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.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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	3/4 2-4a
3/4 2-7	3/4 2-7
3/4 2-8	3/4 2-8
3/4 2-8a	3/4 2-8a
3/4 2-8b	3/4 2-8b
3/4 2-9	3/4 2-9
3/4 2-10	3/4 2-10
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	B 3/4 2-4
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

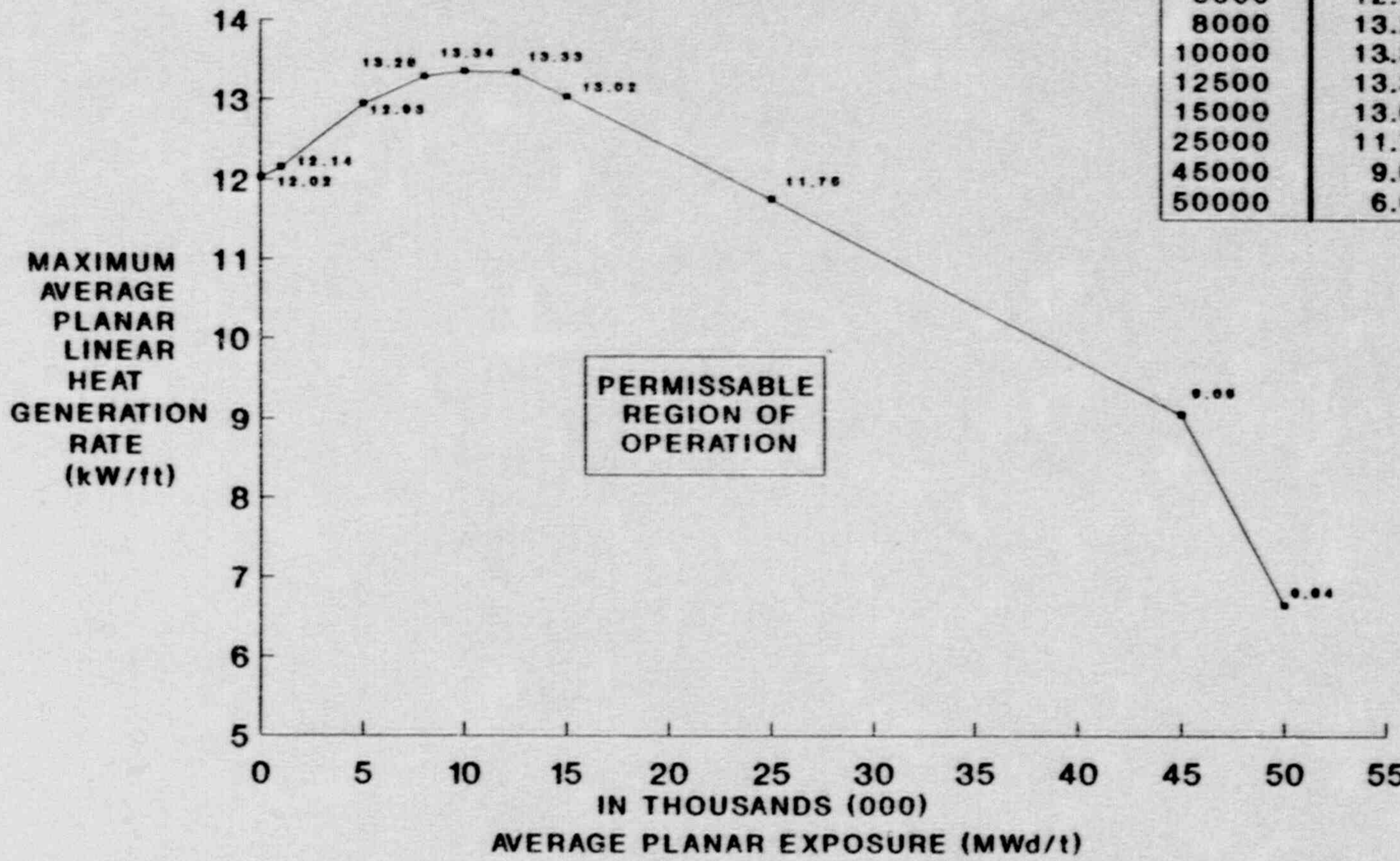
BASES

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.

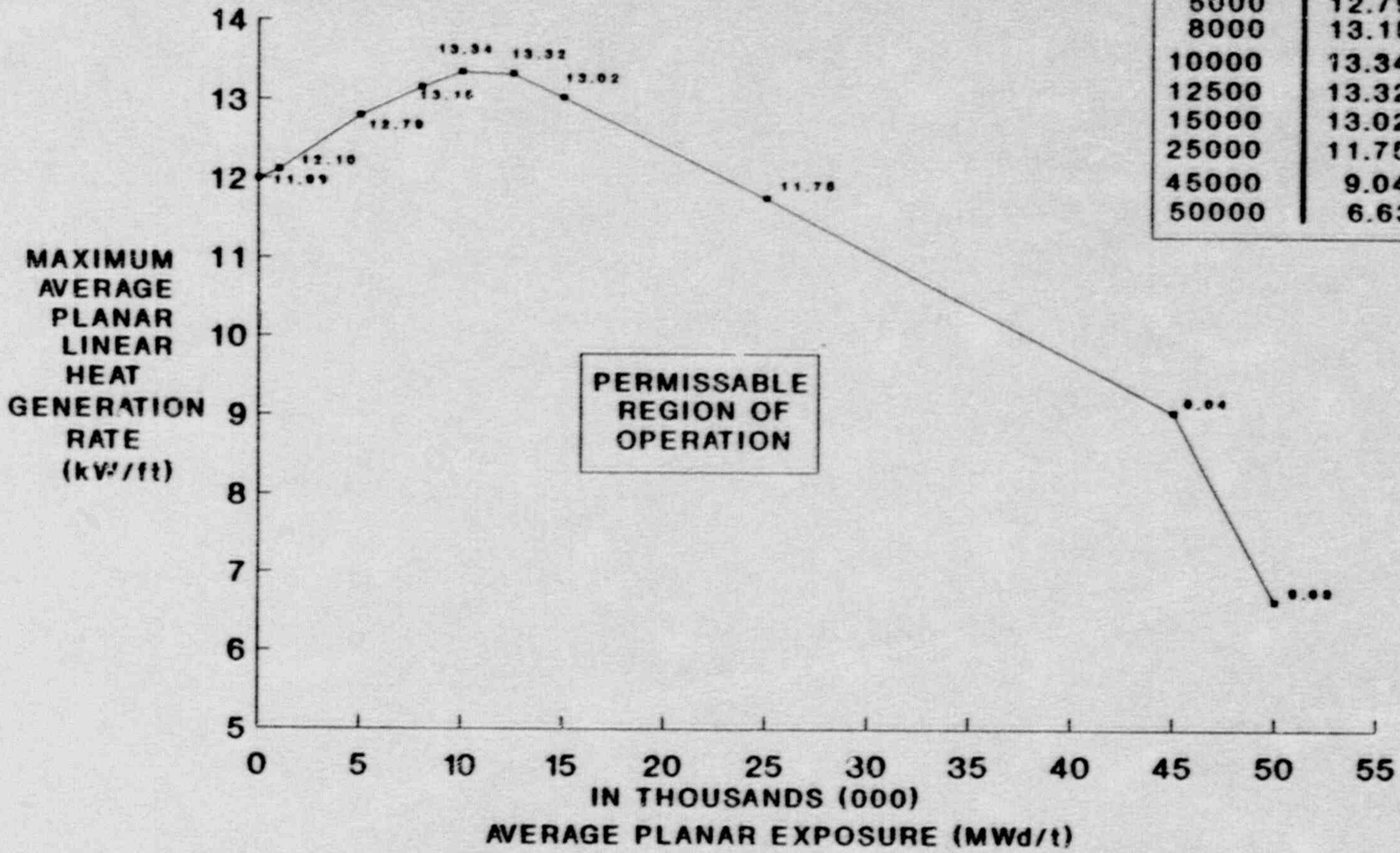


MWd/t	kW/ft
0	12.02
1000	12.14
5000	12.93
8000	13.28
10000	13.34
12500	13.33
15000	13.02
25000	11.75
45000	9.05
50000	6.64

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

FIGURE 3.2.1-3

MWd/t	kW/ft
0	11.99
1000	12.10
5000	12.79
8000	13.15
10000	13.34
12500	13.32
15000	13.02
25000	11.75
45000	9.04
50000	6.63



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

FIGURE 3.2.1-4

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:

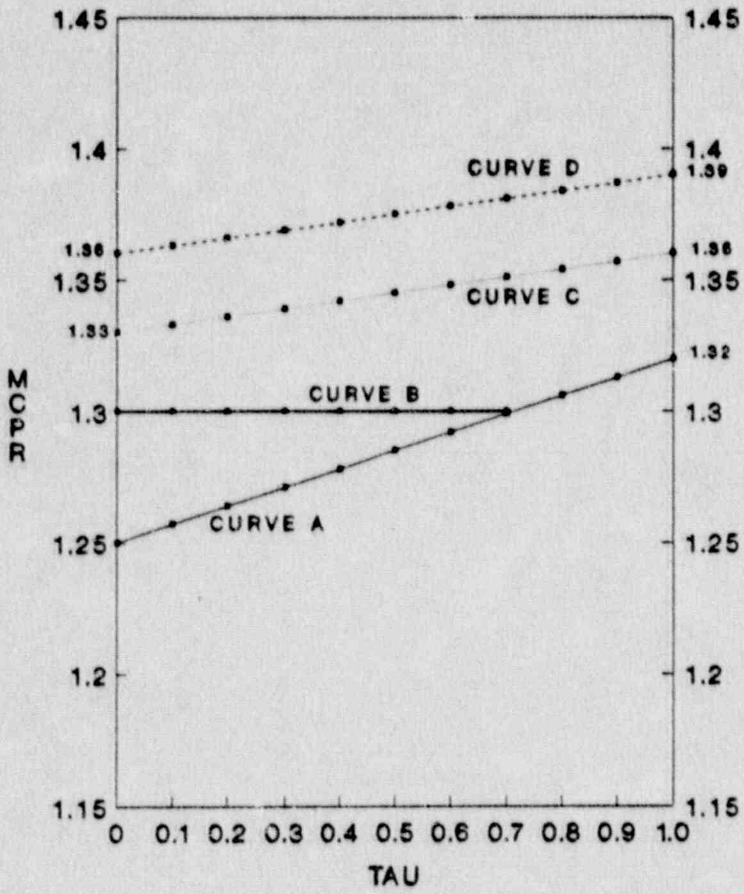
a. At least once per 24 hours,

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 seperator reheater in service.

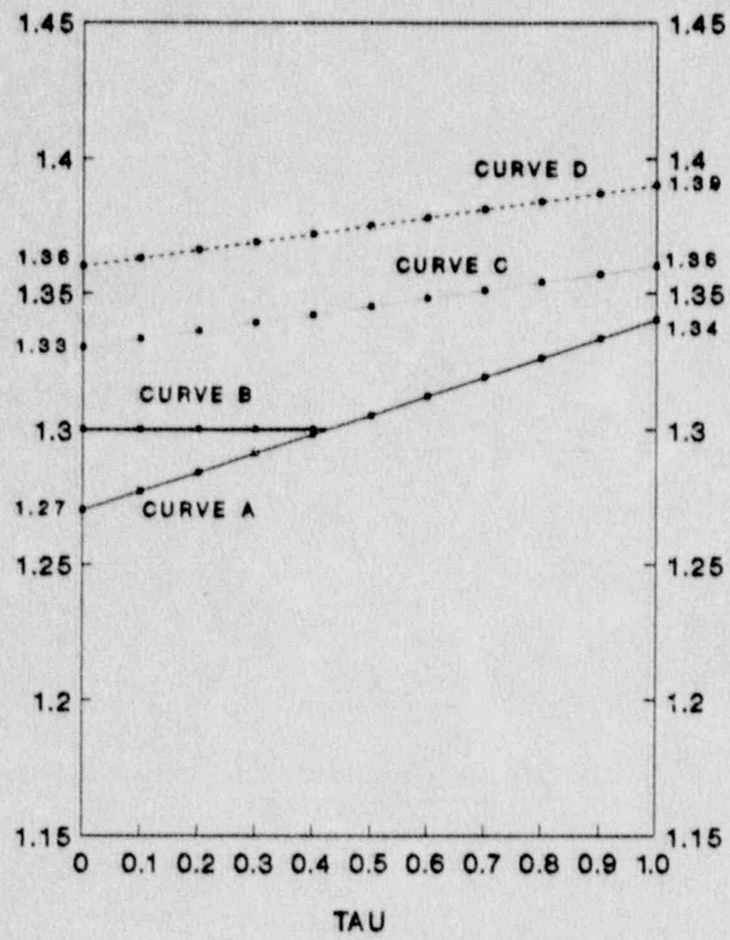
CURVE B - MCPR limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.

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.

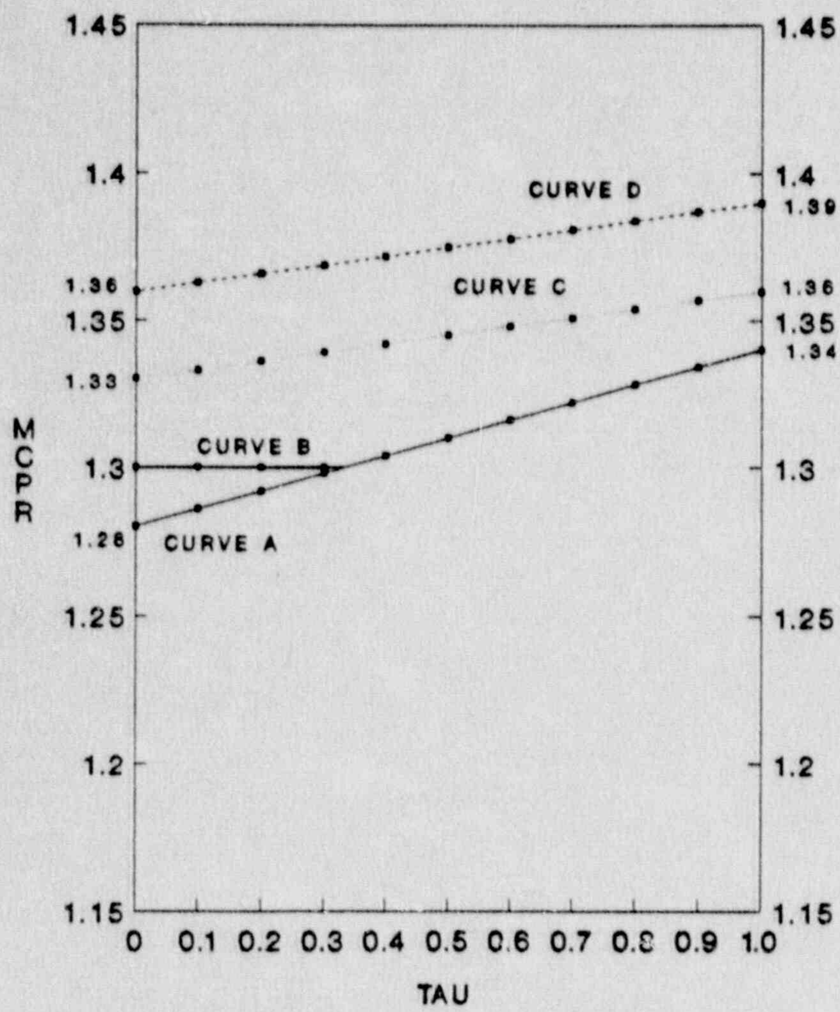
CURVE B - MCPR limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.

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 - MCPR limit for CCC operational mode with both turbine bypass and moisture separator reheater in service.

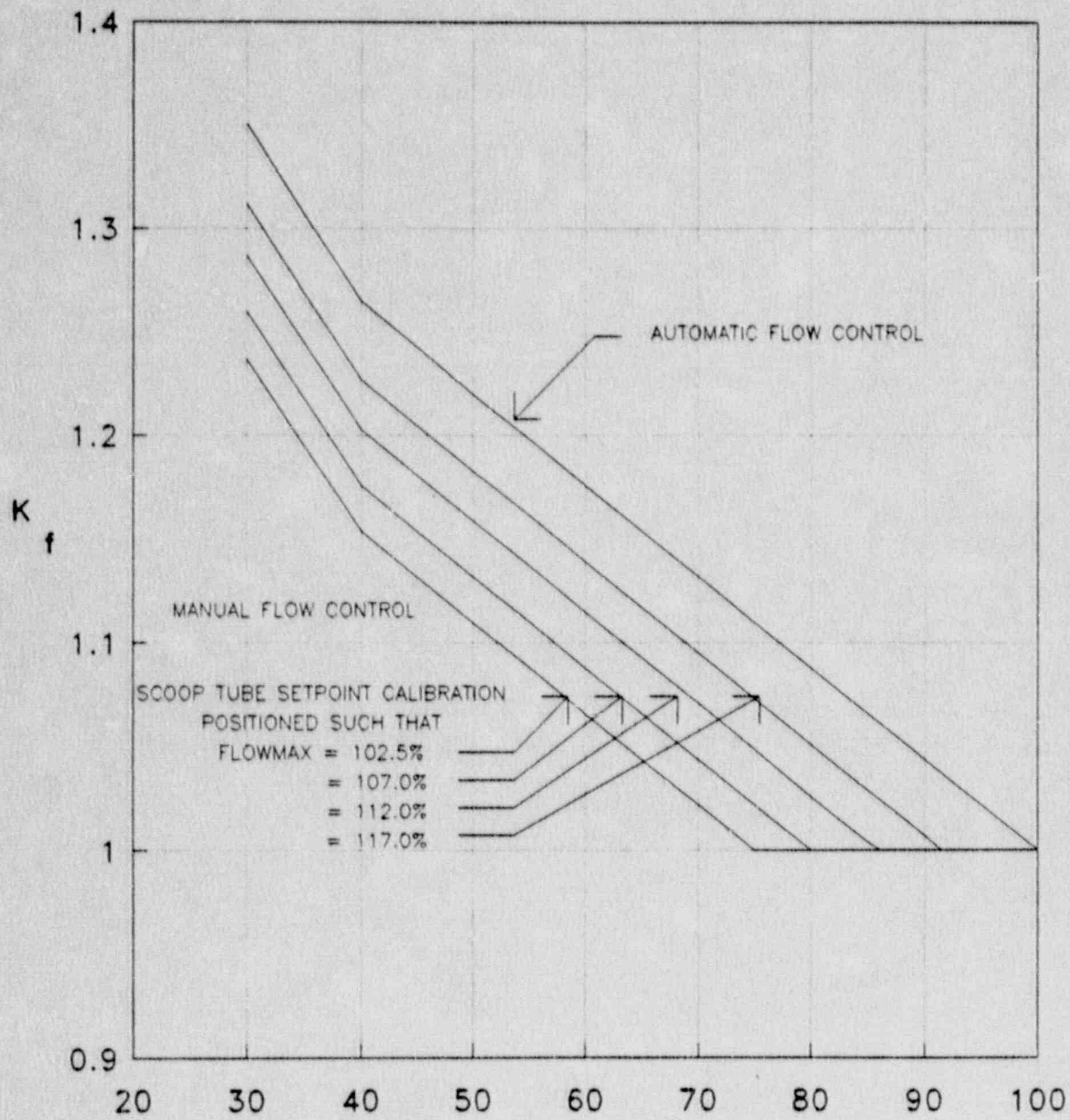
CURVE B - MCPR limit for non-CCC operational mode with both turbine bypass and moisture separator reheater in service.

CURVE C - MCPR limit for both CCC and 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.

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

FIGURE 3.2.3-1B



FLOW CORRECTION (K_f) FACTOR

FIGURE 3.2.3-2

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.

3/4.1.1 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.

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.

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 inoperable. 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 inoperable. The operating domain includes the

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 then 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.