrees with the licensee's assertion that the operation of the MSIV Leakage Control System will not be impacted by power uprate.

3.3.5.2 Post-LOCA Combustible Gas Control

In their submittal, the licensee confirmed the ability of the combustible gas control system (CGCS) to maintain oxygen and hydrogen concentrations within acceptable levels following a LOCA. This conclusion is consistent with that reached by GE in Reference 3. The licensee stated that although the amount of oxygen liberated by radiolytic decomposition of water is expected to increase slightly due to power uprate, the expected concentrations are well within the capacity of the CGCS. The licensee also stated that hydrogen recombiners may need to be started sooner following a postulated LOCA after uprate; however, current procedures which direct control room operators to initiate the recombiners are based on combustible gas concentrations, not on a fixed time following a LOCA.

Additionally, the revised hydrogen generation calculations provided by the licensee indicate that less hydrogen will be liberated due to corewide metal-water reactions than previously predicted. This slight decrease is primarily due to significantly lower predicted fuel cladding temperatures during a postulated LOCA. The decrease in expected PCT is a result of the use of more realistic calculational methods in the ECCS/LOCA analysis (See Section 3.3.3). Based upon our review of the licensee's submittals, the staff concludes that the existing post-LOCA combustible gas control systems will continue to perform their design function after power uprate.

3.3.5.3 Main Control Room Atmosphere Control System (CRACS)

The CRACS is one of the control room habitability systems. The CRACS includes an emergency filtration system which, in turn, contains an emergency make-up air filter train and an emergency recirculation filter train. The emergency make-up air filter train consists of a pre-filter, electric heaters, a 2-inch charcoal adsorber and HEPA filters, one upstream and another downstream of the adsorber. The make-up air filter train filters the radioiodine and radioactive material in particulate form present in the outside make-up air intake during an emergency situation such as a design basis accident (DBA). The emergency recirculation filter train consists of a pre-filter, a 4-inch charcoal adsorber, HEPA filters, one upstream and another downstream of the adsorber and emergency recirculation air fans. The emergency recirculation filter train filters a mixture of the control room recirculated air and already once filtered outside make-up air. The filters are designed in accordance with RG 1.52 (NUREG-0798, Fermi-2 SER, Section 9.4.1) guidelines.

The emergency filtration system is designed to maintain the control room envelope at a slight positive pressure (1/8" water gauge) relative to the outside atmosphere and thus minimize unfiltered inleakage of contaminated outside air into the control room during an accident. The system accomplishes the above design objective by bringing in controlled and filtered outside air and filtering the recirculated air to keep the control room operator doses within the GDC 19 limits during an accident. The staff concludes that the proposed slight uprate in power (4.2 percent) by itself will not cause any increase in unfiltered inleakage of contaminated outside air into the control room during an accident since it does not change the ventilation design aspect of the control room emergency filtration system.

The staff recognizes that iodine loading in the make-up air filters and recirculation air filters will increase marginally (4.2 percent) due to the proposed power uprate. As noted above, the staff had concluded earlier that the control room emergency filtration system filters meet the guidelines of RG 1.52, one of which deals with the filter loading capability. By telephone conversation with the staff on July 29, 1992, the licensee confirmed that its earlier calculation on the subject filter loading of iodine showed sufficient margin between the calculated value and RG 1.52 acceptance criterion to accommodate the slight increase in iodine loading that can be expected from power uprate. Based on the above conversation, the staff concludes that its earlier conclusion regarding the filters meeting the guidelines of RG 1.52 continues to be valid for the proposed uprated power situation.

In its submittal dated September 24, 1991, the licensee calculated control room operator doses of 0.28 rem whole body and 7.1 rem thyroid for the uprated power case. The licensee utilized χ/Q values from their UFSAR which are different from those used by the staff during the original licensing of the plant. However, by earlier Safety Evaluation Report (SER) and its supplements for Fermi-2 (NUREG-0798, SER, July 1981; SSER 3, January 1983; SSER 5, March 1985; and SSER 6, July 1985), the staff had approved the control room habitability systems for Fermi-2, stating that they meet GDC 19 with respect to control room operator doses and applicable RG 1.95 guidelines with respect to toxic gas (chlorine) protection provisions. The SSER 5 calculated limiting (design basis LOCA) control room operator doses of 16.1 rem thyroid and 1.5 rem whole body, both of which are within the GDC 19 limit of 5 rem whole body or its equivalent to any part of the body (the staff considers 30 rem as the equivalent thyroid dose limit on the above basis). In assessing the impact of power uprate, the staff used the same χ/Q values as during the original licensing of the plant, which are more conservative than those used by the licensee. The effect of power uprate on the control room operator doses will be small (a maximum increase of 4.2 percent) and will still be well within the GDC 19 limits. Based on the above findings, the staff concludes that the slight power uprate of 4.2 percent by itself will not increase the control room doses in excess of the GDC 19 limits.

Based on the above findings, the staff concludes that the uprated power level by itself will not have any impact on CRACS meeting its design objectives.

3.4 Instrumentation and Control

The staff's evaluation of setpoint changes associated with power uprate was limited to those setpoint changes for instrumentation identified in the licensee's submittals to the staff. Although the staff has not completed its review of GE Topical Report NEDC-31336P, "General Electric Instrument Setpoint Methodology," the staff is sufficiently familiar with the methods to permit their application to plant-specific data within the limits stated in the Topical Report.

A review of the licensee's submittals indicates that GE performed plant-specific calculations for the licensee using methods recommended by the Instrument Society of America (ISA) as outlined in GE Topical Report NEDC-31336P (Reference 6).

The following setpoint changes have been proposed by the licensee:

(a) Flow Biased Simulated Thermal Power for Two-Loop Operation

Change trip from (0.66W + 64%) to (0.63W + 61.4%)

Change Allowable Value from (0.66W + 67%) to (0.63W + 64.3%)

(b) Flow Biased Simulated Thermal Power for One-Loop Operation

Change trip from (0.66W + 58.7%) to (0.63W + 56.3%)

Change Allowable Value from (0.66W + 61.7%) to (0.63W + 59.2%)

(c) Reactor Vessel Steam Dome Pressure High

Change trip from 1068 psig to 1093 psig

Change Allowable Value from 1088 psig to 1113 psig

(d) Main Steam High Flow

Change trip from 109 psid to 115.4 psid

Change Allowable Value from 112 psid to 118.4 psid

(e) Rod Block for Two-Loop Operation

Change trip from (0.66W + 58%) to (0.63W + 55.6%)

Change Allowable Value from (0.66W + 61%) to (0.63W + 58.5%)

(f) Rod Block for One-Loop Operation

Change trip from (0.66W + 52.7%) to (0.63W + 50.5%)

Change Allowable Value from (0.66W + 55.7%) to (0.63W + 53 [illegible])

(g) Turbine Stop Valve and Turbine Control Valve Fast Closure Scram Bypass

The turbine first stage pressure setpoint was changed to reflect the expected pressure at the new 30 percent power point.

(h) APRM Rod Block and APRM Simulated Thermal Power High Power Clamps and APRM Neutron Flux Scram

These setpoints were not physically changed. However, the change in the definition of rated thermal power (from 3293 MWt to 3430 MWt) will result in an increase of approximately 137 MWt to each of these points.

To verify the results o. icensee-sponsored calculations and to better understand the quantitative effects of the assumed instrument errors, the staff audited the calculations for the reactor vessel steam dome high pressure trip, the main steam high flow trip, and the APRM trips (both fixed and flow biased). The review demonstrated that the instrumentation errors assumed in the analyses were conservative with respect to the manufacturers' ratings and that the methods of analysis generally conform to those described in Reference 6. Exceptions to the methods described in Reference 6 are based on plant-specific data and instrumentation calibration procedures. The staff also acknowledges that these changes represent more current knowledge than was available when the Topical Report was issued in 1986.

The proposed setpoint changes are designed to maintain the existing margins between the proposed operating conditions and the new trip points. The same margins to the new safety limits are also maintained. These new setpoints do not significantly increase the likelihood of a false trip or a failure to trip upon demand. Therefore, the staff finds the setpoint changes, as described in the licensee's submittals, to be acceptable for power uprate.

3.5 Auxiliary Systems

3.5.1 Spent Fuel Pool Cooling

The spent fuel cooling system is designed to remove the decay heat generated by the stored spent fuel assemblies. The system consists of two 50 percent capacity spent fuel pool cooling pumps and heat exchangers. Backup or supplemental cooling is provided to the spent fuel pool by the residual heat removal (RHR) system.

As a result of operation at the uprated power level, each reload will affect the decay heat generation in the spent fuel discharged from the reactor and the spent fuel pool heat load will slightly increase. The licensee performed an analysis which indicates that for the normal uprated power fuel cycle, the maximum pool temperature will be 127 °F and, for the emergency full core offload with spent fuel cooling system at maximum cooling capacity and supplemental RHR cooling, the pool temperature will be 125 °F. Consequently, the licensee determined that the changes are small and are within the design limits of the affected systems and components.

Based on its review, the staff agrees with the licensee that the effects of uprated power level operation on the spent fuel pool cooling is insignificant. Therefore, the staff concludes that there is no need for the licensee to modify its spent fuel pool cooling system design.

3.5.2 Water Systems

The licensee evaluated the impact of power uprate on the various plant water systems, including the safety-related and nonsafety-related service water systems, closed loop cooling systems, circulating water system, and the plant ultimate heat sink. The licensee's evaluations considered increased heat loads, temperatures, pressures, and flow rates. The results of the staff's review of these evaluations are discussed below.

3.5.2.1 Service Water Systems

3.5.2.1.1 Safety-Related Loads

These systems are the emergency equipment service water (EESW) system, the diesel generator service water (DGSW) system and the residual heat removal service water (RHRSW) system. All heat removed by these systems is rejected to the atmosphere via the ultimate heat sink (UHS) which includes the RHRSW cooling tower. The staff's evaluation of the effects of uprated power level operation on each of these systems is provided below:

The EESW system is designed to provide a cooling water source for the emergency equipment cooling water (EECW) system during a loss of offsite power, high drywell pressure, or upon failure of the reactor building closed cooling water (RBCCW) system. Based on its review, the staff finds that the original design loads for this system were based on maximum equipment loads which are greater than the anticipated equipment loads resulting from the uprated power level operation. Therefore, the staff concludes that the uprated power level operation has no impact to the EESW system design.

The DGSW system is designed to provide cooling water to the emergency diesel generators (EDGs) during testing and emergency operation. Based on its review, the staff agrees with the licensee that no change in heat load for this system due to the uprated power level operation will be anticipated since no new or increased electrical loads are imposed on the EDGs.

The RHRSW system which takes suction from the ultimate heat sink and returns water to the ultimate heat sink via the cooling tower is designed for the following functions:

(a) to remove decay heat and residual heat from the nuclear system during refueling and nuclear system servicing,

(b) to supplement the spent fuel pool cooling system with additional cooling capacity,

(c) to remove decay heat and residual heat from the suppression pool following a LOCA,

(d) to flood the reactor pressure vessel (if needed) following a LOCA, and

(e) to flood the primary containment (if needed) following a LOCA.

As a result of uprated power level operation, the following functions of the RHRSW system will be affected (due to increased heat loads) to a minor degree:

(a) when operating in the reactor shutdown cooling mode,

(b) when operating in the spent fuel pool cooling (backup system) mode, and

(c) when operating in the suppression pool cooling mode following a LOCA.

However, the RHRSW cooling towers are designed to provide cooling water with a temperature of 89 °F at design ambient conditions and RHRSW return water temperature of 116 °F. The licensee has performed a calculation which indicates that the maximum RHRSW return water temperature will be 115 °F and no increase in the RHRSW supply water temperature results.

Based on our review, the staff concludes that the uprated power level operation has no impact to the RHRSW design.

3.5.2.1.2 <u>Nonsafety-Related Loads</u>

The licensee indicated that the increase in general service water (GSW) system heat loads is projected to be approximately proportional to the uprated power level operation and that this increase of heat loads is insignificant to the design of the system. The GSW is capable of supplying sufficient water to remove the additional heat loads.

Since the GSW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the GSW system design and performance.

3.5.2.2 Main Condenser/Circulating Water System/Cooling Tower Performance

The main condenser, circulating water system, and cooling tower are designed to remove the heat rejected to the condenser and, thereby, maintain adequately low condenser backpressure. The licensee indicated that the performance of the main condenser was evaluated and confirmed that the condenser, circulating water system, and cooling towers are adequate for uprated powe level operation.

Since the main condenser, circulating water system, and associated cooling tower do not perform any safety function, the staff has not reviewed the impact of the uprated power level operation to the designs and performances of these systems.

3.5.2.3 Reactor Building Closed Cooling Water (RBCCW) System

The licensee indicated that the RBCCW system is designed to remove heat from the auxiliary equipment located in the reactor building. The increase in this heat load due to uprate power level operation is insignificant, therefore, there is no impact to the RBCCW system design.

Based on our review, the staff agrees with the licensee's conclusion that the effects of uprated power level operation on the RBCCW system is insignificant and there is no need to modify its RBCCW system design.

3.5.2.4 Turbine Building Closed Cooling Water (TBCCW) System

The TBCCW system is designed to remove heat from both generator-related and nongenerator-related equipment. The licensee has indicated that the increase in heat loads from this equipment due to the uprated power level operation is insignificant and that the TBCCW system design cooling capacity will not be exceeded.

Since the TBCCW system does not perform any safety function, the staff has not reviewed the impact of the uprated power level on the TBCCW system design and performance.

3.5.2.5 Ultimate Heat Sink (UHS)

The licensee indicated that, as a result of operation at the uprated power level, the post-LOCA UHS water temperature will increase, primarily due to an increased reactor decay heat load. This results in a higher evaporation rate and, therefore, a higher minimum water inventory requirement in the RHR reservoir. The licensee further indicated that a review was performed to evaluate the need for a revised TS water inventory requirement. The licensee determined that the exiting UHS system will provide a sufficient quantity of water at a temperature less than 89 °F (design temperature) following a LOCA and that the TS for RHR reservoir water level is adequate due to the conservatism in the original water requirement calculations. Consequently, the licensee concluded that the UHS design is adequate for the uprated power level operation.

Based on our review, the staff concludes that the licensee has shown that the UHS design is adequate for the uprated power operation and no modification to the UHS system is needed.

3.5.3 Standby Liquid Control System (SLCS)

In order to accommodate increased fuel energy requirements for power uprate, the licensee will increase the ^{235}U enrichment of the fuel to a maximum of 5 percent by weight. The increased excess reactivity associated with this increase in fuel enrichment will impact the reactivity requirements of the SLCS. In particular, the licensee will increase the amount of poison (^{10}B) available to shut down the reactor by increasing the required minimum SLCS storage tank level. The boron concentration limits will range from 8.5 to 9.5 percent sodium pentaborate by weight in solution. The licensee utilizes sodium pentaborate which is enriched to 65 atom percent of ^{10}B. The SLCS requirements for future operating cycles will be evaluated by the licensee on a cycle-specific basis.

3.5.4 Power Dependent Heating, Ventilation and Air Conditioning (HVAC)

The licensee indicated that operation of the plant at the uprated power level will result in a maximum increase in process fluid temperatures of approximately 6 °F in the steam cycle systems and a maximum of 1 °F in other auxiliary systems, with the exception of the fuel pool which will increase approximately 2 °F during its maximum loading. The licensee evaluated the impact of the slight increase of process fluid temperatures on the HVAC systems in all affected areas. The result of this evaluation indicates that the assumed heat loads in the original design calculations are adequate for operation at the uprated power level. Consequently, the licensee has concluded that the uprated power level operation has no impact to the plant HVAC systems.

Based on our review, the staff concludes that the licensee has shown that operation of the plant at uprated power level will have no impact to the plant HVAC systems.

3.5.5 Fire Protection

The licensee indicated that the operation of the plant at the uprated power level does not affect the fire suppression or detection systems. There are no physical plant configuration or combustible load changes resulting from the uprate. The safe shutdown systems and equipment used to achieve and maintain cold shutdown conditions do not change, and are adequate for the uprated conditions. The operator actions required to mitigate the consequences of a fire are not affected. Therefore, the fire protection systems and analyses are not affected by the plant power uprate.

Based on our review of the licensee's submittal, the staff finds that the fire suppression and detection systems and their associated analyses are not affected by power uprate.

3.6 Radwaste Systems and Radiation Sources

In reviewing the radiological portions of this amendment request, the staff considered only the effects of a 2 percent uncertainty factor on the radiological evaluations, since the original licensing calculations were previously performed at the design power level of 3430 MWt. The licensee evaluated the radiological impact of the proposed amendment to show that the applicable regulatory acceptance criteria continue to be satisfied for the uprated power conditions. In conducting this evaluation, the licensee considered the effect of the higher power levels on source terms, onsite and offsite doses and control room habitability during both normal and accident conditions.

3.6.1 Liquid Waste Management

The largest source of liquid waste from the Fermi-2 facility arises from backwash of the condensate demineralizers. As a result of power uprate, the licensee expects that the average time between condensate demineralizer backwash/precoat cycles will be reduced slightly. In addition, the licensee noted that the floor drain and waste collector subsystems would not be expected to experience a significant increase in the total volume of liquid waste as a result of power uprate.

The licensee also noted that an increase in activated corrosion products would be expected proportional to the power uprate, but that the total volume of processed waste would not be expected to increase appreciably. The licensee concluded, based on a review of plant operating effluent reports and a consideration of the expected slight increase in effluents as a result of power uprate, that the requirements related to 10 CFR Part 20 and 10 CFR Part 50, Appendix I will continue to be satisfied. Based upon the staff's review of available plant data and experience with previous power uprates, the staff concludes that no significant adverse effect on liquid effluents will occur due to power uprate.

3.6.2 Gaseous Waste Management

The licensee noted that gaseous wastes generated during both normal and abnormal operation are collected, controlled, processed, stored, and disposed utilizing the gaseous waste processing treatment systems. These systems include the offgas system and standby gas treatment system, as well as other building ventilation systems. Various devices and processes, such as radiation monitors, filters, isolation dampers, and fans, are used to control airborne radioactive gases. Finally, the licensee noted that airborne effluent activity released through building vents is not expected to increase significantly after power uprate. Based on review of available plant data and previous experience with other power uprates, the staff concludes that no significant adverse effect on airborne effluents will occur as a result of power uprate.

3.6.3 Radiation Sources in the Core and Coolant

Radioactive materials in the reactor core are produced in direct proportion to the fission rate. Thus, the expected increase in the levels of radioactive materials (for both fission and neutron activation products) produced are expected to increase by a maximum of 4.2 percent. The licensee noted that experience to date with operation of Fermi-2 indicates that concentrations of fission and activation products in the reactor coolant will not increase significantly above those currently experienced. Current experience with operation of Fermi-2 indicates that the unit operates well below the 0.1 Curie/sec design basis and that current offsite radiological release rates are well below the original design basis. Based upon a review of available plant data and experience with previous power uprates, the staff concludes that no significant adverse effect on radiation sources in either the core or reactor coolant will occur due to power uprate.

3.6.4 Radiation Levels

The licensee considered the effects of power uprate on radiation levels in the Fermi-2 facility during normal operation as well as during post-accident conditions. The licensee concluded that radiation levels from both normal operation and accident conditions could be increased slightly. However, any such increase would be small and would be bounded by conservatisms in the original plant design and analysis. Further, the licensee noted that the calculated offsite radiological consequences are well below the regulatory limits set forth in 10 CFR Part 20 and 10 CFR Part 50, Appendix I. Based on a review of plant data and prior experience with other power uprates, the staff finds that no significant adverse effect on radiation levels (either onsite or offsite) will result from the proposed power uprate.

3.7 Reactor Safety Performance Evaluations

3.7.1 Reactor Transients

The limiting UFSAR 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 critical power ratio (CPR). This delta CPR added to the SLMCPR gives the operating limit minimum critical power ratio (OLMCPR).

3.7.2 Design Basis Accidents

The licensee reanalyzed a number of events to determine the whole-body and thyroid doses at the exclusion area boundary and in the low population zone. In evaluating the effects of power uprate on accident consequences, the licensee reanalyzed the loss-of-coolant accident, the main steam line break accident, the instrument line break, the fuel handling accident, and the control rod drop accident. These design basis accidents are the same as those analyzed by the licensee in the initial operating license review and discussed in NUREG-0798, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant Unit No. 2."

The staff has reviewed the information provided by the licensee as well as the information contained in NUREG-0798. Based on a review of this information, the staff concludes that the analyzed consequences of evaluated accidents will increase by only the 2 percent uncertainty factor applied to the analyses by the licensee since these accidents were previously evaluated by the staff in NUREG-0798 at a thermal power level of 3430 MWt. The analyzed consequences of postulated accidents remain within staff acceptance criteria and are, therefore, acceptable.

3.7.3 Anticipated Transients Without Scram (ATWS)

The licensee has stated that the response of Fermi-2 to ATWS events is bounded by the generic analyses, the results of which were documented by GE in Supplement 1 to NEDC-31984P (Reference 5). The ATWS analysis included in Reference 5 was performed in a manner consistent with the analysis performed by GE in 1979 and documented in NEDE-24222, "Assessment of BWR Mitigation of ATWS, Volume II." GE provided additional information concerning these generic analyses in a telephone conversation on August 26, 1992, and in a written submittal (Reference 18). The most significant difference in assumptions between the analyses in Reference 5 and the 1979 version is that the Reference 5 analysis assumes that reactor operators would maintain reactor water level near the TAF throughout the event, in accordance with the guidance provided in Revision 4 of the EPGs. Additionally, GE made use of a modified boron mixing model in the Reference 5 analysis, based on the results of testing. All other assumptions used in the Reference 5 analysis are at least as conservative as those used in the NEDE-24222 analysis.

The analysis in Reference 5 assumed the same representative BWR/4 plant as was assumed in the NEDE-24222 analysis. The two most limiting ATWS events were evaluated: (1) the inadvertent MSIV closure at full power (MSIVC), and (2) pressure regulator failure at full power - maximum demand (PREGO). The most limiting results of these analyses are discussed below. The MSIVC analysis predicts a peak reactor pressure of 1398 psig. For the PREGO event, the analysis predicts a maximum fuel clad temperature of 1672 °F, a peak suppression pool temperature of 166 °F, and a peak containment pressure of 6.9 psig. All of these peak values are well within previously established acceptance criteria.

The staff is currently reviewing the generic ATWS analyses contained in Reference 5 as part of a separate effort (TAC No. M82663). As such, the staff has not yet made a determination regarding the acceptability of the revised boron mixing model used in the generic ATWS analyses. However, the staff understands that, for Fermi-2, the potential effects of the revised boron mixing model are more than compensated by the change (reduction) in reactor water level throughout the ATWS event. Therefore, use of the revised boron mixing model will not have a significant effect upon the results of the Fermi-2 analysis. Based upon this information, the staff concludes that the response of Fermi-2 to ATWS events will remain acceptable after uprate.

3.7.4 Station Blackout (SBO)

The licensee indicated that the plant response and coping capabilities for an SBO event are impacted slightly by operation at the uprated power level due to the increase in the operating temperature of the primary coolant system, increase in the decay heat, and increase in the main steam safety/relief valve setpoints. The licensee evaluated the impact of these increases to the condensate water requirement and the temperature heat-up in the areas which contain equipment necessary to mitigate the SBO event. The licensee concluded that no changes to the required coping time and to the systems and equipment used to respond to an SBO event are required.

Based on its review, the staff finds that the impact to an SBO event due to the operation of uprated power level will be insignificant and that no changes to the required coping time and to the systems and equipment used to respond to an SBO event are required.

3.8 Additional Aspects of Power Uprate

3.8.1 High Energy Line Break (HELB)

The slight increase in the operating pressure and temperature caused by the power uprate results in a small increase in the mass and energy release rates following HELB. This results in a small increase in the subcompartment pressure and temperature profiles and a negligible change in the humidity profile. The licensee has reevaluated the HELB for the main steam system, feedwater system, high pressure coolant injection system, reactor core isolation cooling system, and reactor water cleanup system. As a result of this reevaluation, the licensee has concluded that the affected building and cubicles that support the safety-related functions are designed to withstand the resulting pressure and thermal loading following a high-energy line break. The staff has reviewed the results of the licensee's re-analysis and finds them acceptable.

The licensee has also evaluated the calculations supporting the disposition of potential targets of pipe whip and jet impingement from the postulated HELBs and determined that they are adequate for the safe shutdown effects in the uprated power condition. Existing pipe whip restraints and jet impingement shields and their supporting structures have also been determined to be adequate for the power uprate.

The licensee also verified that the power uprate has no impact on the moderate-energy line crack evaluation. Based on a review of the moderate-energy systems involved, the staff also concludes that the original moderate-energy line break analysis is not affected.

Based on the above, the staff concludes that the analyses for high-energy line breaks outside containment and moderate-energy pipe breaks are acceptable for the proposed power uprate.

3.8.2 Equipment Qualification

3.8.2.1 Environmental Qualification of Electrical Equipment

The licensee evaluated safety-related electrical equipment to assure qualification for the normal and accident conditions expected in the areas where the equipment is located. For equipment located inside containment, the licensee indicated that current accident and normal design conditions for temperature, pressure, and humidity are unchanged for power uprate. Accident and normal radiation levels increase in proportion to the increase in power. For equipment outside containment, normal operational temperature, pressure, and humidity conditions are unchanged. However, accident temperatures increase less than 5 °F and pressures increase less than 1 psi. Normal operational and accident radiation levels increase in relationship to the increase in power.

The licensee indicated, based on the evaluation, that no safety-related equipment was identified as unqualified for power uprate environmental conditions. However, the qualified life of certain identified equipment will be reduced based on increased environments. Documentation to direct early replacement of equipment prior to exceeding its qualified life will be based on aging analysis.

Based on our review, the staff finds the licensee's approach to qualification of safety-related electrical equipment for power uprate conditions acceptable.

3.8.2.2 Environmental Qualification of Mechanical Equipment with Non-Metallic Components

The licensee indicated that operation at the uprated power level increases the normal process temperatures up to 6 °F. As in the case of electrical equipment, normal operational and accident radiation levels also increase slightly due to uprate.

The licensee indicated that their reevaluation of equipment in this category has not identified any equipment which is unqualified for power uprate environmental conditions. However, the qualified life of certain identified equipment will be reduced based on increased environments. Documentation to direct early replacement of equipment prior to exceeding its qualified life will be based on aging analysis.

Based on our review, the staff finds the licensee's approach to qualification of mechanical equipment with non-metallic components for power uprate conditions acceptable.

3.8.2.3 Mechanical Component Design Qualification

Based upon review of the proposed power uprate amendment, the staff finds that the original seismic and dynamic qualification of the safety-related mechanical and electrical equipment is not affected by the power uprate conditions for the following reasons:

(1) seismic loads are unchanged by power uprate,

(2) the original LOCA load conditions bound the power uprate conditions as stated in Section 3.2.3,

(3) the slight increase (about 1 to 2 percent) in AP, JR and SRV loads as delineated in Section 3.2.3 has a negligible effect on equipment dynamic response, and

(4) no new pipe break locations resulted from the uprated conditions.

3.8.3 Startup Testing

The licensee has committed to perform a startup testing program as described in GE LTR NEDC-31897P. In particular, the licensee's startup testing program for power uprate includes performance of acceptance testing of the RCIC and HPCI systems, system testing of process control systems such as the feedwater flow and main steam pressure control systems. Additionally, steady-state operation data will be taken during various portions of the power ascension to the higher licensed power level so that predicted equipment performance characteristics can be verified. The conduct of the startup testing program will be done in accordance with the licensee's procedures. Corrective actions for equipment failing to pass the performance testing will be made in accordance with the procedures or Technical Specification action statements, as appropriate.

3.9 Evaluation of Impact on Responses to Generic Communications

In Reference 3, GE provided an assessment of the impact of power uprate on licensee responses to generic NRC and industry communications. GE reviewed both NRC and industry communications to determine whether parameter changes associated with power uprate could potentially affect previous licensee commitments or responses. A large number of documents were reviewed (over 3,000 items), with GE identifying only a small number of these as being potentially affected by power uprate. The list of affected topics was then divided into those which could be bounded generically by GE, and those which would require plant-specific reevaluation. The NRC staff audited the GE assessment in December of 1991, and approved the assessment in Reference 4.

In addition to assessing those items requiring a plant-specific reevaluation, the licensee also reviewed the potential effects of uprate on internal commitments, such as Deviation Event Reports (DERs), Temporary Modifications (TMs), and the Regulatory Action Commitment Tracking System (RACTS) database. The licensee found no additional commitments which require modification to accommodate power uprate.

4. 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

Pursuant to 10 CFR 51.21, 51.32 and 51.35 an Environmental Assessment and Finding of No Significant Impact has been prepared and published in the Federal Register on August 31, 1992, (57 FR 39407). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 CONCLUSION

The Commission 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.

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Date: September 9, 1992

7.0 REFERENCES

(1) GE Licensing Topical Report, NEDC-31897P-1, "Generic Guidelines for General Electric Boiling Water Reactor (BWR) Power Uprate," June 1991. (Proprietary information. Not publicly available.)

(2) NRC letter, "Staff Position Concerning General Electric Boiling Water Reactor Power Uprate Program (TAC NO. M79384)," dated September 30, 1991.

(3) GE Licensing Topical Report NEDC-31984P, "Generic Evaluations of General Electric Boiling Water Reactor (BWR) Power Uprate," July 1991, Volumes I and II. (Proprietary information. Not publicly available.)

(4) NRC letter, "Staff Safety Evaluation of General Electric Boiling Water Reactor Power Uprate Generic Analyses (TAC NO. M81253)," dated July 31, 1992.

(5) GE Licensing Topical Report NEDC-31984P, Supplement 1, dated October 1991. (Proprietary information. Not publicly available.)

(6) GE Topical Report NEDC-31336P, "General Electric Instrument Setpoint Methodology," October 1986. (Proprietary information. Not publicly available.)

(7) ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, 1968 Edition with Summer 1969 Addenda.

(8) ASME Boiler and Pressure Vessel Code, Section III, Division 1, 1983 Edition with Winter 1984 Addenda.

(9) Detroit Edison letter, NRC-91-0102, "Proposed License Amendment - Uprated Power Operation," dated September 24, 1991.

(10) Detroit Edison letter, NRC-92-0048, "Revision to Proposed License Amendment - Uprated Power Operation," dated April 30, 1992.

(11) Detroit Edison letter, NRC-92-0018, "Burnup and Enrichment Levels of 10 CFR 51.52," dated February 24, 1992.

(12) Detroit Edison letter, NRC-92-0038, "Detroit Edison Response to NRC Mechanical Engineering Branch Questions on Fermi-2 Power Uprate Submittal," dated March 23, 1992.

(13) Detroit Edison letter, NRC-92-0043, "Detroit Edison Response to NRC Instrumentation & Controls Branch Questions on Fermi-2 Power Uprate Submittal," dated March 26, 1992.

(14) Detroit Edison letter, NRC-92-0050, "Detroit Edison Response to NRC Reactor Systems Branch Questions on Fermi-2 Power Uprate Submittal," dated April 23, 1992

(15) Detroit Edison letter, NRC-92-0065, "Detroit Edison Response to NRCInstrumentation & Controls Systems Branch (ICSB) Additional Questions on Fermi-2 Power Uprate Submittal (TAC NO. 82102)," dated May 11, 1992.

(16) Detroit Edison letter, NRC-92-0098, "Additional Information Requested by the NRC Project Manager for Preparation of an Environmental Impact Statement for the Fermi-2 Power Uprate License Amendment Request," dated August 12, 1992.

(17) Detroit Edison letter, NRC-92-0095, "Detroit Edison Response to NRC Plant System Branch (SPLB) Verbal Request for Additional Information on Fermi-2 Power Uprate Submittal (TAC NO. 82102)," dated August 13, 1992.

(18) GE letter, "Response to NRC Questions on the Generic Power Uprate ATWS Analysis," dated September 1, 1992.