



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

November 21, 1989

Docket No. 50-341

Mr. B. Ralph Sylvia
Senior Vice President - Nuclear
Operations
Detroit Edison Company
6400 North Dixie Highway
Newport, Michigan 48166

Dear Mr. Sylvia:

SUBJECT: AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-43:
(TAC NO. 75279)

The Commission has issued the enclosed Amendment No. 44 to Facility Operating License No. NPF-43 for the Fermi-2 facility. This amendment consists of changes to the plant Technical Specifications in response to your letter dated November 16, 1989.

The amendment revises the TS Figure 3.2.3-2, Flow Correction (K_f) Factor. The figure was part of a previous license amendment request dated April 3, 1989, and issued by the NRC as Amendment No. 42 to the Fermi-2 Operating License and was recently found to be in error. Further review of other figures issued in Amendment 42 determined that five other figures submitted should be modified to more clearly show the limits they impose upon plant operation. In addition, some administrative or typographical errors were found in the designation of the fuel bundle types in Specification 3.2.4 and in the referencing of new figures in Specification 4.2.3.1. Minor corrections to the Bases were also found to be necessary.

Your November 16, 1989, letter requested that this amendment be processed under the provisions of 10 CFR 50.91(a)(5) as an emergency situation. The bases for the emergency situation is that reactor startup is scheduled for November 20, 1989, and lack of timely action would unnecessarily prevent resumption of plant operation. The staff has reviewed the bases for the emergency circumstances and concurs that the proposed amendment does fall under the provisions of 10 CFR 50.91(a)(5).

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A copy of the Safety Evaluation supporting this amendment is also enclosed. The Notice of Issuance and final determination of no significant hazards consideration and opportunity for a hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/s/

John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 44 to NPF-43
2. Safety Evaluation

cc w/enclosures:
See next page

DISTRIBUTION

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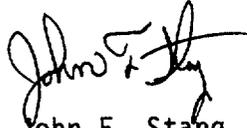
FERMI EMERG T/S AMEND

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|---|------------------------------------|-------------------------------------|-----------------------------------|--------------------------------|----------------------------|
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OFFICIAL RECORD COPY

A copy of the Safety Evaluation supporting this amendment is also enclosed. The Notice of Issuance and final determination of no significant hazards consideration and opportunity for a hearing will be included in the Commission's biweekly Federal Register notice.

Sincerely,



John F. Stang, Project Manager
Project Directorate III-1
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Enclosures:

1. Amendment No. 44 to NPF-43
2. Safety Evaluation

cc w/enclosures:

See next page

Mr. B. Ralph Sylvia
Detroit Edison Company

Fermi-2 Facility

cc:

Mr. Ronald C. Callen
Adv. Planning Review Section
Michigan Public Service Commission
6545 Mercantile Way
P. O. Box 30221
Lansing, Michigan 48909

Ms. Lynn Goodman
Supervisor - Licensing
Detroit Edison Company
Fermi Unit 2
6400 North Dixie Highway
Newport, Michigan 48166

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

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
P. O. Box 30035
Lansing, Michigan 48909

Mr. Walt Rogers
U.S. Nuclear Regulatory Commission
Resident Inspector's Office
6450 W. Dixie Highway
Newport, Michigan 48166

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

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



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

DETROIT EDISON COMPANY

WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

DOCKET NO. 50-341

FERMI-2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
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 November 16, 1989, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;
 - B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;
 - C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;
 - D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and
 - E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.
2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. NPF-43 is hereby amended to read as follows:

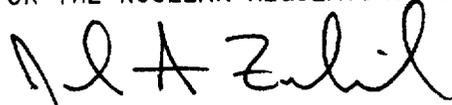
Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A, as revised through Amendment No. 44, and the Environmental Protection Plan contained in Appendix B, are hereby incorporated in the license. DECo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

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

FOR THE NUCLEAR REGULATORY COMMISSION

A handwritten signature in black ink, appearing to read "John Zwolinski". The signature is written in a cursive style with a large initial "J" and "Z".

John Zwolinski, Assistant Director
for Region III
Division of Reactor Projects - III,
IV, V & Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 21, 1989

ATTACHMENT TO LICENSE AMENDMENT NO. 44

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. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE

INSERT

B 2-1

B 2-1

3/4 2-4

3/4 2-4

3/4 2-4a

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3/4 2-8a

3/4 2-8a

3/4 2-8b

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3/4 2-10

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B 3/4 1-1*

B 3/4 1-1*

B 3/4 1-2

B 3/4 1-2

B 3/4 2-3

B 3/4 2-3

B 3/4 2-4

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B 3/4 2-4b

B 3/4 2-4b

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

2.1 SAFETY LIMITS

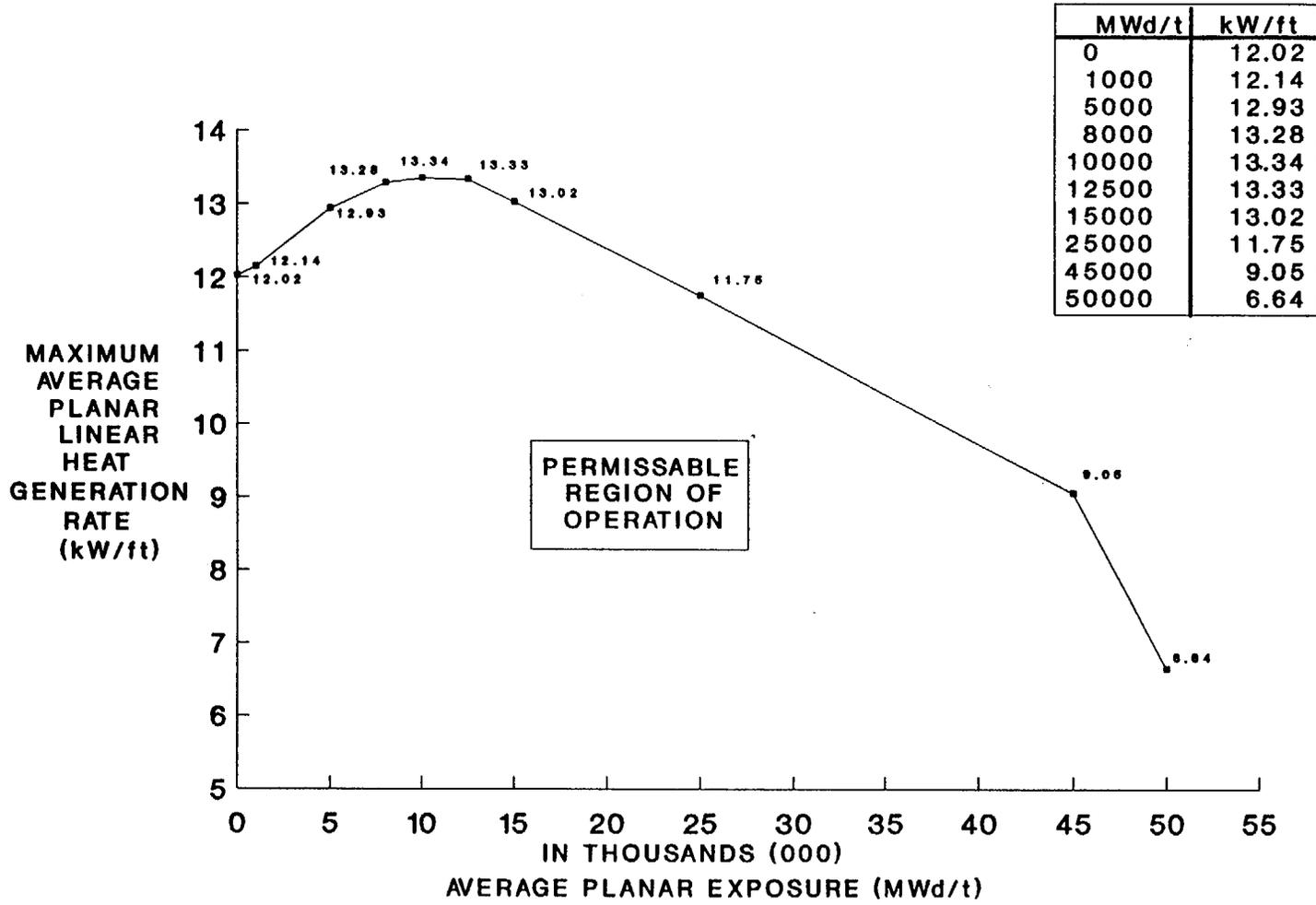
BASES

2.0 INTRODUCTION

The fuel cladding, reactor pressure vessel and primary system piping are the principal barriers to the release of radioactive materials to the environs. Safety Limits are established to protect the integrity of these barriers during normal plant operations and anticipated transients. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Because fuel damage is not directly observable, a step-back approach is used to establish a Safety Limit. MCPR greater than the Safety Limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity. The fuel cladding is one of the physical barriers which separate the radioactive materials from the environs. The integrity of this cladding barrier is related to its relative freedom from perforations or cracking. Although some corrosion or use related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the Limiting Safety System Settings. While fission product migration from cladding perforation is just as measurable as that from use related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding Safety Limit is defined with a margin to the conditions which would produce onset of transition boiling, MCPR of 1.0. These conditions represent a significant departure from the condition intended by design for planned operation.

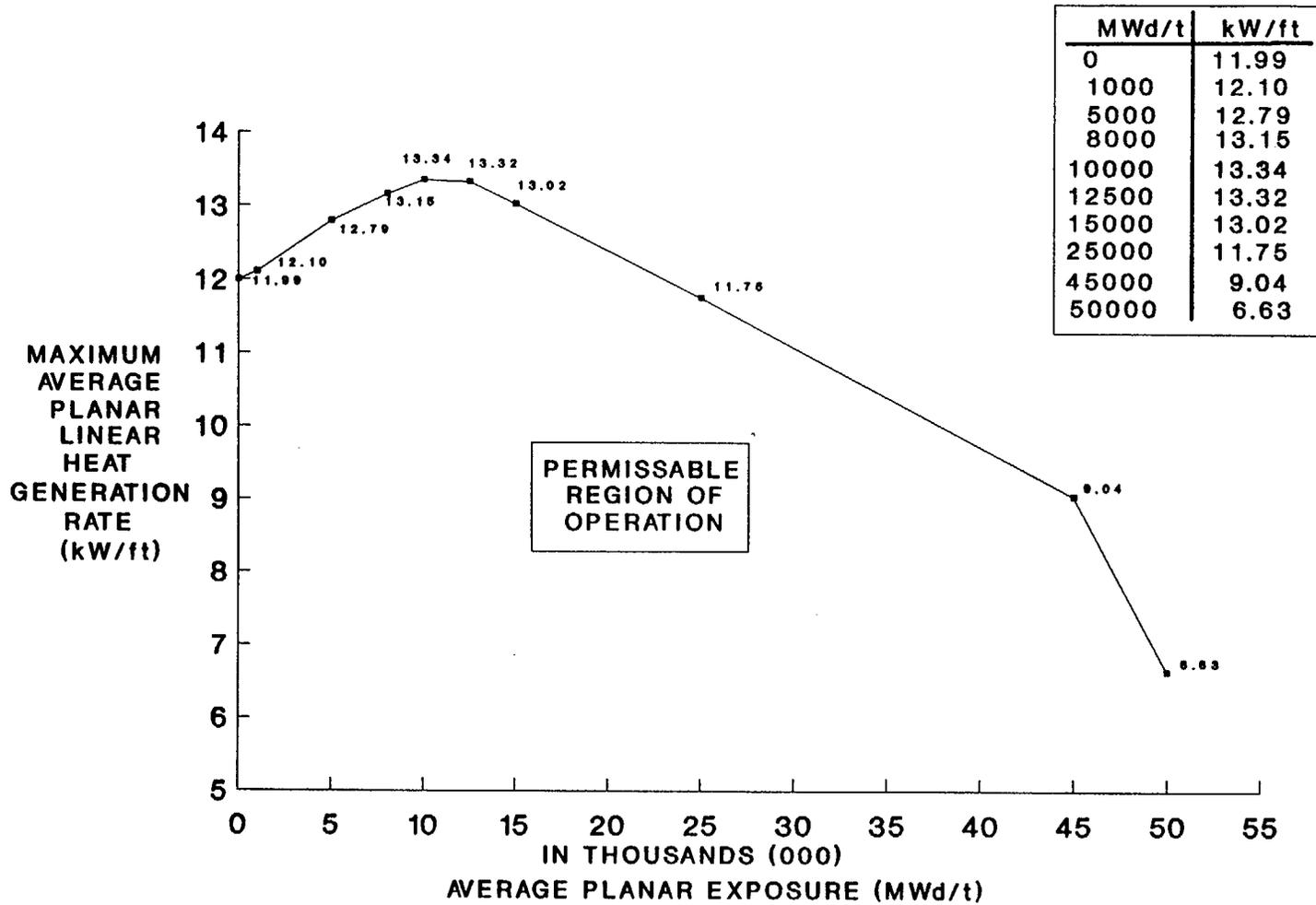
2.1.1 THERMAL POWER, Low Pressure or Low Flow

The use of the approval critical power correlation is not valid for all critical power calculations at pressures below 785 psig or core flows less than 10% of rated flow. Therefore, the fuel cladding integrity Safety Limit is established by other means. This is done by establishing a limiting condition on core THERMAL POWER with the following basis. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28×10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28×10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE RELOAD FUEL TYPE BC318D

FIGURE 3.2.1-3



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VERSUS AVERAGE PLANAR EXPOSURE RELOAD FUEL TYPE BC318E

FIGURE 3.2.1-4

POWER DISTRIBUTION LIMITS

SURVEILLANCE REQUIREMENTS

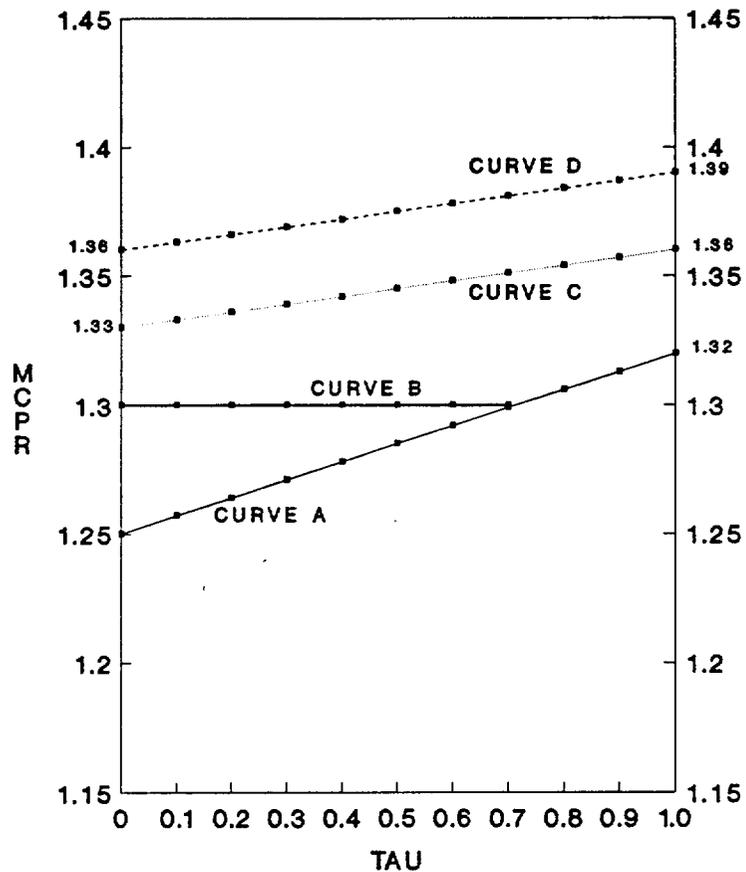
4.2.3.1 MCPR, with:

- a. $t = 1.0$ prior to performance of the initial scram time measurements for the cycle in accordance with Specification 4.1.3.2, or
- b. t as defined in Specification 3.2.3 used to determine the limit within 72 hours of the conclusion of each scram time surveillance test required by Specification 4.1.3.2,

shall be determined to be equal to or greater than the applicable MCPR limit determined from Figures 3.2.3-1 through 3.2.3-1B and 3.2.3-2:

- 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 MCPR.
- d. The provisions of Specification 4.0.4 are not applicable.

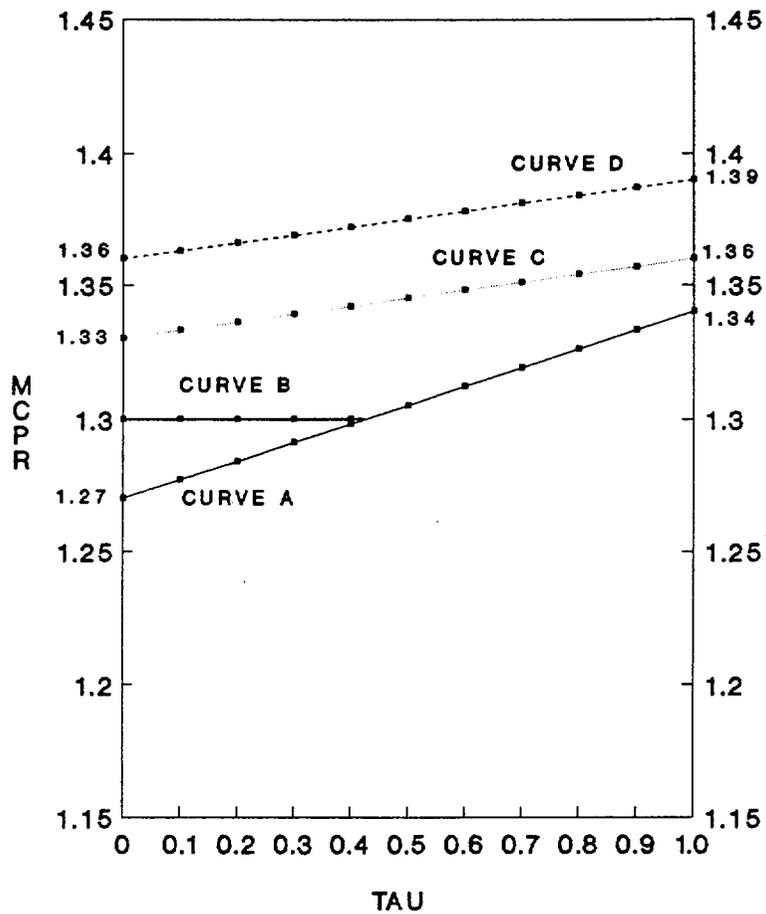
4.2.3.2 Prior to the use of Curve A and whenever Surveillance Requirement 4.2.3.1 is performed while using Curve A of Figures 3.2.3-1 through 3.2.3-1B, verify that all non-CCC control rods are fully withdrawn from the core. Non-CCC control rods are all control rods excluding A2 rods, A1 shallow rods inserted less than or equal to notch position 36, all peripheral rods, and rods inserted to position 46. Normal control rod operability checks, coupling checks, scram time testing, and friction testing of non-CCC control rods does not require the utilization of the more restrictive non-CCC operational mode MCPR limits.



- CURVE A** - MCPR limit for CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE B** - MCPR limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE C** - MCPR limit for both CCC or non-CCC operational modes with either turbine bypass or moisture separator reheater out of service.
- CURVE D** - MCPR limit for both CCC and non-CCC operational modes with both turbine bypass and moisture separator reheater out of service.

**BOC TO 12,700 MWD/ST
MINIMUM CRITICAL POWER RATIO
(MCPR) VERSUS TAU AT RATED FLOW**

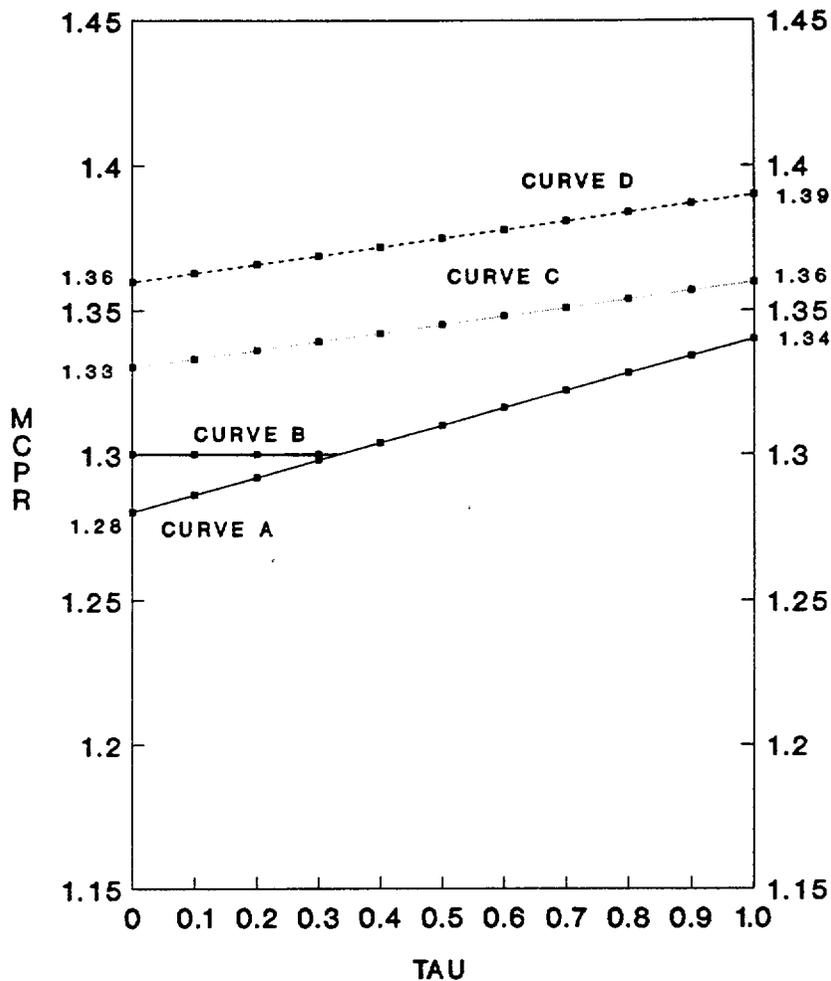
FIGURE 3.2.3-1



- CURVE A** - MCPR limit for CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE B** - MCPR limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE C** - MCPR limit for both CCC or non-CCC operational modes with either turbine bypass or moisture separator reheater out of service.
- CURVE D** - MCPR limit for both CCC or non-CCC operational modes with both turbine bypass and moisture separator reheater out of service.

12,700 MWD/ST TO 13,700 MWD/ST
 MINIMUM CRITICAL POWER RATIO
 (MCPR) VERSUS TAU AT RATED FLOW

FIGURE 3.2.3-1A



- CURVE A - MCP R limit for CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE B - MCP R limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.
- CURVE C - MCP R limit for both CCC and non-CCC operational modes with either turbine bypass or moisture separator reheater out of service.
- CURVE D - MCP R limit for both CCC or non-CCC operational modes with both turbine bypass and moisture separator reheater out of service.

13,700 MWD/ST TO EOC
 MINIMUM CRITICAL POWER RATIO
 VERSUS TAU AT RATED FLOW

FIGURE 3.2.3-1B

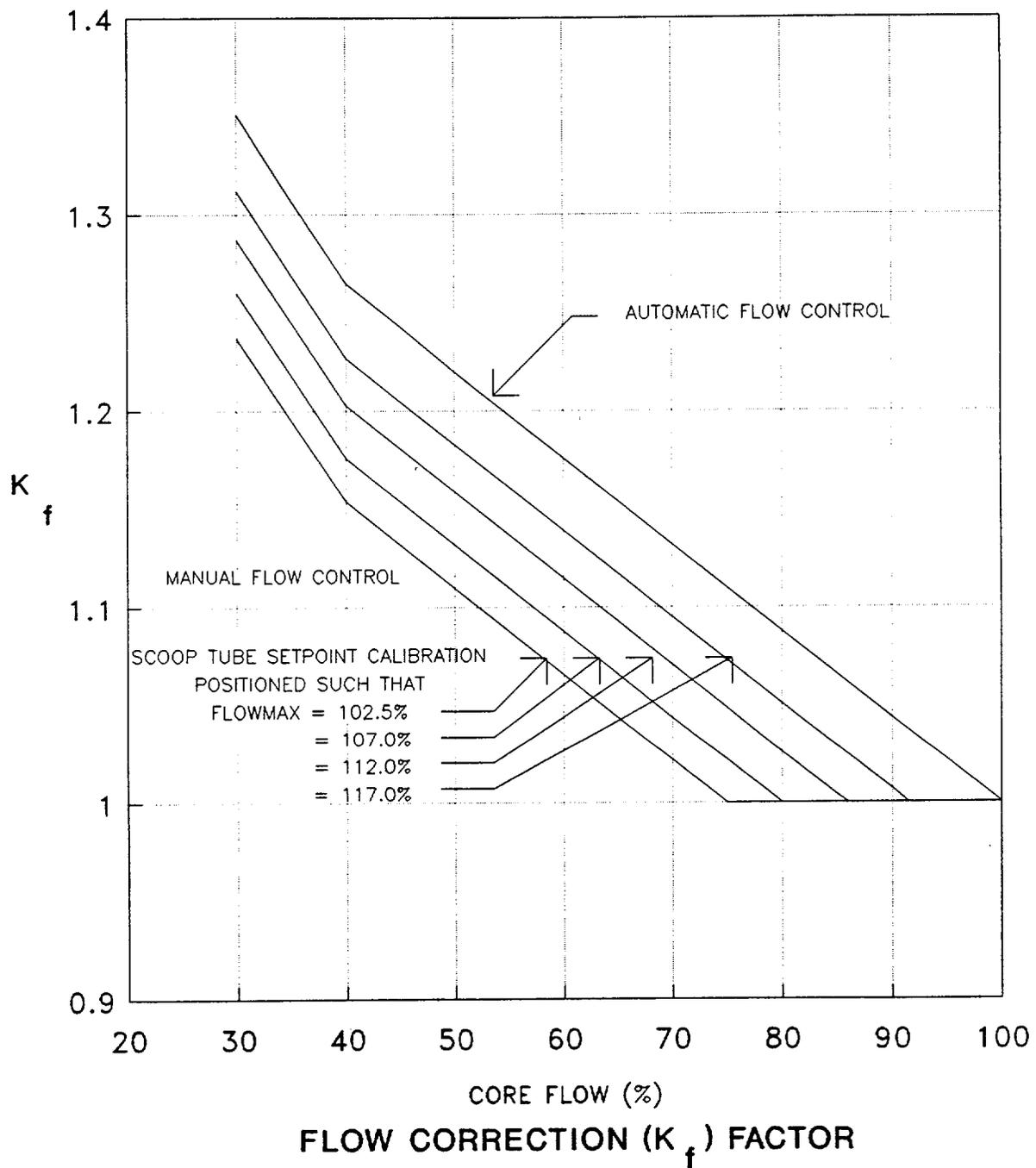


FIGURE 3.2.3-2

POWER DISTRIBUTION LIMITS

3/4.2.4 LINEAR HEAT GENERATION RATE

LIMITING CONDITION FOR OPERATION

3.2.4 The LINEAR HEAT GENERATION RATE (LHGR) shall not exceed 13.4 kw/ft for bundle types 8CR183 and 8CR233 or 14.4 kw/ft for bundle types BC318D and BC318E.

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

ACTION:

With the LHGR of any fuel rod exceeding the limit, initiate corrective action within 15 minutes and restore the LHGR to within the limit within 2 hours or reduce THERMAL POWER to less than 25% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.4 LHGR's shall be determined to be equal to or less than the limit:

- 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 on a LIMITING CONTROL ROD PATTERN FOR LHGR.
- d. The provisions of Specification 4.0.4 are not applicable.

3/4.1 REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.1 SHUTDOWN MARGIN

A sufficient SHUTDOWN MARGIN ensures that 1) the reactor can be made subcritical from all operating conditions, 2) the reactivity transients associated with postulated accident conditions are controllable within acceptable limits, and 3) the reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

Since core reactivity values will vary through core life as a function of fuel depletion and poison burnup, the demonstration of SHUTDOWN MARGIN will be performed in the cold, xenon-free condition and shall show the core to be subcritical by at least $R + 0.38\% \Delta k/k$ or $R + 0.28\% \Delta k/k$, as appropriate. The value of R in units of $\% \Delta k/k$ is the difference between the calculated value of maximum core reactivity during the operating cycle and the calculated beginning-of-life core reactivity. The value of R must be positive or zero and must be determined for each fuel loading cycle.

Two different values are supplied in the Limiting Condition for Operation to provide for the different methods of demonstration of the SHUTDOWN MARGIN. The highest worth rod may be determined analytically or by test. The SHUTDOWN MARGIN is demonstrated by an insequence control rod withdrawal at the beginning of life fuel cycle conditions, and, if necessary, at any future time in the cycle if the first demonstration indicates that the required margin could be reduced as a function of exposure. Observation of subcriticality in this condition assures subcriticality with the most reactive control rod fully withdrawn.

This reactivity characteristic has been a basic assumption in the analysis of plant performance and can be best demonstrated at the time of fuel loading, but the margin must also be determined anytime a control rod is incapable of insertion.

3/4.1.2 REACTIVITY ANOMALIES

Since the SHUTDOWN MARGIN requirement for the reactor is small, a careful check on actual conditions to the predicted conditions is necessary, and the changes in reactivity can be inferred from these comparisons of rod patterns. Since the comparisons are easily done, frequent checks are not an imposition on normal operations. A 1% change is larger than is expected for normal operation so a change of this magnitude should be thoroughly evaluated. A change as large as 1% would not exceed the design conditions of the reactor and is on the safe side of the postulated transients.

REACTIVITY CONTROL SYSTEMS

BASES

3/4.1.3 CONTROL RODS

The specifications of this section ensure that (1) the minimum SHUTDOWN MARGIN is maintained, (2) the control rod insertion times are consistent with those used in the safety analyses, and (3) limit the potential effects of the rod drop accident. The ACTION statements permit variations from the basic requirements but at the same time impose more restrictive criteria for continued operation. A limitation on inoperable rods is set such that the resultant effect on total rod worth and scram shape will be kept to a minimum. The requirements for the various scram time measurements ensure that any indication of systematic problems with rod drives will be investigated on a timely basis.

Damage within the control rod drive mechanism could be a generic problem, therefore with a control rod immovable because of excessive friction or mechanical interference, operation of the reactor is limited to a time period which is reasonable to determine the cause of the inoperability and at the same time prevent operation with a large number of inoperable control rods.

Control rods that are inoperable for other reasons are permitted to be taken out of service provided that those in the nonfully inserted position are consistent with the SHUTDOWN MARGIN requirements.

The number of control rods permitted to be inoperable could be more than the eight allowed by the specification, but the occurrence of eight inoperable rods could be indicative of a generic problem and the reactor must be shut down for investigation and resolution of the problem.

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent the MCPR from becoming less than the Safety Limit MCPR during the limiting power transient analyzed in Chapter 15 of the UFSAR. This analysis shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the specifications, provide the required protection and MCPR remains greater than the Safety Limit MCPR. The occurrence of scram times longer than those specified should be viewed as an indication of a systematic problem with the rod drives and therefore the surveillance interval is reduced in order to prevent operation of the reactor for long periods of time with a potentially serious problem.

The scram discharge volume is required to be OPERABLE so that it will be available when needed to accept discharge water from the control rods during a reactor scram and will isolate the reactor coolant system from the containment when required.

Control rods with inoperable accumulators are declared inoperable and Specification 3.1.3.1 then applies. This prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed even though control rods with inoperable accumulators may still be inserted with normal drive water pressure. Operability of the accumulator ensures that there is a means available to insert the control rods even under the most unfavorable depressurization of the reactor.

BASES TABLE B 3.2.1-1
SIGNIFICANT INPUT PARAMETERS TO THE
LOSS-OF-COOLANT ACCIDENT ANALYSIS

Plant Parameters:

Core THERMAL POWER..... 3430 MWt* which corresponds to
105% of rated steam flow

Vessel Steam Output..... 14.86×10^6 lbm/hr which
corresponds to 105% of rated
steam flow

Vessel Steam Dome Pressure..... 1055 psia

Design Basis Recirculation Line
Break Area for:

a. Large Breaks 4.1 ft^2

b. Small Breaks 0.1 ft^2

Fuel Parameters:

| FUEL TYPE | FUEL BUNDLE GEOMETRY | PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kW/ft) | DESIGN AXIAL PEAKING FACTOR | INITIAL MINIMUM CRITICAL POWER RATIO |
|--------------|-------------------------|--|--------------------------------------|--|
| Initial Core | 8 x 8 | 13.4 | 1.4 | 1.18 |
| First Reload | 8 x 8 | 14.4 | 1.4 | 1.18 |

A more detailed listing of input of each model and its source is presented in Section II of Reference 1 and subsection 6.3 of the FSAR.

*This power level meets the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its Technical Specification LINEAR HEAT GENERATION RATE limit.

POWER DISTRIBUTION LIMITS

BASES

3/4.2.3 MINIMUM CRITICAL POWER RATIO

The required operating limiting MCPRs at steady-state operating conditions as specified in Specification 3.2.3 are derived from the established fuel cladding integrity Safety Limit MCPR, and an analysis of abnormal operational transients. For any abnormal operating transients analysis evaluation with the initial condition of the reactor being at the steady state operating limit, it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip setting given in Specification 2.2.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in CRITICAL POWER RATIO (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease. The limiting transient yields the largest delta MCPR. When added to the Safety Limit MCPR, the required minimum operating limiting MCPR of Specification 3.2.3 is obtained and presented in Figures 3.2.3-1, 3.2.3-1A, and 3.2.3-1B.

The MCPR curves illustrated in Figures 3.2.3-1 thru 3.2.3-1B were derived as described above for the following assumed operating conditions:

- Curve A - MCPR limit with turbine bypass system, moisture separator reheater systems in service and CCC (Control Cell Core) operational mode (A2 rods, A1 shallows inserted less than or equal to notch position 36, all peripheral rods, and all rods inserted to position 46). The operating domain includes the 100% power/flow region and extended load line region with 100% power and reduced flow.
- Curve B - MCPR limit with the turbine bypass system, moisture separator reheater systems in service and non-CCC operational mode (any non-CCC control rod inserted in the core). The operating domain includes the 100% power/flow region and the extended load line region with 100% power and reduced flow.
- Curve C - MCPR limit for either CCC or non-CCC operational modes with either the main turbine bypass system inoperative and the moisture separator reheater system available or the main turbine bypass system available and the moisture separator reheater system inoperative. The operating domain includes the 100% power/flow region and the extended load line region with 100% power with reduced flow.
- Curve D - MCPR limit for either CCC or non-CCC operational modes with the main turbine bypass system inoperative and the moisture separator reheater system inoperative. The operating domain includes the

POWER DISTRIBUTION LIMITS

BASES

3/4.2.3 MINIMUM CRITICAL POWER RATIO (Continued)

bypass system or the moisture separator reheater be inoperable as 25-percent RATED THERMAL POWER is exceeded, the MCPR check must be completed within one hour.

The evaluation of a given transient begins with the system initial parameters shown in UFSAR Table 15.0.1 that are input to a GE-core dynamic behavior transient computer program. The codes used to evaluate transients are described in GESTAR II. The principal result of this evaluation is the reduction in MCPR caused by the transient.

The purpose of the K_f factor of Figure 3.2.3-2 is to define operating limits at other than rated core flow conditions. At less than 100% of rated flow the required MCPR is the product of the MCPR and the K_f factor. The K_f factors assure that the Safety Limit MCPR will not be violated during a flow increase transient resulting from a motor-generator speed control failure. The K_f factors may be applied to both manual and automatic flow control modes.

The K_f factor values shown in Figure 3.2.3-2 were developed generically and are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to RATED THERMAL POWER at rated core flow, although they are applicable for the extended operating region.

For the manual flow control mode, the K_f factors were calculated such that for the maximum flow rate, as limited by the pump scoop tube setpoint and the corresponding THERMAL POWER along the rated flow control line, the limiting bundle's relative power was adjusted until the MCPR changes with different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR, determines the K_f .



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
RELATED TO AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-43

DETROIT EDISON COMPANY

WOLVERINE POWER SUPPLY COOPERATIVE, INCORPORATED

FERMI-2

DOCKET NO. 50-341

1.0 INTRODUCTION

By letter dated November 16, 1989, the Detroit Edison Company (DECo or the licensee) requested amendment to the Technical Specifications (TS) appended to Facility Operating License No. NPF-43 for Fermi-2. The proposed amendment would revise TS Figure 3.2.3-2, Flow Correction (K_f) factor. The figure was part of a previous license amendment application dated April 3, 1989, and issued by the NRC as Amendment No. 42 to the Fermi-2 Operating License and was recently found, on November 9, 1989, to be in error. Upon the discovery of this error, the licensee began a detailed review of Amendment No. 42 against its base-line documents. Particular attention was paid to figures. The following figures as submitted to the NRC in the original April 3, 1989, application were found to need modification to ensure the limits contained in the figures clearly express the limits required by the supporting analysis. The problems were either editorial or related to the clarity of the figure. The affected figures:

- Figure 3.2.1-3 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Reload Fuel Type BC318D
- Figure 3.2.1-4 Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Versus Average Planar Exposure Reload Fuel Type BC318E
- Figure 3.2.3-1 BOC to 12,700 MWD/ST Minimum Critical Power Ratio (MCPR) Versus Tau At Rated Flow
- Figure 3.2.3-1A 12,700 MWD/ST to 13,700 MWD/ST Minimum Critical Power Ratio (MCPR) Versus Tau At Rated Flow
- Figure 3.2.3-1B 13,700 MWD/ST to ECC Minimum Critical Power Ratio Versus Tau At Rated Flow

Bases Section 2.0 and 3/4.1.3 were found to retain references to the Cycle 1 MCPR Safety Limit of 1.06. Also, Bases Section 2.1.1 refers to the GEXL correlation, which applied only to Cycle 1. New Bases pages are attached which

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replace cycle-specific references with general references which are not cycle-specific. This is consistent with other Bases changes made in Amendment No. 42. Typographical and editorial errors were found in Bases Section 3/4.2.3 and Table B 3.2.1-1, the errors are corrected in the attached page changes.

In addition, typographical errors were found in the designation of fuel bundle types in Specification 3.2.4 and a reference to figures in Specification 4.2.3.1 was not changed to match the addition of new figures. The purpose of the November 16, 1989, application is to correct these errors.

2.0 EVALUATION

Each correction to the TS is addressed individually below.

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Limit Figures

These figures provide the MAPLHGR limits, in units of KW/ft, as a function of Average Planar Exposure, in units of MWD/t. The information is presented graphically as a series of line segments. Each point where the segments connect is given in tabular format.

Two new fuel types, BC318D and BC318E, have been loaded at Fermi-2 for Cycle 2; therefore, two new MAPLHGR limit figures were proposed. These figures are Figures 3.2.1-3 and 3.2.1-4, respectively, in the April 3, 1989, application.

In both figures, one point specified for the joining of two line segments was not graphically plotted consistent with the tabular information. In both figures, the point tabularly indicated to be associated with 12,500 MWD/t was graphically plotted at 12,000 MWD/t. Further, the heading for the MWD/t column in both figures is incorrectly given as "MWD/" rather than the correct units of "MWD/t."

The GE Nuclear Energy Supplemental Reload Licensing Submittal (RLS), provided as an attachment to the April 3, 1989, application gives the results of the accident analysis which is the basis for these curves. This information is located in Item 17 on page 15 of the RLS. The analysis results correspond to the tabular data on the figures.

The November 16, 1989 submittal contained revised Figures 3.2.1-3 and 3.2.1-4 with the graphs replotted to correctly indicate the RLS information. A corrected heading for the Average Planar Exposure column of the table is also included.

The change does not reflect any change to the accident analysis which was reviewed and approved by the NRC in conjunction with Amendment No. 42. The change strictly corrects administrative errors made in the preparation of the April 3, 1989, application.

Minimum Critical Power Ratio (MCPR) Versus Tau at Rated Flow Limit Figures

Amendment No. 42 issued three figures, each for a specific period of core life (measured in MWD/ST), which provide MCPR operating limits as a function of a variable, tau. Tau, is defined in Specification 3.2.3 and is a measure of the control rod scram speed performance as related to that assumed in the General

Electric analysis. Since scram times can be less conservative than that assumed in the analysis, but still within those allowed by the Technical Specifications, the MCPR operating limits is adjusted conservatively upward when necessary by referring to these figures.

MCPR versus tau curves are provided in each figure for four possible operating conditions. These conditions are combinations of Control Cell Core (CCC) or non-CCC operating modes, and the availability of turbine bypass capability or moisture separator reheater.

The nomenclature associated with Curve C and Curve D, which appears on each of these figures, was found to be unclear as proposed in the April 3, 1989, application and as issued in Amendment No. 42. The nomenclature is proposed to more clearly indicate which equipment is assumed to be out-of-service for the two curves. Previously, the terminology "without a piece of equipment" was used to mean "with the equipment out-of-service." The proposal is strictly editorial and makes no change to the intent or application of the curves.

On Figure 3.2.3-1, the MCPR value for tau equal to 1.0 for the merged Curve A and B is not given on the figure. This point corresponds to a MCPR of 1.32. This value is the "Option A" most limiting pressurization event for the exposure range "BOC2 to EOC2 - 2000 MWD/St" located on RLS page 13.

On all three figures, Curve B is a hybrid of two limiting events. One is the non-pressurization event of a rod withdrawal error which has a limiting MCPR of 1.30 for GE8x8EB fuel (see RLS page 13). The other is the pressurization event for the core exposure of concern which yielded Curve A. Thus, Curve B is a constant 1.30 for tau less than the tau for the point where the Curve A event becomes more limiting.

On Figures 3.2.3-1A and 3.2.3-1B a value of less than 1.30 appears to be indicated for a portion of Curve B which should be a constant 1.30. This occurs near the point where Curves A and B join. This proposal corrects Curve B for these two figures by indicating the correct value of 1.30 for the constant MCPR segments of Curve B.

In summary, all of the changes to the MCPR limit figures are either editorial or corrections of administrative errors in the plotting of RSL supplied data points. There is no change in the underlying accident analysis which was reviewed and approved by the NRC in conjunction with Amendment No. 42.

Flow Correction (K_f) Factor

Flow Correction (K_f) Factor as a function of Core Flow (%) is provided in Figure 3.2.3-2. The Operating Limit MCPR determined from the use of Figure 3.2.3-1, 3.2.3-1A, or 3.2.3-1B (depending on core exposure) is multiplied by K_f . This increases the MCPR limit further to account for a potential Reactor Recirculation Pump Runout transient. Since the severity of this transient depends on the Flow Control Mode and, if in manual mode, the Scoop Tube Set Point Calibration, five curves are provided to account for the possible configuration.

As discussed above under Introduction, these figures were incorrectly drawn in the preparation of the April 3, 1989, application. The sole change to these figures is to provide the correct curves.

The new K_f values were made necessary for Cycle 2 due to the application of the GEXL-PLUS critical power correlation to the Cycle 2 reload licensing analyses. The Cycle 1 analysis used the GEXL correlations. Both correlations provide a means to relate bundle critical power, that power which leads to transient boiling, to parameters which can be sensed and provided to the plant process computer.

The sole purpose of the proposed change is to correct an administrative error made in the preparation of the April 3, 1989, application. No change to the underlying accident analysis has been made.

Administrative Errors

Amendment No. 42 revised Specification 3.2.4 to indicate separate Linear Heat Generation Rate (LHGR) limits for different fuel bundle types. In Amendment No. 42, bundle types 8CR183 and 8CR233 were incorrectly identified as BCR183 and BCR233. Also, in Specification 4.2.3.1 reference to Figure 3.2.3-1 was not changed to include new Figures 3.2.3-1A and 3.2.3-1B. The new figures were created by the specification of MCPR limits based upon core exposure. Specification 4.2.3.1 is proposed to be changed in a manner similar to that approved in Amendment No. 42 for Specification 4.2.3.2 to ensure that all of the curves are referenced.

Bases Sections 2.0 and 3/4.1.3 were found to mention that the MCPR Safety Limit is 1.06, which was the Cycle 1 limit. These sections were not originally identified as needing changing. Elsewhere the term "Safety Limit MCPR" was inserted in the April 3, 1989, application where the specific value previously had been. These sections are now corrected to reference the Safety Limit MCPR. Also, since Bases Section 3/4.1.3 is being revised, the reference to Section 15B of the FSAR is being updated to Section 15 of the UFSAR (Updated Final Safety Analysis Report). A mention of the Cycle 1 GEXL analyses was inadvertently left in Bases Section 2.1.1. As discussed above, the GEXL-PLUS correlation was used for Cycle 2. The wording is being corrected to mention the "approved critical power correlation." On Bases Table B 3.2.1-1, the unit of area for a small break is listed as "ft" rather than "ft²". This is being corrected. The last administrative errors needing correction are in Bases Section 3/4.2.3. The word "extended" was mistyped in Amendment No. 42 as "extending." Additionally, the descriptions of the CCC operations mode and non-CCC mode need slight changes to match descriptions provided elsewhere in the Technical Specifications. These errors are being corrected.

Based on the above evaluation the staff finds the proposed changes are either administrative or editorial in nature. The technical analysis as presented in the April 3, 1989, application and reviewed and approved by the staff in Amendment No. 42 is not changed in anyway. Therefore, the staff finds the proposed changes to correct the TS pages are acceptable.

3.0 EMERGENCY CIRCUMSTANCES

In the November 16, 1989, letter the licensee requested that this amendment be proposed as an emergency because if the NRC does not process the application the licensee's restart schedule will be affected. Fermi-2 is currently shutdown for refueling and scheduled to commence reactor startup on November 20, 1989. Escalating above 25% power is scheduled to occur as soon as plant conditions allow.

As described above, the proposed changes correct or make clear Technical Specification limits associated with the Cycle 2 fuel loading. The affected Specifications are all applicable at greater than 25% rated thermal power. The licensee does not believe operation with applicable Technical Specifications which are non-conservatively incorrect is acceptable. In this case, the need for the amendment arose from the diligent investigation of a recently discovered discrepancy between the Technical Specifications for Cycle 2 and the process computer data for Cycle 2. Upon discovery of the need for a license amendment, the November 16, 1989 application was made in a prompt manner. Due to the schedule detailed above, adequate time for the usual 30 day period for public comment does not exist.

In accordance with 10 CFR 50.91(a)(5), the licensee has provided justification that it could not make a timely application and emergency circumstances do exist. Thus, the NRC staff does not believe that the licensee has abused the emergency provisions in this instance. Accordingly, the Commission has determined that there are emergency circumstances warranting prompt approval by the Commission.

4.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations, if operation of the facility, in accordance with the amendment would not:

- (1) Involve a significant increase in the probability or consequences of any accident previously evaluated; or
- (2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or
- (3) Involve a significant reduction in a margin of safety.

This amendment has been evaluated against the standards in 10 CFR 50.92. It does not involve a significant hazards consideration based on the following:

- (1) The changes do not involve a significant increase in the probability or consequences of an accident previously evaluated (10 CFR 50.92(c)(1)) because there will be no physical changes to the facility and all operating procedures, limiting conditions for operation, limiting safety system settings, and safety limits currently delineated in the Technical Specifications remain unchanged.

- (2) The changes do not create the possibility of a new or different kind of accident from any previously analyzed or evaluated (10 CFR 50.92)(c)(2)) because neither plant operation or design are affected by the proposed changes. The proposed amendment creates no new accident scenario.
- (3) The change does not involve a significant reduction in a margin of safety (10 CFR 50.92(c)(3)) because there will be no physical changes to the facility and the requirements delineated in the current Technical Specifications for limiting conditions for operation, limiting safety system settings, and safety limits remain unchanged.

On the basis of the above consideration, the staff proposes to find that the changes do not involve a significant hazards consideration.

5.0 STATE CONSULTATION

In accordance with the Commission's regulations, efforts were made to contract the Michigan State representative. The staff representative was contacted and had no comments.

6.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that this amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents which may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

7.0 CONCLUSION

We have 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, (3) and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: John Stang

Date: November 21, 1989