EXHIBIT B

Revision 1 License Amendment Request Dated - Sept 7, 1976

Exhibit B, attached, consists of the following revised pages of the Appendix A Technical Specifications which incorporate the proposed changes.

Pages

6
7
8
14
15
17
20
56
57
114
114a (new page)
119
211
213
214
215
216
217

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2.1 FUEL CLADDING INTEGRITY

Applicability:

Applies to the interrelated variables associated with fuel thermal behavior.

Objectives:

To establish limits below which the integrity of the fuel cladding is preserved.

Specification:

A. Core Thermal Power Limit (Reactor Pressure > 800 Psia and Core Flow is > 10% of Rated)

When the reactor pressure is >800 Psia and core flow is >10% of rated, the existence of a minimum critical power ratio (MCPR) less than 1.07 for two recirculation loop operation or less than 1.08 for single loop operation for 8x8 and 8x8R fuel shall constitute violation of the fuel cladding integrity safety limit.

2.3 FUEL CLADDING INTEGRITY

Applicability:

Applies to trip settings of the instruments and devices which are provided to prevent the reactor system safety limits from being exceeded.

Objectives:

To define the level of the process variables at which automatic protective action is initiated to prevent the safety limits from being exceeded.

Specification:

The Limiting safety system settings shall be as specified below:

A. Neutron Flux Scram

1. APRM - The APRM flux scram trip setting shall be:

$$S \leq 0.65\ (W-dw) + 55\%$$

where,

S = Setting of percent of rated thermal power, rated power being 1670 MWT

W = recirculation drive flow in percent

dw = single loop operation recirculation reverse flow in the idle loop.

dw = 0 For two recirculation loop operation

dw = 5.4 For one recirculation loop operation

B. Core Thermal Power Limit (Reactor Pressure $\leq$ 800 psia or Core Flow $\leq$ 10% of rated)

When the reactor pressure is $\leq$ 800 psia or core flow is $\leq$ 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transients

To insure that the safety limit established in Specification 2.1.A is not exceeded, each required scram shall be initiated by its primary source signal as indicated by the plant process computer

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq [0.65\ (W\text{-}dw) + 55\%]\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

2. IRM - Flux Scram setting shall be 20% of rated neutron flux

B. APRM Rod Block - The APRM rod block setting shall be:

$$S \leq 0.65\ (W\text{-}dw) + 43\%$$

where,

S = Setting of percent of rated thermal power, rated power being 1670 MWT

W = recirculation drive flow in percent

dw = Single loop operation recirculation reverse flow in the idle loop.

dw = 0 For two recirculation loop operation

dw = 5.4 For one recirculation loop operation

D. Reactor Water Level (Shutdown Condition)

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core. This level shall be continuously monitored whenever the recirculation pumps are not operating.

except in the event of operation with a maximum fraction of limiting power density for any fuel type in the core greater than the fraction of rated power, when the setting shall be modified as follows:

$$S \leq [0.65\ (W\text{-}dw) + 43\%]\ \frac{FRP}{MFLPD}$$

where,

FRP = fraction of rated thermal power, rated power being 1670 MWt

MFLPD = maximum fraction of limiting power density for any fuel type in the core.

C. Reactor Low Water Level Scram setting shall be $\geq$ 10'6" above the top of the active fuel.

D. Reactor Low Low Water Level ECCS initiation shall be $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel.

Bases:

2.3 The abnormal operational transients applicable to operation of the Monticello Unit have been analyzed throughout the spectrum of planned operating conditions up to the thermal power level of 1670 MWt. The analyses were based upon plant operation in accordance with the operating map given in Figure 3-2-3 of the FSAR. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Transient analysis performed each reload are given in Reference 1. Models and model conservatisms are also described in this reference. As discussed in Reference 2, the core wide transient analysis for one recirculation pump operation is conservatively bounded by two-loop operation analysis and the flow-dependent rod block and scram setpoint equations are adjusted for one-pump operation.

Bases Continued:

Deviations from as-left settings of setpoints are expected due to inherent instrument error, operator setting error, drift of the setpoint, etc. Allowable deviations are assigned to the limiting safety system settings for this reason. The effect of settings being at their allowable deviation extreme is minimal with respect to that of the conservatisms discussed above. Although the operator will set the setpoints within the trip settings specified, the actual values of the various setpoints can vary from the specified trip setting by the allowable deviation.

A violation of this specification is assumed to occur only when a device is knowlingly set outside of the limiting trip setting or when a sufficient number of devices have been affected by any means such that the automatic function is incapable of preventing a safety limit from being exceeded while in a reactor mode in which the specified function must be operable. Sections 3.1 and 3.2 list the reactor modes in which the functions listed above are required.

The bases for individual trip settings are discussed in the following paragraphs.

A. Neutron Flux Scram The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady state conditions, reads in percent of rated thermal power (1670 MWt). Because fission chambers provide the basic input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel. Therefore, during abnormal operation transients, the thermal power of the fuel will be less than backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be closed. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 18.

B. <u>APRM Control Rod Block Trips</u> Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit

Bases Continued:

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 18.

References

1. "Generic Reload Fuel Application", NEDE 24011-P-A-1, July 1979

2. "Monticello Nuclear Generating Plant Single-Loop Operation" NEDO 24271, June 1980

Table 3.2.3
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup	Run			
1. SRM							
a. Upscale	$\leq 5x10^5$ cps	X	X(d)		2	1 (Note 3)	A or B or C
b. Detector not fully inserted		X(a)	X(a)		2	1 (Note 3)	A or B or C
2. IRM							
a. Downscale	$\geq$ 3/125 full scale	X(b)	X(b)		4	2 (Note 4)	A or B or C
b. Upscale	$\leq$ 108/125 full scale	X	X		4	2 (Note 4)	A or B or C
3. APRM							
a. Upscale (flow referenced)	See Technical Specifications 2.3.B.			X	3	1 (Note 7)	D or E
b. Downscale	$\geq$ 3/125 full scale			X	3	1 (Note 7)	D or E

Table 3.2.3 - continued
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must be Operable or Operating and Allowable Bypass Conditions**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup	Run			
4. RBM							
a. Upscale (flow referenced)	See Technical Specifications 2.3.B			X(c)	1	1 (Note 5)	D or E
b. Downscale	$\geq$3/125 full			X(c)	1	1 (Note 5)	D or E
5. Scram Discharge Volume							
Water Level-High	$\leq$18 gal		X	X	1	1	B and D, or A

Notes:

(1) There shall be two operable or operating trip systems for each function. If the minimum number of operable or operating instrument channels cannot be met for one of the two trip systems, this condition may exist up to seven days provided that during this time the operable system is functionally tested immediately and daily thereafter.

(2) (deleted)

(3) Only one of the four SRM channels may be bypassed.

(4) There must be at least one operable or operating IRM channel monitoring each core quadrant.

(5) One of the two RBMs may be bypassed for maintenance and/or testing for periods not in excess of 24 hours in any 30 day period. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

3.0 LIMITING CONDITIONS FOR OPERATION

I. Recirculation System

1. Except as specified in 3.5.1.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

3. Reactor operation with one loop recirculation may continue at up to 50% of rated power if the following conditions are met within 24 hours after one pump operation commences. If the conditions cannot be met or two pump operation cannot be restored by the end of 24 hours, an orderly reactor shutdown shall be initiated.

   a. The Minimum Critical Power Ratio (MCPR) Safety Limit will be increased per T.S. 2.1.A

   b. The MCPR Limiting Condition for Operation (LCO) will be changed per T.S. 3.11.C.

   c. The Maximum Average Planar Linear Heat Generation (MAPLHGR) will be changed as noted in Table 3.11.1

4.0 SURVEILLANCE REQUIREMENTS

I. Recirculation System

1. Once per month, when irradiated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross time valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

3. When in one loop operation, the following surveillances will be completed:

   a. APRM flux noise will be measured once per shift and the recirculation pump speed will be reduced if the flux noise average over ½ hour exceeds 5% peak to peak as measured on the APRM chart recorder.

   b. The core plate delta P noise will be measured once per shift and the recirculation pump speed will be reduced if the noise exceeds 1 psi peak to peak.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
d. The APRM scram and rod block setpoints and the RBM setpoints shall be reduced as noted in T.S. 2.3.A and T.S. 2.3.B. e. The suction valve or the main discharge and main discharge bypass valves in the idle loop is closed and electrically isolated until the idle loop is being prepared for return to service. f. The equalizer line shall be isolated.	

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

H. Deleted

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. Such a break involves 3.9 sq. ft. when the cross tie valves are closed and 5.3 sq. ft. when the cross tie valves are open. Specification 3.11.A is based on an ECCS evaluation assuming a break area of 3.9 sq. ft.; the limitations of 3.11.A do not apply to the larger break area. Therefore, at least one cross tie valve must remain closed during power operation to reduce the potential break area.

An analysis of one-pump operation (equalizer valve closed) identifies certain limitations peculiar to that mode of operation. Reference the September 7, 1976 License Amendment Request from NSP to NRR. Operation with only one pump is not a normal mode; it will generally involve a forced outage of equipment. There may be insufficient time to make adjustments to the RBM and APRM flow referenced rod block and scram prior to commencing one-pump operation. The reduction in power with the reduced core flow will cause the APLHGR to reduce accordingly, naturally moving in the direction of the new limit. Specification 3.5.I.3 allows 24 hours before these new limits are required to be implemented.

3.11 REACTOR FUEL ASSEMBLIES

Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

During power operation, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed for two recirculation loop operation the limiting value given in Table 3.11.1 based on a straight line interpolation between data points and for one recirculation loop operation the values in Table 3.11.1 reduced by 0.85 for all fuel types. When core flow is less than 90% of rated core flow, the APLHGR shall not exceed 95% of the limiting value given in Table 3.11.1. When core flow is less than 70% of rated core flow, the APLHGR shall not exceed 90% of the limiting value given in Table 3.11.1. If any time during operation it is determined that the limit for APLHGR is being exceeded, action shall be initiated within 15

4.11 REACTOR FUEL ASSEMBLIES

Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

Objective

The objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

Specifications

A. Average Planar Linear Heat Generation Rate (APLHGR)

The APLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at 25% rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

1. During power operation the Operating MCPR Limit shall be $\geq$ 1.43 for 8x8 and 8x8R fuel, $\geq$ 1.47 for P8x8R fuel at rated power and flow for two recirculation loop operation, provided $\tau_B \geq \tau_{AVE}$* (see section 3.3.C.3). If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown conditions within 36 hours. For core flows other than rated the Operating MCPR Limit shall be the above applicable MCPR value time K_f where K_f is as shown in Figure 3.11.3.

   For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

2. If the gross radioactivity release rate of noble gases at the steam jet air ejector monitors exceeds, for a period greater than 15 minutes, the equivalent of 236,000 uCi/sec following a 30-minute decay, the Operating MCPR Limits specified in 3.11.C.1 shall be adjusted to $\geq$1.48 for all fuel types, times the appropriate K_f. Subsequent operation with the adjusted MCPR values shall be per paragraph 3.11.C.1.

   For one recirculation loop operation the MCPR limits at rated flow are 0.01 higher than the comparable two-loop values.

*If $\tau_{AVE} > \tau_B$, the operating MCPR Limit shall be a linear interpolation between the limits in 3.11.C.1 and 1.48 for 8x8 and 8x8R fuel and 1.52 for P8x8R fuel.

4.0 SURVEILLANCE REQUIREMENTS

C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at $\geq$ 25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR Limit.

TABLE 3.11.1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE vs. EXPOSURE

Exposure	MAPLHGR FOR EACH FUEL TYPE (kw/ft) (Note 1)							
MWD/STU	8DB262	8DB250	8DB219L	8DRB265L	P8DRB265L	8DRB282	P8DRB282	P8DRB284LB
200	11.1	11.2	11.4	11.5	11.6	11.2	11.2	11.4
1,000	11.3	11.3	11.5	11.6	11.6	11.2	11.2	11.4
5,000	11.9	11.9	11.9	11.7	11.8	11.6	11.8	11.8
10,000	12.1	12.1	12.0	11.8	11.9	11.7	11.9	11.9
15,000	12.1	12.1	11.9	11.7	11.9	11.7	11.8	11.9
20,000	12.0	11.9	11.8	11.6	11.8	11.5	11.7	11.7
25,000	11.6	11.5	11.3	11.3	11.3	11.3	11.3	11.4
30,000	10.3	10.6	10.2	10.3	10.5	10.4	10.7	10.6
35,000	9.3	9.3	9.3	9.2	9.5	9.2	9.5	9.5
(36,000)	9.1	9.0	9.1	9.0	9.3	9.0	9.3	9.3
40,000	8.9*							
45,000	8.0*							
50,000	7.3*							

(1) For two recirculation loop operation. For one recirculation loop operation multiply these values by 0.85

*For extended burnup program test bundles

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10CFR50, Appendix K.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than $\pm$ 20° relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are within the 10CFR50 Appendix K limit. The limiting value for APLHGR is given by this specification.

Reference 6 demonstrates that for lower initial core flow rates the potential exists for earlier DNB during postulated LOCA's. Therefore a more restrictive limit for APLHGR is required during reduced flow conditions.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding APLHGR limits in such cases need not be reported.

Reduction factors for one recirculation loop operation were derived in Reference 8.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation if fuel pellet densification is postulated. The power spike penalty specified is based on the analysis presented in Section 3.2.1 of Reference 1 and in References 2 and 3, and assumes a linearly increasing variation and axial gaps between core bottom and top and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding LHGR limits in such cases need not be reported.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 and Reference 6 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be 1.24 for all fuel types for normal and reduced flow. The Operating MCPR Limit for two recirculation loop operation is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

For one recirculation loop operation the MCPR limits at rated flows are 0.01 higher than the comparable two-loop values.

Use of GE's new ODYN code Option B will require average scram time to be a factor in determining the MCPR (Reference 7). In order to increase the operating envelope for MCPR below $MCPR_A$ (ODYN code Option A), the cycle average scram time (τ_{AVE}) must be determined (see Bases 3.3.C). If τ_{AVE} is below the adjusted analysis scram time, the $MCPR_B$ Limit can be used. If $\tau_{AVE} > \tau_B$ a linear interpolation must be used to determine the appropriate MCPR. For example:

$$MCPR = MCPR_B + \frac{\tau_{AVE} - \tau_B}{0.9 - \tau_B}(MCPR_A - MCPR_B)$$

$MCPR_A$ and $MCPR_B$ have been determined from the most limiting accident analyses.

For operation with less than rated core flow the Operating MCPR Limit is adjusted by multiplying the above limit by K_f. Reference 5 discusses how the transient analysis done at rated conditions encompasses the reduced flow situation when the proper K_f factor is applied.

Noble gas activity levels above that stated in 3.11.C.2 are indicative of fuel failure. Since the failure mode cannot be positively identified, a more conservative Operating MCPR Limit must be applied to account for a possible fuel loading error.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "Fuel Densification Effects in General Electric Boiling Water Reactor Fuel," Supplements 6,7, and 8, NEDM-10735, August, 1973.

2. Supplement 1 to Technical Report on Densification of General Electric Reactor Fuels, December 14, 1974 (USAEC Regulatory Staff).

3. Communication: VA Moore to IS Mitchell, "Modified GE Model for Fuel Densification," Docket 50-321, March 27, 1974.

4. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant," NEDO-24050 -1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.

5. "General Electric BWR Generic Reload Application for 8x8 Fuel," NEDO-20360, Revision 1, November 1974.

6. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWR's," R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.

7. "Response to NRC Request for Information on ODYN Computer Mode," R H Buchholz (GE) to P S Check (USNRC), September 5, 1980.

8. "Monticello N.G.P. Single-loop Operation NEDO 24271, June 1980"

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement have caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. In addition, the MCPR is checked whenever changes in the core power level or distribution are made which have the potential of bringing the fuel rods to their thermal-hydraulic limits.