Docket No. 50-341

September 9, 1992

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

Dear Mr. Orser:

SUBJECT: FERMI-2 - AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-43 (TAC NO. M82102)

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the License and Plant Technical Specifications in response to your letter dated September 24, 1991, and modified January 31, and April 30, 1992.

The amendment changes 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 BWR power uprate program established by the General Electric Company and approved by the NRC staff in a letter dated September 30, 1991.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,
Original signed by

Timothy G. Colburn, Sr. Project Manager
Project Directorate - III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 87 to NPF-43
2. Safety Evaluation

*SEE PREVIOUS CONCURRENCE

OFFICE	LA:PD31	PM:PD31	BC:SICB*	BC:SPLB*	BC:SRXB
NAME	MShuttleworth	TColburn:dmy	SNewberry	CMcCracken	RJones
DATE	9/1/92	9/1/92	8/18/92	8/20/92	9/1/92
OFFICE	BC:PRPB*	BC:EMEB*	OGC*	D:PD31	AD:RIII
NAME	LCunningham	JNorberg	MYoung	LMarsh	JZwolinski
DATE	8/13/92	8/14/92	8/31/92	9/1/92	9/1/92
OFFICE	D:DST	D:DRPW	ADT	ADPR	D:NRR
NAME	AThadani	BBoger	WRussell	JPartlow	TEMurley
DATE	9/4/92	9/2/92	9/4/92	9/4/92	9/9/92

OFFICIAL RECORD COPY
FILENAME: C:\WP\WPDOCS\FERMI\FE.AMD

CP-1

9/4/92

DF01
1/0

DATED: September 9, 1992

AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-43-FERMI-2

Docket File
NRC & Local PDRs
PDIII-1 Reading
Fermi Plant File
B. Boger
J. Zwolinski
L. Marsh
M. Shuttleworth
T. Colburn
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (4), P-137
Wanda Jones, MNBB-7103
C. Grimes, 11/F/23
K. Eccleston, SPLB
T. Chandrasekaran, 8/D/1
G. Hubbard, 8/D/1
R. Frahm, SRXB
R. Goel, SPLB
J. Kudrick, SPLB
R. Scholl, SICB
R. Stransky, PD31
C. Wu, EMEB
ACRS (10)
OPA
OC/LFMB
W. Shafer, R-III

cc: Plant Service list

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

September 9, 1992

Docket No. 50-341

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

Dear Mr. Orser:

SUBJECT: FERMI-2 - AMENDMENT NO. 87 TO FACILITY OPERATING LICENSE NO. NPF-43 (TAC NO. M82102)

The Commission has issued the enclosed Amendment No. 87 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the License and Plant Technical Specifications in response to your letter dated September 24, 1991, and modified January 31, and April 30, 1992.

The amendment changes 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 BWR power uprate program established by the General Electric Company and approved by the NRC staff in a letter dated September 30, 1991.

A copy of our Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

Timothy G. Colburn

Timothy G. Colburn, Sr. Project Manager
Project Directorate - III-1
Division of Reactor Projects - III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 87 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

Mr. William Orser
Detroit Edison Company

Fermi-2

cc:

John Flynn, Esquire
Senior Attorney
Detroit Edison Company
2000 Second Avenue
Detroit, Michigan 48226

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

Mr. Stan Stasek
U.S. Nuclear Regulatory Commission
Resident Inspector Office
6450 W. Dixie Highway
Newport, Michigan 48166

Monroe County Office of Civil
Preparedness
963 South Raisinvile
Monroe, Michigan 48161

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

Mr. A. Cecil Settles
Director - Nuclear Licensing
Detroit Edison Company
Fermi-2
6400 North Dixie Highway
Newport, Michigan 48166

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

DETROIT EDISON COMPANY

FERMI-2

DOCKET NO. 50-341

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 87
License No. NPF-43

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

   A. The application for amendment by the Detroit Edison Company (the licensee) dated September 24, 1991, and modified January 31, and April 30, 1992, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, paragraph 2.C.(1) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

   (1) Maximum Power Level

   DECo is authorized to operate the facility at reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

3. Further, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

   (2) Technical Specifications and Environmental Protection Plan

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 87 , and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

4. This license amendment is effective as of beginning of the third refueling outage, currently scheduled for September 12, 1992, with full implementation prior to startup from the third refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Pages 3 and 4 of license*
2. Changes to the Technical Specifications

Date of Issuance: September 9, 1992

*Pages 3 and 4 are attached, for convenience, for the composite license to reflect this change.

(4) DECo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use at any time any byproduct, source and special nuclear material such as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(5) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(6) DECo, pursuant to the Act and 10 CFR Parts 30, 40 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This license shall be deemed to contain and is subject to the conditions specified in the Commission's regulations set forth in 10 CFR Chapter I and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified of incorporated below:

(1) Maximum Power Level

DECo is authorized to operate the facility at the reactor core power levels not in excess of 3430 megawatts thermal (100% power) in accordance with the conditions specified herein and in Attachment 1 to this license. The items identified in Attachment 1 to this license shall be completed as specified. Attachment 1 is hereby incorporated into this license.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A as revised through Amendment No. 87 and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Antitrust Conditions

DECo shall abide by the agreements and interpretations between it and the Department of Justice relating to Article I, Paragraph 3 of the Electric Power Pool Agreement between Detroit Edison Company and

Consumers Power Company as specified in a letter from DECo to the Director of Regulation, dated August 13, 1971, and the letter from Richard W. McLaren, Assistant Attorney General, Antitrust Division, U.S. Department of Justice, to Bertram H. Schur, Associate General Counsel, Atomic Energy Commission, dated August 16, 1971.

(4) Safety/Relief Valve In-Plant Testing (Section 3.8.1, SSER #5)*

Prior to completing the startup test program, DECo shall perform a series of in-plant tests of the safety/relief valves (SRVs). The acceptance criteria for these tests are contained in Section 2.13.9, "SRV Load Assessment by In-Plant Tests" of NUREG-0661, "NRC Acceptance Criteria for the Mark I Containment Long-Term Program." The results of these tests shall be reported to the NRC staff within six months of completing this test series.

(5) Suppression Pool Temperature Measurements (Section 3.8.1, SSER #5)

DECo shall accomplish during the first fuel cycle, all the tasks described in its letter dated March 6, 1985, regarding the series of SRV tests which will confirm its methodology for measuring the suppression pool bulk temperature.

(6) Environmental Qualification (Section 3.11, SSER #5)

No later than November 30, 1985, DECo shall environmentally qualify all electrical equipment according to the provisions of 10 CFR 50.49.

(7) Control Room Habitability (Section 6.4.1, SSER #6)

Prior to startup following the first refueling outage, DECo shall provide assurance to the NRC staff that potential contamination pathways through those portions of the control room air-conditioning system which are external to the control room zone will not have a significant adverse impact on control room habitability, or will propose a technical specification which provides for periodic leakage testing to assure the integrity of those external portions of the control room air-conditioning system.

---

*The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the license condition is discussed.

ATTACHMENT TO LICENSE AMENDMENT NO. 87

FACILITY OPERATING LICENSE NO. NPF-43

DOCKET NO. 50-341

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain a vertical line indicating the area of change. Pages marked with an asterisk (*) indicate overleaf pages and do not contain changes.

REMOVE	INSERT
1-5	1-5
2-4	2-4
3/4 1-20	3/4 1-20
3/4 1-21	3/4 1-21
3/4 2-1	3/4 2-1
3/4 3-4	3/4 3-4
3/4 3-5	3/4 3-5
3/4 3-15	3/4 3-15
3/4 3-44	3/4 3-44
3/4 4-1	3/4 4-1
3/4 4-2	3/4 4-2
3/4 4-6a	3/4 4-6a
3/4 4-7	3/4 4-7
3/4 4-8*	3/4 4-8*
3/4 4-9*	3/4 4-9*
3/4 4-10	3/4 4-10
3/4 4-21	3/4 4-21
3/4 4-23	3/4 4-23
3/4 4-24*	3/4 4-24*
3/4 4-31	3/4 4-31
3/4 5-3*	3/4 5-3*
3/4 5-4	3/4 5-4
3/4 7-13*	3/4 7-13*
3/4 7-14	3/4 7-14
3/4 8-23	3/4 8-23
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1a	B 3/4 2-1a
B 3/4 2-4	B 3/4 2-4
B 3/4 4-1	B 3/4 4-1
B 3/4 4-1a	B 3/4 4-1a
B 3/4 4-8	B 3/4 4-8
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2	B 3/4 6-2
B 3/4 6-3	B 3/4 6-3
B 3/4 6-4	B 3/4 6-4
B 3/4 7-5	B 3/4 7-5
6-21	6-21

2. Closed by at least one manual valve, blank flange, or deactivated automatic valve secured in its closed position, except as provided in Table 3.6.3-1 or Specification 3.6.3.

b. All primary containment equipment hatches are closed and sealed.

c. Each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. The primary containment leakage rates are within the limits of Specification 3.6.1.2.

e. The suppression chamber is in compliance with the requirement of Specification 3.6.2.1.

f. The sealing mechanism associated with each primary containment penetration, e.g., welds, bellows, or O-rings, is OPERABLE.

g. The suppression chamber to reactor building vacuum breakers are in compliance with Specification 3.6.4.2.

THE PROCESS CONTROL PROGRAM

1.30 The PROCESS CONTROL PROGRAM (PCP) shall contain the current formulas, sampling, analyses, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, State regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

PURGE - PURGING

1.31 PURGE or PURGING is the controlled process of discharging air or gas from a confinement to maintain temperature, pressure, humidity, concentration or other operating condition, in such a manner that replacement air or gas is required to purify the confinement.

RATED THERMAL POWER

1.32 RATED THERMAL POWER shall be a total reactor core heat transfer rate to the reactor coolant of 3430 MWT.

TABLE 2.2.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor, Neutron Flux - High	≤ 120/125 divisions of of full scale	≤ 122/125 divisions of full scale
2. Average Power Range Monitor:		
a. Neutron Flux-Upscale, Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-Upscale		
1) During two recirculation loop operation:		
a. Flow Biased	≤ 0.63 W+61.4%, with a maximum of	≤ 0.63 W+64.3%, with a maximum of
b. High Flow Clamped	≤ 113.5% of RATED THERMAL POWER	≤ 115.5% of RATED THERMAL POWER
2) During single recirculation loop operation:		
a. Flow Biased	≤ 0.63W+56.3%,**	≤ 0.63W+59.2%,**
b. High Flow Clamped	NA	NA
c. Fixed Neutron Flux-Upscale	≤ 118% of RATED THERMAL POWER	≤ 120% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1093 psig	≤ 1113 psig
4. Reactor Vessel Low Water Level - Level 3	≥ 173.4 inches*	≥ 171.9 inches

*See Bases Figure B 3/4 3-1.

**During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.1% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

b. At least once per 31 days by:

   1. Verifying the continuity of the explosive charge.

   2. Determining that the concentration of boron in solution is within the limits of Figure 3.1.5-1 by chemical analysis.*

   3. Verifying that each valve (manual, power-operated, or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. Demonstrating that, when tested pursuant to Specification 4.0.5, the minimum flow requirement of 41.2 gpm at a pressure of greater than or equal to 1215 psig is met.

d. At least once per 18 months during shutdown by:

   1. Initiating one of the standby liquid control system loops, including an explosive valve, and verifying that a flow path from the pumps to the reactor pressure vessel is available by pumping demineralized water into the reactor vessel. The replacement charge for the explosive valve shall be from the same manufactured batch as the one fired or from another batch which has been certified by having one charge of that batch successfully fired. Both injection loops shall be tested in 36 months.

   2. Demonstrating that the pump relief valve setpoint is less than or equal to 1400 psig and verifying that the relief valve does not actuate during recirculation to the test tank.

   3. Demonstrating that all piping between the storage tank and the explosive valves is unblocked by pumping from the storage tank to the test tank and then draining and flushing the piping with demineralized water.**

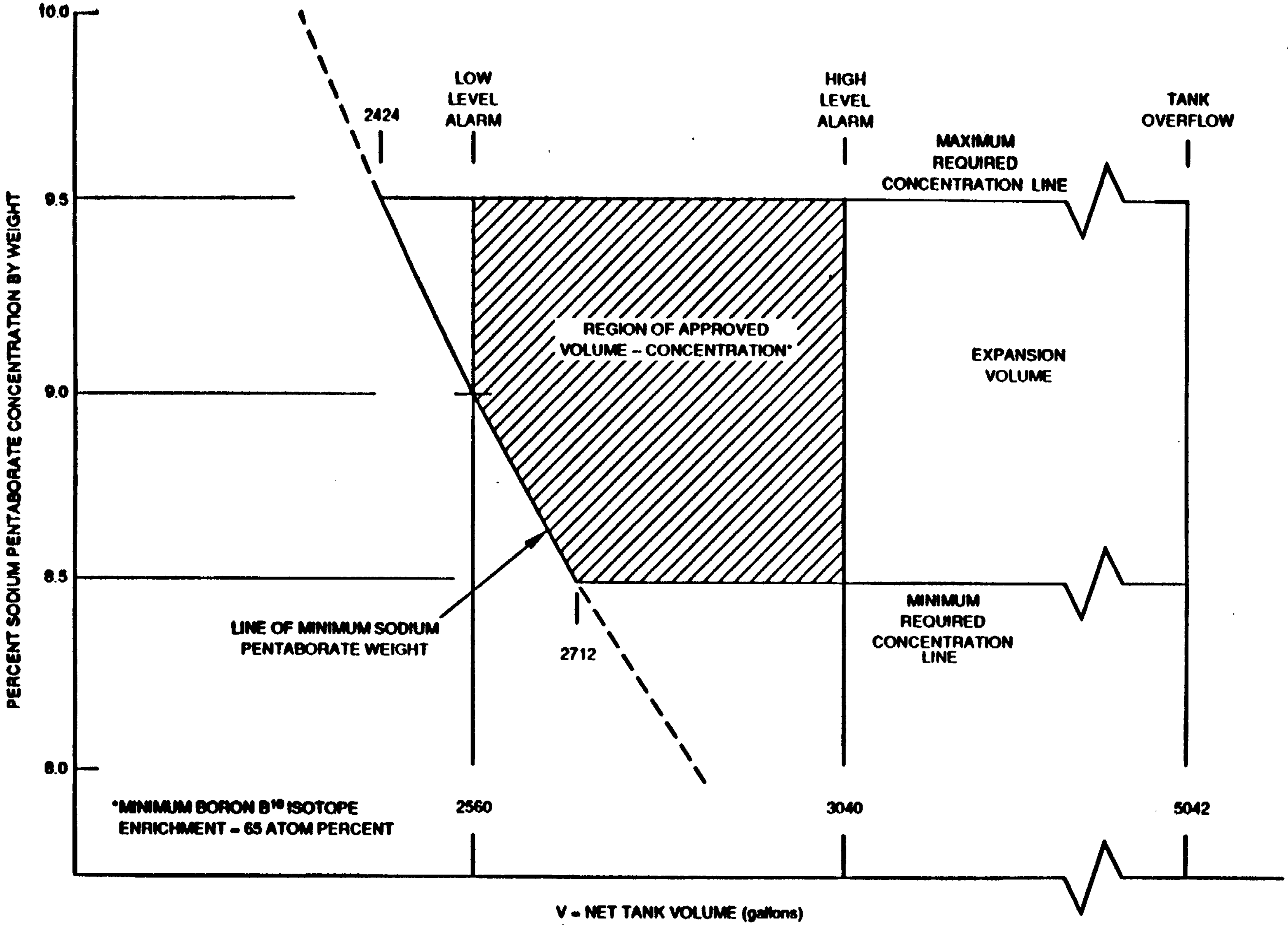
   4. Demonstrating that the storage tank heaters are OPERABLE for mixing by verifying the expected temperature rise of the sodium pentaborate solution in the storage tank after the heaters are energized.

e. At least once per 18 months sample and analyze the sodium pentaborate solution to verify that the Boron-10 Isotope enrichment exceeds 65 atom percent.

---

*This test shall also be performed anytime water or boron is added to the solution or when the solution temperature drops below the 48°F limit.

**This test shall also be performed whenever the solution temperature drops below the 48°F limit and may be performed by any series of sequential, overlapping or total flow path steps such that the entire flow path is included.



SODIUM PENTABORATE VOLUME/CONCENTRATION REQUIREMENTS

FIGURE 3.1.5-1

3/4.2 POWER DISTRIBUTION LIMITS

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.1 All AVERAGE PLANAR LINEAR HEAT GENERATION RATES (APLHGRs) shall not exceed:

a. The MAPLHGR limit which has been approved for the respective fuel and lattice type as a function of the average planar exposure (as determined by the NRC approved methodology described in GESTAR-II), or

b. When hand calculations are required, the most limiting lattice type MAPLHGR limit as a function of the average planar exposure shown in the CORE OPERATING LIMITS REPORT (COLR) for the applicable fuel type.

APPLICABILITY: OPERATIONAL CONDITION 1, when THERMAL POWER is greater than or equal to 25% of RATED THERMAL POWER.

ACTION:

With an APLHGR exceeding the above limits, initiate corrective action within 15 minutes and restore APLHGR to within the required limits within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.1 All APLHGRs shall be verified to be equal to or less than the limits required by Specification 3.2.1:

a. At least once per 24 hours,

b. Within 12 hours after completion of a THERMAL POWER increase of at least 15% of RATED THERMAL POWER, and

c. Initially and at least once per 12 hours when the reactor is operating with a LIMITING CONTROL ROD PATTERN for APLHGR.

d. The provisions of Specification 4.0.4 are not applicable.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION STATEMENTS

ACTION 1	-	Be in at least HOT SHUTDOWN within 12 hours.
ACTION 2	-	Verify all insertable control rods to be inserted in the core and lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 3	-	Suspend all operations involving CORE ALTERATIONS and insert all insertable control rods within 1 hour.
ACTION 4	-	Be in at least STARTUP within 6 hours.
ACTION 5	-	Be in STARTUP with the main steam line isolation valves closed within 6 hours or in at least HOT SHUTDOWN within 12 hours.
ACTION 6	-	Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to $\leq$ 161.9 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER, within 2 hours.
ACTION 7	-	Verify all insertable control rods to be inserted within 1 hour.
ACTION 8	-	Lock the reactor mode switch in the Shutdown position within 1 hour.
ACTION 9	-	Suspend all operations involving CORE ALTERATIONS, and insert all insertable control rods and lock the reactor mode switch in the Shutdown position within 1 hour.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 6 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) This function shall be automatically bypassed when the reactor mode switch is in the Run position.

(c) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1, the "shorting links" shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(d) When the "shorting links" are removed, the Minimum OPERABLE Channels Per Trip System is 4 APRMs, 6 IRMs and per Specification 3.9.2, 2 SRMs.

(e) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(f) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(g) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(h) This function is not required to be OPERABLE when PRIMARY CONTAINMENT INTEGRITY is not required.

(i) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(j) This function shall be automatically bypassed when turbine first stage pressure is ≤ 161.9 psig, equivalent to THERMAL POWER less than 30% of RATED THERMAL POWER.

---

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

TABLE 3.3.2-2

ISOLATION ACTUATION INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. PRIMARY CONTAINMENT ISOLATION		
a. Reactor Vessel Low Water Level		
1) Level 3	≥ 173.4 inches*	≥ 171.9 inches
2) Level 2	≥ 110.8 inches*	≥ 103.8 inches
3) Level 1	≥ 31.8 inches*	≥ 24.8 inches
b. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig
c. Main Steam Line		
1) Radiation - High	≤ 3.0 x full power background	≤ 3.6 x full power background
2) Pressure - Low	≥ 756 psig	≥ 736 psig
3) Flow - High	≤ 115.4 psid	≤ 118.4 psid
d. Main Steam Line Tunnel Temperature - High	≤ 200°F	≤ 206°F
e. Condenser Pressure - High	≤ 6.85 psia	≤ 7.05 psia
f. Turbine Bldg. Area Temperature - High	≤ 200°F	≤ 206°F
g. Deleted		
h. Manual Initiation	NA	NA

TABLE 3.3.6-2

CONTROL ROD BLOCK INSTRUMENTATION SETPOINTS

TRIP FUNCTION	TRIP SETPOINT	ALLOWABLE VALUE
1. ROD BLOCK MONITOR		
a. Upscale	As specified in the CORE OPERATING LIMITS REPORT	As specified in the CORE OPERATING LIMITS REPORT
b. Inoperative	NA	NA
c. Downscale	≥ 94% of Reference Level	≥ 92.3% of Reference Level
2. APRM		
a. Flow Biased Neutron Flux - High		
1) During two recirculation loop operation	≤ 0.63 W + 55.6%* with a maximum of 108%	≤ 0.63 W + 58.5%* with a maximum of 110%
2) During single recirculation loop operation	≤ 0.63 W + 50.5%#*	≤ 0.63 W + 53.4%#*
b. Inoperative	NA	NA
c. Downscale	≥ 5% of RATED THERMAL POWER	≥ 3% of RATED THERMAL POWER
d. Neutron Flux - Upscale, Setdown	≤ 12% of RATED THERMAL POWER	≤ 14% of RATED THERMAL POWER
3. SOURCE RANGE MONITORS		
a. Detector not full in	NA	NA
b. Upscale	≤ 1.0 x 10^5 cps	≤ 1.6 x 10^5 cps
c. Inoperative	NA	NA
d. Downscale	≥ 3 cps**	≥ 2 cps**

*The APRM rod block function is varied as a function of recirculation loop drive flow (W).

**May be reduced to ≥ 0.7 cps provided the signal-to-noise ratio ≥ 20.

#During single recirculation loop operation, rather than adjusting the APRM Flow Biased Setpoints to comply with the single loop values, the gain of the APRMs may be adjusted for a period not to exceed 72 hours such that the final APRM readings are at least 5.1% of rated power greater than 100% times FRTP, provided that the adjusted APRM readings do not exceed 100% of RATED THERMAL POWER and a notice of adjustment is posted on the reactor control panel.

3/4.4 REACTOR COOLANT SYSTEM
3/4.4.1 RECIRCULATION SYSTEM
RECIRCULATION LOOPS
LIMITING CONDITION FOR OPERATION

3.4.1.1 Two reactor coolant system recirculation loops shall be in operation.

APPLICABILITY: OPERATIONAL CONDITIONS 1 and 2*.

ACTION:

a. With one reactor coolant system recirculation loop not in operation:

   1. Within 4 hours:

      a) Place the individual recirculation pump flow controller for the operating recirculation pump in the Manual mode.

      b) Reduce THERMAL POWER to less than or equal to 67.2% of RATED THERMAL POWER.

      c) Limit the speed of the operating recirculation pump to less than or equal to 75% of rated pump speed.

      d) Increase the MINIMUM CRITICAL POWER RATIO (MCPR) Safety Limit by 0.01 to 1.08 per Specification 2.1.2.

      e) Reduce the Average Power Range Monitor (APRM) Scram and Rod Block Trip Setpoints and Allowable Values to those applicable for single recirculation loop operation# per Specifications 2.2.1 and 3.3.6.

      f) Perform Surveillance Requirement 4.4.1.1.4 if THERMAL POWER is less than or equal to 30% of RATED THERMAL POWER or the recirculation loop flow in the operating loop is less than or equal to 50% of rated loop flow.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours.

b. With no reactor coolant system recirculation loop in operation while in OPERATIONAL CONDITION 1, immediately place the Reactor Mode Switch in the SHUTDOWN position.

c. With no reactor coolant system recirculation loops in operation, while in OPERATIONAL CONDITION 2, initiate measures to place the unit in at least HOT SHUTDOWN within the next 6 hours.

*See Special Test Exception 3.10.4.
#APRM gain adjustments may be made in lieu of adjusting the APRM Flow Biased Setpoints to comply with the single loop values for a period of up to 72 hours.

SURVEILLANCE REQUIREMENTS

4.4.1.1.1 Each pump discharge valve shall be demonstrated OPERABLE by cycling each valve through at least one complete cycle of full travel during each STARTUP* prior to THERMAL POWER exceeding 25% of RATED THERMAL POWER.

4.4.1.1.2 Each pump MG set scoop tube mechanical and electrical stop shall be demonstrated OPERABLE with overspeed setpoints less than or equal to 110% and 107%, respectively, of rated core flow, at least once per 18 months.

4.4.1.1.3 With one reactor coolant system recirculation loop not in operation, at least once per 12 hours verify that:

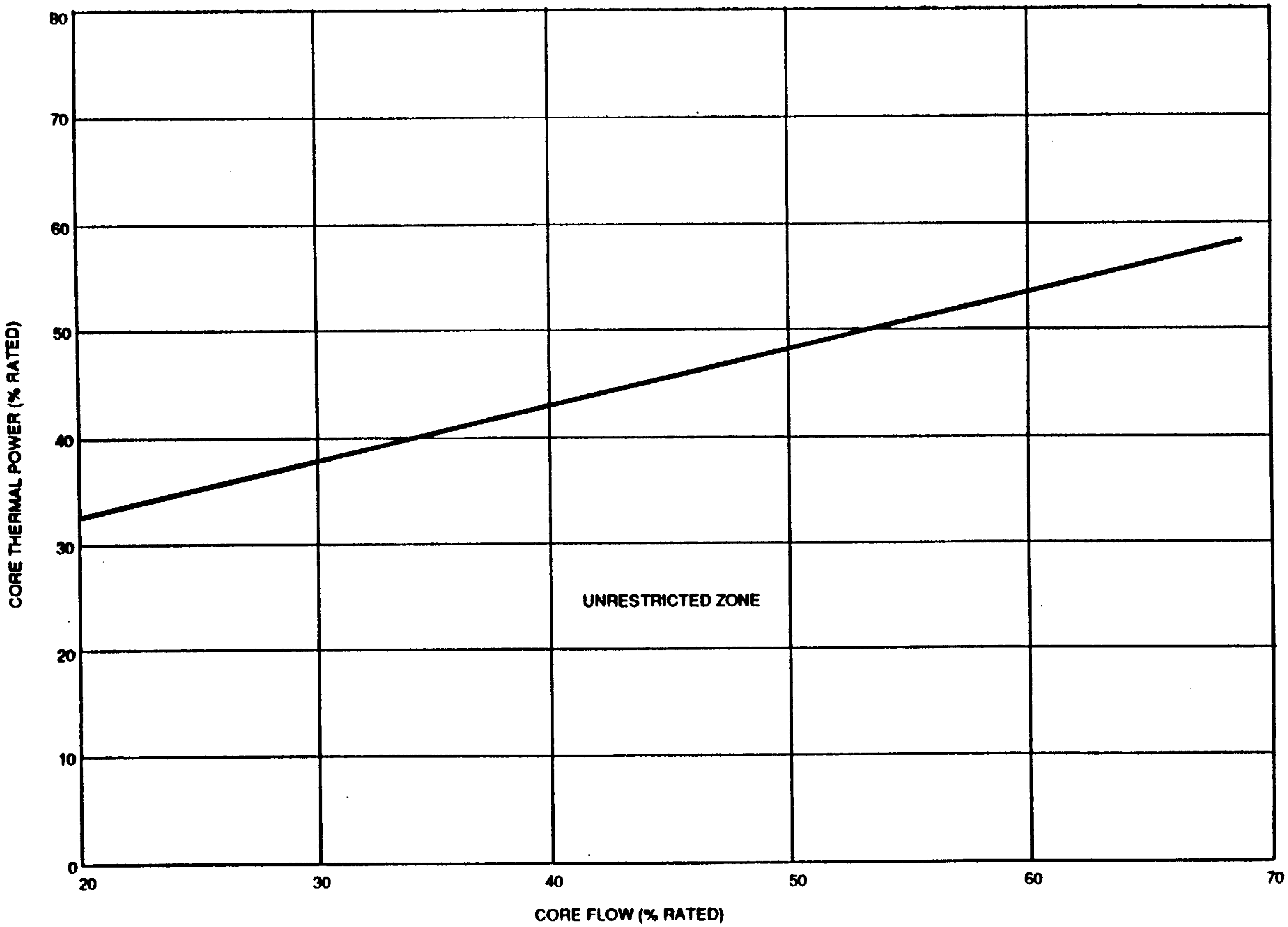
a. THERMAL POWER is less than or equal to 67.2% of RATED THERMAL POWER, and
b. The individual recirculation pump flow controller for the operating recirculation pump is in the Manual mode, and
c. The speed of the operating recirculation pump is less than or equal to 75% of rated pump speed.

4.4.1.1.4 With one reactor coolant system loop not in operation with THERMAL POWER less than or equal to 30% of RATED THERMAL POWER or with recirculation loop flow in the operating loop less than or equal to 50% of rated loop flow, verify the following differential temperature requirements are met within no more than 15 minutes prior to either THERMAL POWER increase or recirculation flow increase:

a. Less than or equal to 145°F between reactor vessel steam space coolant and bottom head drain line coolant, and
b. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the coolant in the reactor pressure vessel**, and
c. Less than or equal to 50°F between the reactor coolant within the loop not in operation and the operating loop.**

*If not performed within the previous 31 days.

**Requirement does not apply when the recirculation loop not in operation is isolated from the reactor pressure vessel.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.1.4-1

3/4.4.2 SAFETY/RELIEF VALVES

SAFETY/RELIEF VALVES

LIMITING CONDITION FOR OPERATION

3.4.2.1 The safety valve function of at least 11 of the following reactor coolant system safety/relief valves shall be OPERABLE with the specified code safety valve function lift settings:*

5 safety/relief valves @ 1135 psig ±1%
5 safety/relief valves @ 1145 psig ±1%
5 safety/relief valves @ 1155 psig ±1%

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the safety valve function of less than 11 of the above safety/relief valves OPERABLE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With one or more safety/relief valves stuck open, provided that suppression pool average water temperature is less than 95°F, close the stuck open safety/relief valve(s); if unable to close the stuck open valve(s) within 2 minutes or if suppression pool average water temperature is 95°F or greater, place the reactor mode switch in the Shutdown position.

c. With one or more safety/relief valve position indicators inoperable, restore the inoperable indicator(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.1.1 The valve position indicator for each safety/relief valve shall be demonstrated OPERABLE with the pressure setpoint of each of the tail-pipe pressure switches verified to be 30 ± 5 psig by performance of a CHANNEL CALIBRATION at least once per 18 months.

4.4.2.1.2 At least 1/2 of the safety relief valves shall be set pressure tested at least once per 18 months, such that all 15 safety relief valves are set pressure tested at least once per 40 months.

*The lift setting pressure shall correspond to ambient conditions of the valves at nominal operating temperatures and pressures.

REACTOR COOLANT SYSTEM

SAFETY/RELIEF VALVES LOW-LOW SET FUNCTION

LIMITING CONDITION FOR OPERATION

3.4.2.2 The low-low set function of the following reactor coolant system safety/relief valves shall be OPERABLE with the following settings:

Valve No.	Low-Low Set Function Setpoint (psig) Open	Low-Low Set Function Setpoint (psig) Close	Low-Low Set Function Allowable Value (psig) Open	Low-Low Set Function Allowable Value (psig) Close
F013A	1017	905	1037	*
F013G	1047	935	1067	*

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the low-low set function of one of the above required reactor coolant system safety/relief valves inoperable, restore the inoperable low-low set function to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the low-low set function of both of the above required reactor coolant system safety/relief valves inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.2.2 The low-low set function pressure actuation instrumentation shall be demonstrated OPERABLE by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days.

b. CHANNEL CALIBRATION, LOGIC SYSTEM FUNCTIONAL TEST and simulated automatic operation of the entire system at least once per 18 months.

*Closing pressure must be at least 100 psi less than actual opening pressure.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITION FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment atmosphere gaseous radioactivity monitoring system channel.

b. The primary containment sump flow monitoring system consisting of:

   1. The drywell floor drain sump level, flow and pump-run-time system, and

   2. The drywell equipment drain sump level, flow and pump-run-time system.

c. The drywell floor drain sump level monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With only two of the above required leakage detection systems OPERABLE, restore the inoperable detection system to OPERABLE status within 30 days; when the required gaseous radioactive monitoring system is inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 24 hours, otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow and drywell floor drain sump level monitoring systems-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

LIMITING CONDITION FOR OPERATION

3.4.3.2 Reactor coolant system leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.

b. 5 gpm UNIDENTIFIED LEAKAGE.

c. 25 gpm total leakage averaged over any 24-hour period.

d. 1 gpm leakage at a reactor coolant system pressure of 1045 ± 10 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.

e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 4-hour period.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any reactor coolant system leakage greater than the limits in b and/or c, above, reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any reactor coolant system pressure isolation valve leakage greater than the above limit, isolate the high pressure portion of the affected system from the low pressure portion within 4 hours by use of at least one other closed manual, deactivated automatic, or check* valve, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

e. With any reactor coolant system UNIDENTIFIED LEAKAGE increase greater than 2 gpm within any 4-hour period, identify the source of leakage increase as not service sensitive Type 304 or 316 austenitic stainless steel within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which has been verified not to exceed the allowable leakage limit at the last refueling outage or after the last time the valve was disturbed, whichever is more recent.

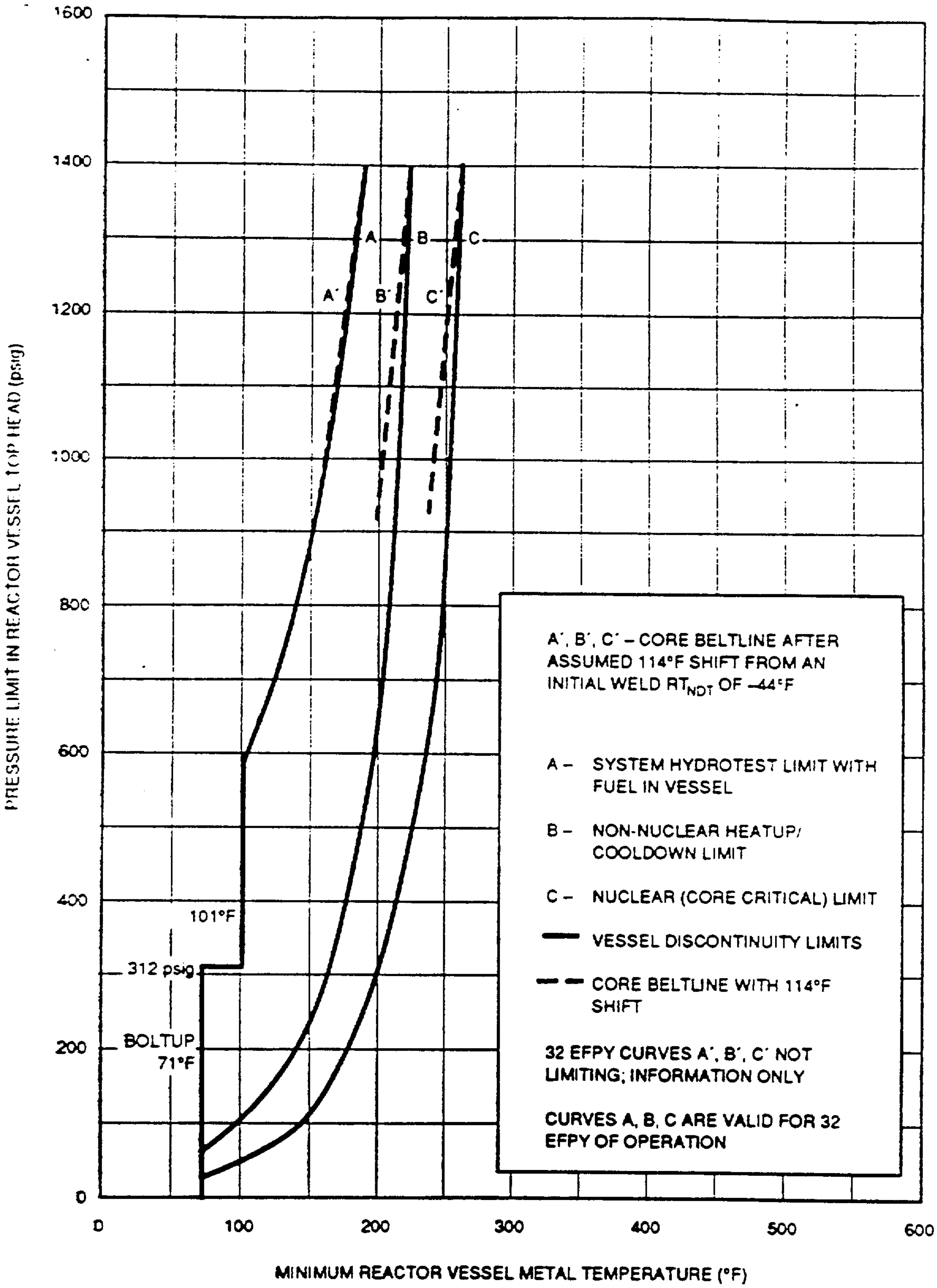


FIGURE 3.4.6.1-1

MINIMUM REACTOR PRESSURE VESSEL METAL TEMPERATURE VS. REACTOR VESSEL PRESSURE

REACTOR COOLANT SYSTEM

REACTOR STEAM DOME

LIMITING CONDITION FOR OPERATION

3.4.6.2 The pressure in the reactor steam dome shall be less than 1045 psig.

APPLICABILITY: OPERATIONAL CONDITIONS 1* and 2*.

ACTION:

With the reactor steam dome pressure exceeding 1045 psig, reduce the pressure to less than 1045 psig within 15 minutes or be in at least HOT SHUTDOWN within 12 hours.

SURVEILLANCE REQUIREMENTS

4.4.6.2 The reactor steam dome pressure shall be verified to be less than 1045 psig at least once per 12 hours.

*Not applicable during anticipated transients.

REACTOR COOLANT SYSTEM

3/4.4.7 MAIN STEAM LINE ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.4.7 Two main steam line isolation valves (MSIVs) per main steam line shall be OPERABLE with closing times greater than or equal to 3 seconds and less than or equal to 5 seconds.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With one or more MSIVs inoperable:

   1. Maintain at least one MSIV OPERABLE in each affected main steam line that is open and within 8 hours, either:

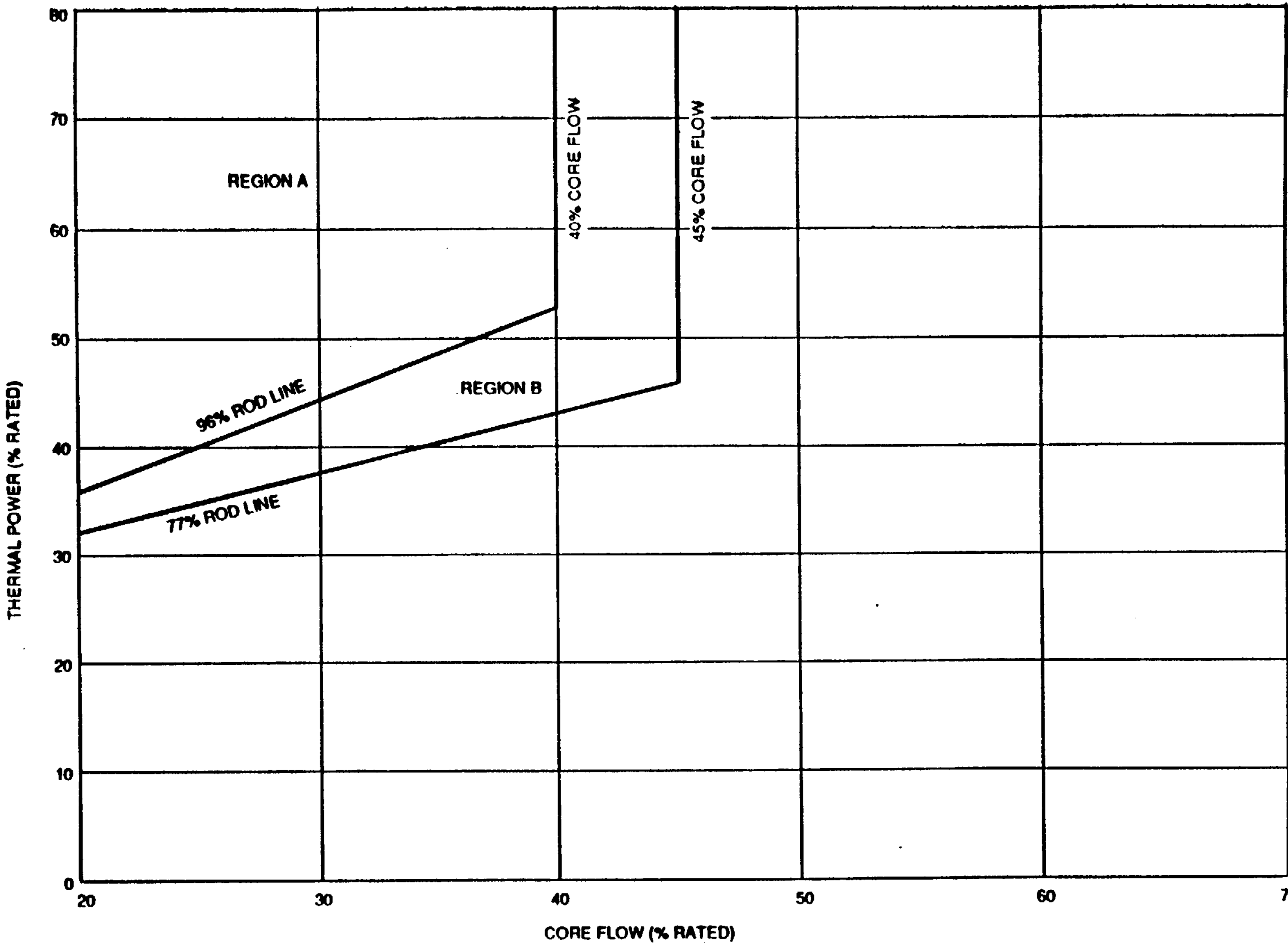
      a) Restore the inoperable valve(s) to OPERABLE status, or

      b) Isolate the affected main steam line by use of a deactivated MSIV in the closed position.

   2. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.7 Each of the above required MSIVs shall be demonstrated OPERABLE by verifying full closure between 3 and 5 seconds when tested pursuant to Specification 4.0.5.



THERMAL POWER VERSUS CORE FLOW

FIGURE 3.4.10-1

ACTION: (Continued)

d. For the ADS:

1. With one of the above required ADS valves inoperable, provided the HPCI system, the CSS and the LPCI system are OPERABLE, restore the inoperable ADS valve to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the next 24 hours.

2. With two or more of the above required ADS valves inoperable, be in at least HOT SHUTDOWN within 12 hours and reduce reactor steam dome pressure to ≤ 150 psig within the next 24 hours.

e. With a CSS header ΔP instrumentation channel inoperable, restore the inoperable channel to OPERABLE status within 72 hours or determine the CSS header ΔP locally at least once per 12 hours; otherwise, declare the associated CSS subsystem inoperable.

f. With an LPCI or CSS system discharge line "keep filled" alarm instrumentation inoperable, perform Surveillance Requirement 4.5.1.a.1.a.

g. In the event an ECCS system is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date. The current value of the usage factor for each affected safety injection nozzle shall be provided in this Special Report whenever its value exceeds 0.70.

SURVEILLANCE REQUIREMENTS

4.5.1 The emergency core cooling systems shall be demonstrated OPERABLE by:

a. At least once per 31 days:

1. For the CSS, the LPCI system, and the HPCI system:

a) Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.
b) Verifying that each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct* position.

2. For the LPCI system, verifying that the cross-tie valve is open.

*Except that an automatic valve capable of automatic return to its ECCS position when an ECCS signal is present may be in position for another mode of operation.

3. For the HPCI system, verifying that the HPCI pump flow controller is in the correct position.

b. Verifying that, when pursuant to Specification 4.0.5:

1. The two CSS pumps in each subsystem together develop a flow of at least 6350 gpm against a test line pressure of greater than or equal to 270 psig, corresponding to a reactor vessel pressure of ≥ 100 psig.

2. Each LPCI pump in each subsystem develops a flow of at least 10,000 gpm against a test line pressure of ≥ 230 psig, corresponding to a reactor vessel to primary containment differential pressure of ≥ 20 psig.

3. The HPCI pump develops a flow of at least 5000 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 1025 +20, -80 psig.*

c. At least once per 18 months:

1. For the CSS, the LPCI system, and the HPCI system, performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test.

2. For the HPCI system, verifying that:

a) The system develops a flow of at least 5000 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 165 + 50, -0 psig.*

b) The suction for the HPCI system is automatically transferred from the condensate storage tank to the suppression chamber on a condensate storage tank water level - low signal and on a suppression chamber - water level high signal.

3. Performing a CHANNEL CALIBRATION of the CSS and the LPCI system discharge line "keep filled" alarm instrumentation.

4. Performing a CHANNEL CALIBRATION of the CSS header ΔP instrumentation and verifying the setpoint to be ≤ the allowable value of 1.0 psid.

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

TABLE 3.7.3-1 (Continued)

SURVEY POINTS FOR SHORE BARRIER*

SURVEY POINT	LOCATION** NORTH-SOUTH	EAST-WEST	DECEMBER 1984 CONTROL ELEVATION
9A	N7529	E5948	583.04
9B	N7531	E5961	582.10
9C	N7531	E5965	579.91
9D	N7526	E5973	575.13
10A	N7612	E5937	583.85
10B	N7610	E5950	582.21
10C	N7618	E5961	582.56
10D	N7616	E5972	576.58
11A	N7721	E5940	583.15
11B	N7721	E5956	582.08
11C	N7718	E5963	579.82
11D	N7722	E5971	576.43
12A	N7814	E5949	581.86
12B	N7809	E5955	581.11
12C	N7814	E5965	578.88
12D	N7815	E5975	577.81

*Measuring reference points are anchored into the capstones using center notched self-drilling bolts.

**See Figure B 3/4.7.3-1 for location sketch.

PLANT SYSTEMS

3/4.7.4 REACTOR CORE ISOLATION COOLING SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.4 The reactor core isolation cooling (RCIC) system shall be OPERABLE with an OPERABLE flow path capable of taking suction from the suppression pool and transferring the water to the reactor pressure vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3 with reactor steam dome pressure greater than 150 psig.

ACTION:

With the RCIC system inoperable, operation may continue provided the HPCI system is OPERABLE; restore the RCIC system to OPERABLE status within 14 days, otherwise be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 150 psig within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.7.4 The RCIC system shall be demonstrated OPERABLE:

a. At least once per 31 days by:

   1. Verifying by venting at the high point vents that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verifying that each valve (manual, power-operated or automatic) in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

   3. Verifying that the pump flow controller is in the correct position.

b. At least once per 92 days by verifying that the RCIC pump develops a flow of greater than or equal to 600 gpm in the test flow path with a system head corresponding to reactor vessel operating pressure including injection line losses when steam is being supplied to the turbine at 1025 + 20, - 80 psig.*

*The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reactor steam pressure is adequate to perform the test.

TABLE 3.8.4.3-1 (Continued)

MOTOR-OPERATED VALVES THERMAL OVERLOAD PROTECTION

VALVE NUMBER	SYSTEM(S) AFFECTED
E41-F022	HPCI
E41-F041	HPCI
E41-F042	HPCI
E41-F059	HPCI
E41-F075	HPCI
E41-F079	HPCI
E41-F600	HPCI
7. E51-F001	Reactor Core Isolation Cooling System (RCIC)
E51-F002	RCIC
E51-F007	RCIC
E51-F008	RCIC
E51-F010	RCIC
E51-F012	RCIC
E51-F013	RCIC
E51-F019	RCIC
E51-F022	RCIC
E51-F029	RCIC
E51-F031	RCIC
E51-F045	RCIC
E51-F046	RCIC
E51-F059	RCIC
E51-F062	RCIC
E51-F084	RCIC
E51-F095	RCIC
8. G1154-F018	Drywell Floor Drain System
G1154-F600	Drywell Floor Drain System
9. G33-F001	Reactor Water Clean-Up System (RWCU)
G33-F004	RWCU
10. G51-F600	Torus Water Management System (TWMS)
G51-F601	TWMS
G51-F602	TWMS
G51-F603	TWMS
G51-F604	TWMS
G51-F605	TWMS
G51-F606	TWMS
G51-F607	TWMS
11. N11-F607	Main Steam System
N11-F608	Main Steam System
N11-F609	Main Steam System
N11-F610	Main Steam System

3/4.1.5 STANDBY LIQUID CONTROL SYSTEM

The design objective of the Standby Liquid Control (SLC) System is two fold. One objective is to provide backup capability for bringing the reactor from full power to a cold, Xenon-free shutdown, assuming that the withdrawn control rods remain fixed in the rated power pattern. The second objective of the SLC System is to meet the requirement of the ATWS Rule, specifically 10 CFR 50.62 paragraph (c)(4) which states that, in part:

"Each boiling water reactor must have standby liquid control system (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution."

The SLC System uses enriched Boron-10 (contained in the Sodium pentaborate solution) to comply with 10 CFR 50.62 paragraph (c)(4). The methods used to determine compliance with the ATWS Rule are in accordance with Reference 2.

To meet both objectives, it is necessary to inject a minimum quantity of 2560 net gallons of 65 atom percent Boron-10 enriched sodium pentaborate in a solution having a concentration of no less than 9.0 weight percent (see Figure 3.1.5-1 for equivalent volumes and concentration ranges). The equivalent concentration of natural boron required to shutdown the reactor is 720 parts per million (ppm) in the 70°F moderator, including the Recirculation loops and with the RHR Shutdown Cooling Subsystems in operation. In addition to this, a 25 percent margin is provided to allow for leakage and imperfect mixing (900 ppm). The pumping rate of 41.2 gpm provides a negative reactivity insertion rate over the permissible sodium pentaborate solution volume range, which adequately compensates for the positive reactivity effects due to moderator temperature reduction and xenon decay during shutdown. The temperature requirement is necessary to ensure that the sodium pentaborate remains in solution.

With redundant pumps and explosive injection valves and with a highly reliable control rod scram system, operation of the reactor is permitted to continue for short periods of time with the system inoperable or for longer periods of time with one of the redundant components inoperable. The SLC tank heaters are only required when mixing sodium pentaborate and/or water to establish the required solution operating parameters during additions to the SLC tank. Normal operation of the SLCS does not depend on these tank heaters to maintain the solution above its saturation temperature. Technical requirements have been placed on the tank heater circuit breakers to ensure that their failure will not degrade other SLC components (see Specification 3/4.8.4.5).

Surveillance requirements are established on a frequency that assures a high reliability of the system. Once the solution is established, boron concentration will not vary unless more boron or water is added, thus a check on the temperature and volume once each 24 hours assures that the solution is available for use. Analysis of Boron-10 enrichment each 18 months provides sufficient assurance that the minimum enrichment of Boron-10 will be maintained.

3/4.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (Continued)

Power and flow dependent adjustments are provided in the COLR to assure that the fuel thermal-mechanical design criteria are preserved during abnormal transients initiated from off-rated conditions.

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

Details on how evaluations are performed, on the methods used, and how the MCPR limit is adjusted for operation at less than rated power and flow conditions are given in References 1 and 3 and the CORE OPERATING LIMITS REPORT.

At THERMAL POWER levels less than or equal to 25 percent of RATED THERMAL POWER, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. During initial startup testing of the plant, a MCPR evaluation will be made at 25 percent of RATED THERMAL POWER level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. The daily requirement for calculating MCPR when THERMAL POWER is greater than or equal to 25 percent of RATED THERMAL POWER is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when a limiting control rod pattern is approached ensures that MCPR will be known following a change in THERMAL POWER or power shape, regardless of magnitude, that could place operation at a thermal limit.

3.4.2.4 LINEAR HEAT GENERATION RATE

The thermal expansion rate of UO_2 pellets and Zircalloy cladding are different in that, during heatup, the fuel pellet could come into contact with the cladding and create stress. If the stress exceeds the yield stress of the cladding material, the cladding will crack. The LHGR limit assures that at any exposure, 1% plastic strain on the clad is not exceeded. This limit is a function of fuel type and is presented in the CORE OPERATING LIMITS REPORT.

References:

1. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (the approved version at the time the reload analyses are performed shall be identified in the COLR).

2. "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident - SAFER/GESTR Application Methodology", NEDE 23785-1-PA (the approved version at the time the reload analyses are performed shall be identified in the COLR).

3. "Fermi 2 Maximum Extended Operating Domain Analysis", NEDC-31843P, July 1990.

3/4.4.1 RECIRCULATION SYSTEM

The impact of single recirculation loop operation upon plant safety is assessed and shows that single-loop operation is permitted at power level is up to 67.2% of RATED THERMAL POWER if the MCPR fuel cladding safety limit is increased as noted by Specification 2.1.2. APRM scram and control rod block setpoints (or APRM gains) are adjusted as noted in Tables 2.2.1-1 and 3.3.6-2, respectively. A time period of 4 hours is allowed to make these adjustments following the establishment of single loop operation since the need for single loop operation often cannot be anticipated. MCPR operating limits adjustments in Specification 3.2.3 for different plant operating situations are applicable to both single and two recirculation loop operation.

To prevent potential control system oscillations from occurring in the recirculation flow control system, the operating mode of the recirculation flow control system must be restricted to the manual control mode for single-loop operation.

Additionally, surveillance on the pump speed of operating recirculation loop is imposed to exclude the possibility of excessive core internals vibration. The surveillance on differential temperatures below 30% THERMAL POWER or 50% rated recirculation loop flow is to prevent undue thermal stress on vessel nozzles, recirculation pump and vessel bottom head during a power or flow increase following extended operation in the single recirculation loop mode.

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

Recirculation pump speed mismatch limits are in compliance with the ECCS LOCA analysis design criteria for two recirculation loop operation. The limits will ensure an adequate core flow coastdown from either recirculation loop following a LOCA.

In the case where the mismatch limits cannot be maintained during two loop operation, continued operation is permitted in a single recirculation loop mode.

In order to prevent undue stress on the vessel nozzles and bottom head region, the recirculation loop temperatures shall be within 50°F of each other prior to startup of an idle loop. The loop temperature must also be within 50°F of the reactor pressure vessel coolant temperature to prevent thermal shock to the recirculation pump and recirculation nozzles.

3/4.4.1 RECIRCULATION SYSTEM (Continued)

Sudden equalization of a temperature difference greater than 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

Requirements are imposed to prohibit idle loop startup above the 77% rod line to minimize the potential for initiating core thermal-hydraulic instability.

3/4.4.2 SAFETY/RELIEF VALVES

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

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

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

3/4.4.10 CORE THERMAL HYDRAULIC STABILITY

BWR cores typically operate with the presence of global flux noise in a stable mode which is due to random boiling and flow noise. As the power/flow conditions are changed, along with other system parameters (pressure, subcooling, power distribution, etc.) the thermal hydraulic/reactor kinetic feedback mechanism can be enhanced such that random perturbations may result in sustained limit cycle or divergent oscillations in power and flow.

Two major modes of oscillations have been observed in BWRs. The first mode is the fundamental or core-wide oscillation mode in which the entire core oscillates in phase in a given axial plane. The second mode involves regional oscillation in which one half of the core oscillates 180 degrees out of phase with the other half. Studies have indicated that adequate margin to the Safety Limit Minimum Critical Power Ratio (SLMCPR) may not exist during regional oscillations.

Region A and B of Figure 3.4.10-1 represent the least stable conditions of the plant (high power/low flow). Region A and B are usually entered as the result of a plant transient (for example, recirculation pump trips) and therefore are generally not considered part of the normal operating domain. Since all stability events (including test experience) have occurred in either Region A or B, these regions are avoided to minimize the possibility of encountering oscillations and potentially challenging the SLMCPR. Therefore, intentional operation in Regions A or B is not allowed. It is recognized that during certain abnormal conditions within the plant, it may become necessary to enter Region A or B for the purpose of protecting equipment which, were it to fail, could impact plant safety or for the purpose of protecting a safety or fuel operating limit. In these cases, the appropriate actions for the region entered would be performed as required.

Most oscillations that have occurred during testing and operation have occurred at or above the 96% rod line with core flow near natural circulation. This behavior is consistent with analysis which predict reduced stability margin with increasing power or decreasing flow. As core flow is increased or power decreased, the probability of oscillations occurring will decrease. Region A of Figure 3.4.10-1 bounds the majority of the stability events and tests observed in GE BWRs. Since Region A represents the least stable region of the power/flow operating domain, the potential to rapidly encounter large magnitude core thermal hydraulic oscillations is increased. During transients, the operator may not have sufficient time to manually insert control rods to mitigate the oscillations before they reach an unacceptable magnitude. Therefore, the prompt action of manually scramming the plant when Region A is entered is required to ensure protection of the SLMCPR.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the safety analyses. This restriction, in conjunction with the leakage rate limitation, will limit the SITE BOUNDARY radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

PRIMARY CONTAINMENT INTEGRITY is demonstrated by leak rate testing and by verifying that all primary containment penetrations not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by locked valves, blank flanges or deactivated automatic valves secured in the closed position. For test, vent and drain connections which are part of the containment boundary, a threaded pipe cap with acceptable sealant in addition to the containment isolation valve(s) provides protection equivalent to a blank flange.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on primary containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the safety analyses at the peak accident pressure of 56.5 psig, P_a. Updated analysis demonstrates maximum expected pressure is less than 56.5 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J of 10 CFR Part 50 with the exception of exemptions granted for main steam isolation valve leak testing and testing the airlocks after each opening and analyzing the Type A test data.

Appendix J to 10 CFR Part 50, Paragraph III.A.3, requires that all Type A tests be conducted in accordance with the provisions of N45.4-1972, "Leakage-Rate Testing of Containment Structures for Nuclear Reactors." N45.4-1972 requires that Type A test data be analyzed using point-to-point or total time analytical techniques. Specification 4.6.1.2a. requires use of the mass plot analytical technique. The mass plot method is considered the better analytical technique, since it yields a confidence interval which is a small fraction of the calculated leak rate; and the interval decreases as more data sets are added to the calculation. The total time and point-to-point techniques may give confidence intervals, which are large fractions of the calculated leak rate, and the intervals may increase as more data sets are added.

PRIMARY CONTAINMENT AIR LOCKS (Continued)

3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steamline isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR Part 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIVs such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIVs when isolation of the primary system and containment is required.

3/4.6.1.5 PRIMARY CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 56.5 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 DRYWELL AND SUPPRESSION CHAMBER INTERNAL PRESSURE

The limitations on drywell and suppression chamber internal pressure ensure that the containment peak pressure of less than 56.5 psig does not exceed the maximum allowable pressure of 62 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of 2 psid.

3/4.6.1.7 DRYWELL AVERAGE AIR TEMPERATURE

The limitation on drywell average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 340°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM

The drywell and suppression chamber purge supply and exhaust isolation valves are maintained closed during a majority of the plant operating time. Maintaining these valves closed (even though they have been qualified to close against the buildup of pressure in primary containment in the event of DBA/LOCA) reduces the potential for release of excessive quantities of radioactive material.

DRYWELL AND SUPPRESSION CHAMBER PURGE SYSTEM (Continued)

Purging or venting through the Standby Gas Treatment System (SGTS) imposes a vulnerability factor on the integrity of the SGTS. Should a LOCA occur while the purge pathway is through the SGTS the associated pressure surge, before the purge valves close, may adversely affect the integrity of the SGTS charcoal filters. Therefore, PURGING or VENTING through the SGTS is limited to 90 hours per 365 days. This time limit is not imposed when venting through the SGTS with the 1-inch valves or when PURGING or VENTING through the Reactor Building Ventilation System with any of the purge valves.

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

The 6, 10, 20, and 24 inch purge valves are generally configured in a three (3) valve arrangement at each of the associated purge penetrations. The valves are leak tested by pressurizing between the three valves and a total leakage is determined as opposed to a single valve leakage. Verifying that the measured leakage rate is less than 0.5 L_a for this multi-valve arrangement is more conservative than a limit of 0.5 L_a for a single valve.

3/4.6.2 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the primary containment pressure will not exceed the maximum allowable pressure of 62 psig during primary system blowdown from full operating pressure.

The suppression chamber water provides the heat sink for the reactor coolant system energy release following a postulated rupture of the system. The suppression chamber water volume must absorb the associated decay and structural sensible heat released during reactor coolant system blowdown from 1045 psig. Since all of the gases in the drywell are purged into the suppression chamber air space during a loss-of-coolant accident, the pressure of the liquid must not exceed 62 psig, the suppression chamber maximum pressure. The design volume of the suppression chamber, water and air, was obtained by considering that the total volume of reactor coolant to be condensed is discharged to the suppression chamber and that the drywell volume is purged to the suppression chamber.

DEPRESSURIZATION SYSTEMS (Continued)

Using the minimum or maximum water volumes given in this specification, containment pressure during the design basis accident is less than 56.5 psig which is below the maximum allowable pressure of 62 psig. Maximum water volume of 124,220 ft^3 results in a downcomer submergence of 3'4" and the minimum volume of 121,080 ft^3 results in a submergence of 3'0". The maximum temperature at the end of the blowdown tested during the Humboldt Bay and Bodega Bay tests was 170°F.

Should it be necessary to make the suppression chamber inoperable, this shall only be done as specified in Specification 3.5.3.

Under full power operation conditions, a design basis accident blowdown from an initial suppression chamber water temperature of 95°F results in a water temperature of approximately 135°F in the short term following the blowdown. At this temperature and atmospheric pressure, the available NPSH exceeds that required by both the RHR and core spray pumps, thus there is no dependency on containment overpressure during the accident injection phase. If both RHR loops are used for containment cooling, there is no dependence on containment overpressure for post-LOCA operations.

The large thermal capacitance of the suppression pool is also utilized during plant transients requiring safety/relief valve (SRV) actuation. Steam is discharged from the main steam lines through the SRVs and their accompanying discharge lines into the suppression pool where it is condensed, resulting in an increase in the temperature of the suppression pool water. Although stable steam condensation is expected at all pool temperatures, NUREG-0783 imposes a local temperature limit shown in Figure B 3/4.6.2-1 in the vicinity of the T-type quencher discharge device. The limiting plant transients with respect to heat input to the suppression pool have been analyzed. The conservative analysis showed that limiting the average water temperature to less than or equal to 170°F will result in local pool temperatures below the condensation stability limit of Bases Figure B 3/4.6.2-1.

Experimental data indicate that excessive steam condensing loads can be avoided if the peak local temperature of the suppression pool is maintained below 200°F during any period of relief valve operation. Specifications have been placed on the envelope of reactor operating conditions so that the reactor can be depressurized in a timely manner to avoid the regime of potentially high suppression chamber loadings.

Because of the large volume and thermal capacity of the suppression pool, the volume and temperature normally change very slowly and monitoring these parameters daily is sufficient to establish any temperature trends. By requiring the suppression pool temperature to be frequently recorded during periods of significant heat addition, the temperature trends will be closely followed so that appropriate action can be taken. The requirement for an external visual

3/4.7.9 MAIN TURBINE BYPASS SYSTEM AND MOISTURE SEPARATOR REHEATER

The main turbine bypass system is an active bypass system designed to open the bypass valves in the event of a turbine trip to decrease the severity of the pressure transient. Each valve is sized to pass approximately 12½ percent reactor steam flow in the full-open position for a controlled total bypass of approximately 25 percent reactor steam flow. The main turbine bypass system is required to be OPERABLE consistent with the assumptions of the Feedwater Controller Failure analysis.

The primary purpose of the moisture separator reheater is to improve cycle efficiency by using primary system steam to heat the high pressure turbine exhaust before it enters the low-pressure turbines. In doing so, it also provides a passive steam bypass flow of about 10 percent that mitigates the early effects of over-pressure transients. The moisture separator reheater is required to be OPERABLE consistent with the assumptions of the Main Turbine Trip with Turbine Bypass Failure analysis and the Feedwater Controller Failure analysis.

The operation with one or both of the main turbine bypasses inoperable or the moisture separator reheater inoperable to perform preventive or corrective maintenance above 25 percent RATED THERMAL POWER, requires, after one hour, the evaluation of the MCPR in accordance with Specification 3.2.3. If the MCPR is within the bounds established by Specification 3.2.3, power increases to or operation above 25 percent RATED THERMAL POWER is allowed.

3/4.7.11 APPENDIX R ALTERNATIVE SHUTDOWN AUXILIARY SYSTEMS

The systems identified in this section are those utilized for Appendix R Alternative shutdown but not included in other sections of the Technical Specifications. The ACTION statements assure that the auxiliary systems will be OPERABLE or that acceptable alternative means are established to achieve the same objective.

There are four independent Combustion Turbine-Generator units onsite. CTG 11 Unit 1 has a diesel engine starter and thus can be started independently from offsite power. CTG 11 Units 2, 3, and 4 have AC-motor starters and rely on a 480-volt AC feed. The phrase "alternative source of power", as used in Specification 3.7.11, ACTION b.2, is defined as a source of power that is not reliant on offsite power for starting (if required) or operating (if already running) and capable of supplying the required loads on the 4160-volt busses associated with the Alternative Shutdown System.

One of the two installed Standby Feedwater Pumps and one of the two listed Drywell Cooling Units are necessary for Appendix R Alternative shutdown. Therefore unlimited operation with one of the two components inoperable is justified provided increased surveillance is performed on the components which remain OPERABLE.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Regional Administrator of the Regional Office of the NRC within the time period specified for each report.

CORE OPERATING LIMITS REPORT

6.9.3 Selected cycle specific core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT (COLR) before each reload cycle or any remaining part of a reload cycle. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC in General Electric Company reports NEDE-24011-P-A and NEDE-23785-1-PA. The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, and transient and accident analysis limits) of the safety analysis are met. The COLR, including any mid-cycle revisions or supplement thereto, shall be submitted upon issuance to the NRC Document Control Desk, with copies to the Regional Administrator and Resident Inspector prior to use.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering time interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

c. ALL REPORTABLE EVENTS.

d. Records of surveillance activities, inspections, and calibrations required by these Technical Specifications.

e. Records of changes made to the procedures required by Specification 6.8.1.

f. Records of radioactive shipments.

g. Records of sealed source and fission detector leak tests and results.

h. Records of annual physical inventory of all sealed source material of record.

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 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 I, 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 3 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 operation 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 1351 °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 + 53.4%)

(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 of licensee-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 Nonsafety-Related Loads

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 power 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 operational 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.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

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.

Principal Contributors: T. Chandrasekaran
T. Colburn
K. Eccleston
R. Frahm
R. Goel
G. Hubbard
J. Kudrick
R. Scholl
R. Stransky
C. Wu

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.