November 25, 1987

Docket No. 50-263

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Mr. D. M. Musolf, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment No. 54 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. This amendment is in response to your application dated July 27, 1987, as supplemented by letters dated August 28, September 3 and 16, 1987.

The amendment revises the Technical Specifications to reflect the changes supported by analysis for the reload justifying Cycle 13 operation. The issuance of this amendment completes our work effort under TAC 65963.

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next regular bi-weekly <u>Federal Register</u> notice.

Sincerely,

Dominic C. DiIanni

Dominic C. DiIanni, Project Manager
Project Directorate III-3
Division of Reactor Projects

Enclosures:
1. Amendment No. 54 to
   License No. DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. D. M. Musolf
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Monticello Nuclear Generating Plant

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 54
License No. DPR-22

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

   A. The application for amendment by Northern States Power Company (the licensee) dated July 27, 1987, as supplemented August 28 and September 3 and 16, 1987 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. DPR-22 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Kenneth E. Perkins, Director
Project Directorate III-3
Division of Reactor Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 25, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages are identified by amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
vii	vii
14	14
20	20
82	82
90	90
114b	114b
211	211
212	212
213	213
214	214
215	
215b	215b
216	216
217	217
218	218
230	230

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 3-1 of Reference 2. The licensed maximum power level 1670 MWt represents the maximum steady-state power which shall not knowingly be exceeded.

Conservatism is incorporated in the transient analyses in estimating the controlling factors, such as void reactivity coefficient, control rod scram worth, scram delay time, peaking factors, and axial power shapes. These factors are selected conservatively with respect to their effect on the applicable tranisent results as determined by the current analysis model. Conservatism incorporated into the transient analysis is documented in Reference 1.

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 39.

References

1. "General Electric Standard Application for Reactor Fuel", NEDE-24011-P-A (as amended).
2. "Average Power Range Monitor, Rod Block Monitor and Technical Specifications Improvement (ARTS) Program for Monticello Nuclear Generating Plant", NEDC-30492-P, April, 1984.
3. "Monticello Nuclear Generating Plant Single Loop Operation", NEDO-24271, June, 1980.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.3.A are met.

D. Control Rod Accumulators

In the "Startup" or "Run" Mode, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:

1. Inoperable accumulator.

2. Directional control valve electrically disarmed while in a non-fully inserted position.

If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.

In the "Refuel" Mode, the accumulator associated with any withdrawn control rod must be Operable unless all the fuel has been removed from the cell containing that control rod.

D. Control Rod Accumulators

Once a shift check the status in the control room of the required Operable accumulator pressure and level alarms

Bases Continued 3.3 and 4.3:

The analysis assumes 50 milliseconds for Reactor Protection System delay, 200 milliseconds from de-energization of scram solenoids to the beginning of rod motion, and 175 milliseconds later the rods are at the 5% position.

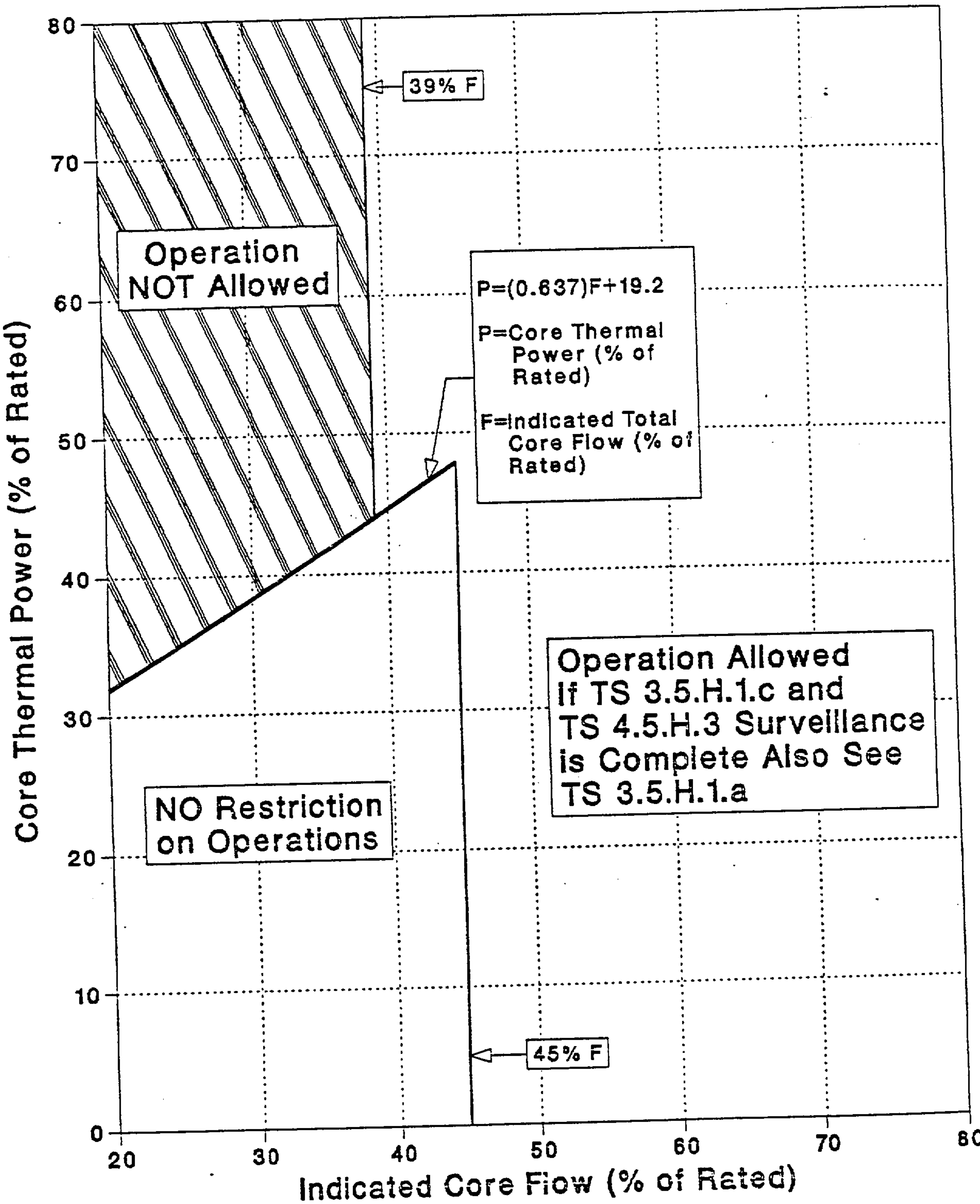


Figure 3.5-1 Single Loop Operation Surveillance Power/Flow Curve

3.11 REACTOR FUEL ASSEMBLES

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 two recirculation loop power operation, the APLHGR limiting condition for operation for each type of fuel as a function of axial location and average planar exposure shall not exceed limits based on applicable APLHGR limit values which have been approved for the respective fuel and lattice types as determined by the approved methodology described in NEDE-24011-P-A (GESTAR II). This approval is based on and limited to GESTAR II methodology. When hand calculations are required, the APLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value for the most limiting lattice (excluding natural uranium) shown in Table 3.11.1 (based on straight line interpolation between data points) multiplied by the smaller of the two MAPFAC factors determined from Figures 3.11.1 and 3.11.2.

During one recirculation loop power operation, the APLHGR limiting condition for operation for each type of fuel shall not exceed the above values multiplied by 0.85.

If at any time during power operation, it is determined that the APLHGR limiting condition for operation 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 APLHGR is not returned to within the prescribed limits within two hours, reduce thermal power to less than 25% within the next four hours.

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 $\geq$25% rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR shall be less than or equal to 13.4 Kw/ft for all fuel types except GE8x8EB(GE8) fuel, and less than or equal to 14.4 Kw/ft for GE8x8EB fuel.

If at any time during operation it is determined that the limiting value for LHGR 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 LHGR is not returned to within the prescribed limits within 2 hours, reduce thermal power to less than 25% within the next 4 hours.

4.0 SURVEILLANCE REQUIREMENTS

B. Linear Heat Generation Rate (LHGR)

The LHGR shall be checked daily during reactor operation at ≥25% of rated thermal power.

3.0 LIMITING CONDITIONS FOR OPERATION

C. Minimum Critical Power Ratio (MCPR)

If thermal power is greater than 45%, the MCPR limit is the greater of:

1) 1.30 multiplied by K_P from Figure 3.11.3 or,

2) $MCPR_F$ from Figure 3.11.4.

If thermal power is less than or equal to 45%, the MCPR limit is obtained from Figure 3.11.3.

The OLMCPR limit for one recirculation loop operation is 0.01 higher than the comparable two loop value.

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 hours, reduce thermal power to less than 25% within the next four hours.

4.0 SURVEILLANCE REQUIREMENTS

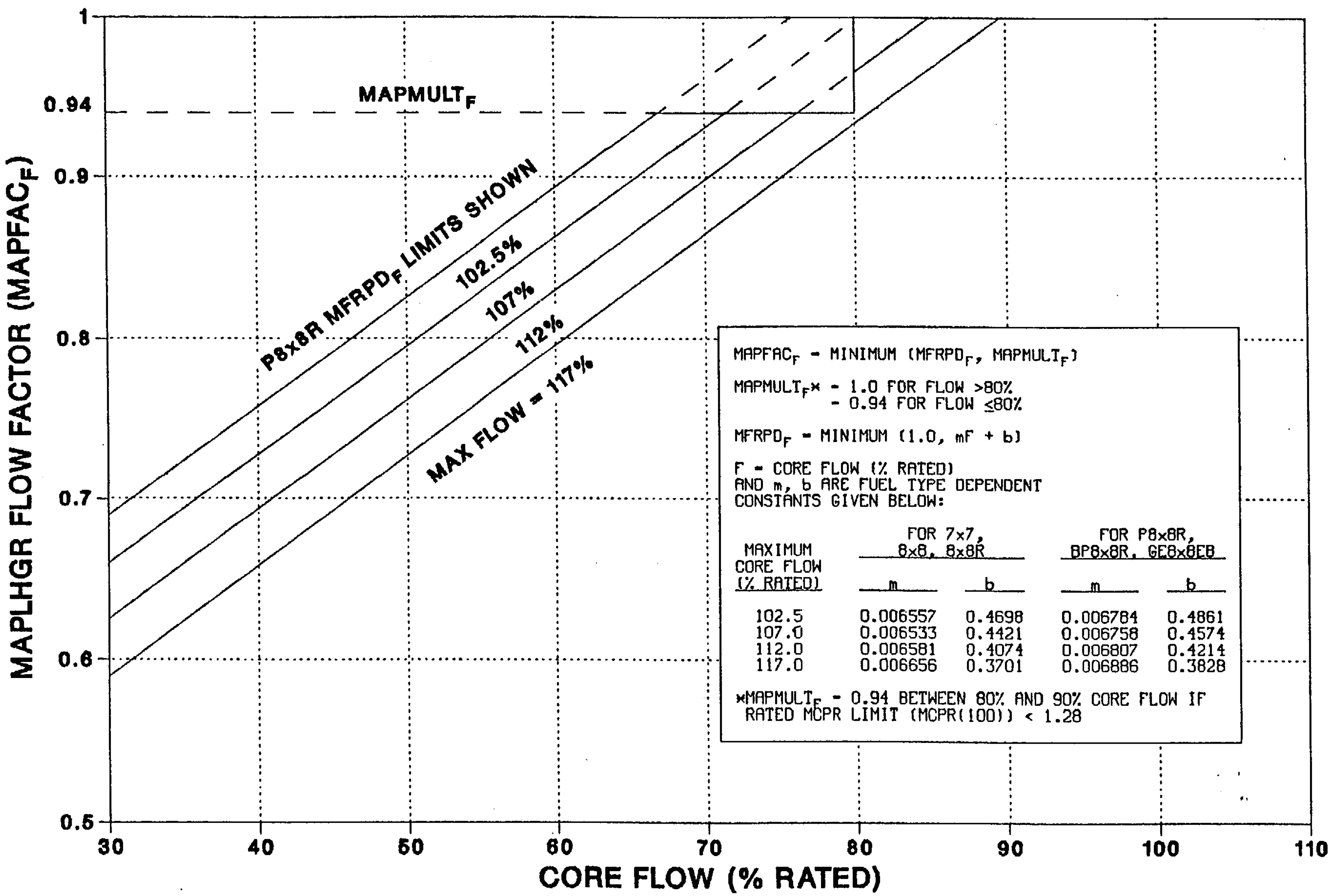
C. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during reactor power operation at ≥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)					
MWD/STU	P8DRB265L BP8DRB265L	P8DRB282 BP8DRB282L	P8DRB284LB BP8DRB284LB	P8DRB299L BP8DRB299L	BD 319B	Other GE 8 Fuel
200	11.6	11.2	11.4	11.0	11.19	10.7
1,000	11.6	11.2	11.4	11.0	11.31	10.8
5,000	11.8	11.8	11.8	11.6	11.99	11.5
10,000	11.9	11.9	11.9	11.9	12.60	12.1
15,000	11.9	11.8	11.9	11.9	12.34	11.8
20,000	11.8	11.7	11.7	11.8	11.95	11.4
25,000	11.3	11.3	11.4	11.5	11.56	11.0
30,000	10.7	11.1	10.8	10.9	10.54	10.0
35,000	10.2	10.4	10.2	10.3	9.53	9.0
40,000	9.6	9.8	9.5	9.7	-	-
45,000	-	-	8.9	9.0	-	-
50,000	-	-	-	-	6.28	5.8

Note: For two recirculation loop operation. For single loop operation multiply these values by 0.85.



MAPFAC$_F$ = MINIMUM (MFRPD$_F$, MAPMULT$_F$)

MAPMULT$_F$* = 1.0 FOR FLOW >80%
= 0.94 FOR FLOW ≤80%

MFRPD$_F$ = MINIMUM (1.0, mF + b)

F = CORE FLOW (% RATED)
AND m, b ARE FUEL TYPE DEPENDENT
CONSTANTS GIVEN BELOW:

MAXIMUM CORE FLOW (% RATED)	FOR 7x7, 8x8, 8x8R m	FOR 7x7, 8x8, 8x8R b	FOR P8x8R, BP8x8R, GE8x8EB m	FOR P8x8R, BP8x8R, GE8x8EB b
102.5	0.006557	0.4698	0.006784	0.4861
107.0	0.006533	0.4421	0.006758	0.4574
112.0	0.006581	0.4074	0.006807	0.4214
117.0	0.006656	0.3701	0.006886	0.3828

*MAPMULT$_F$ = 0.94 BETWEEN 80% AND 90% CORE FLOW IF RATED MCPR LIMIT (MCPR(100)) < 1.28

FIGURE 3.11.2 MAPFAC$_F$ LIMITS

Bases 3.11:

A. Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design bases 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 $\pm 20^{\circ}$ 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 at rated conditions conform to 10CFR50.46. The limiting value for APLHGR is given by this specification.

The flow dependent correction factor (Figure 3.11.2) applied to the rated condition's APLHGR limits assures that 1) the 2200° F PCT limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and 2) the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from less than rated core flow conditions. The power dependent correction factor (Figure 3.11.1) applied to the rated conditions APLHGR limits assures that the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from all conditions (Reference 1).

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

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of 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 rated flow. The Rated

Bases Continued:

MCPR Limit 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.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The MPCRs were calculated such that for the maximum core flow rate and the corresponding thermal power along the 105% of rated power/flow control line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPRs were calculated at different points along the 105% of rated power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow ($MCPR_F$) is defined in Figure 3.11.4 (Reference 1).

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, K_P given in Figure 3.11.3. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are established directly from Figure 3.11.3. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error (Reference 1).

Bases Continued:

This limit was determined based upon bounding analyses for the limiting transient at the given core power level. Further information on MCPR operating limits for off-rated conditions is presented in NEDC-30492-P.(1)

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of requirements by a considerable margin. MCPR evaluation below this power level is therefore unnecessary. The daily requirement for calculating MCPR above 25% of rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

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. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant", NEDC-30492-P, April, 1984.
2. -Deleted-
3. -Deleted-
4. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDE-20566, November, 1975.
5. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWRs", R L Gridley (GE) to D G Eisenhut (USNRC), September 28, 1977.
6. "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.
7. "Monticello Nuclear Generating Plant Single-Loop Operation", NEDO-24271, July, 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.

5.0 DESIGN FEATURES

5.1 Site

A. The reactor center line is located at approximately 850,810 feet North and 2,038,920 feet East as determined on the Minnesota State Grid, South Zone. The nearest site boundary is approximately 1630 feet S 30° W of the reactor center line and the exclusion area is defined by the minimum fenced area shown in FSAR Figure 2.2.2a. Due to the prevailing wind pattern, the direction of maximum integrated dosage is SSE. The southern property line generally follows the northern boundary of the right-of-way for the Burlington Northern Railway. More details on the current property lines can be found in USAR Figure 2.2-1.

5.2 Reactor

A. The reactor core shall consist of not more than 484 fuel assemblies.

B. The reactor core shall contain 121 cruciform-shaped control rods whose design has been reviewed and approved for BWR use by an NRC Safety Evaluation Report. The control rod material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

A. The pressure vessel shall be designed for a pressure of 1250 psig and a temperature of 562°F. The coolant recirculation system shall be designed for a pressure of 1148 psig on suction side of pump and 1248 psig at pump discharge. The applicable design codes shall be as described in Sections 4.2.3 and 4.3.1 of the Monticello Final Safety Analysis Report.

5.4 Containment

A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 134,200 ft^3 and is designed to conform to ASME Boiler and Pressure Vessel Code Section III Class B for an internal pressure of 56 psig at 281°F and an external pressure of 2 psig at 281°F. The absorption chamber shall have a total volume of approximately 176,250 ft^3.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATED TO AMENDMENT NO. 54 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY
MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

1.0 INTRODUCTION

By letter dated July 27, 1987 (Ref. 1) Northern States Power Company (NSP), requested changes to the Technical Specifications to allow operation of the Monticello Nuclear Generating Plant (Monticello) using General Electric-(GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested Technical Specification changes and report (Ref. 2) discussing the reload and analyses done to support and justify Cycle 13 operation. By letters dated August 28 (Ref. 6) and September 3 and 16, 1987, NSP submitted additional information on the proposed Technical Specifications in response to the staff's request for additional information. The August 28 submittal transmitted proprietary information on GE8x8EB fuel designs. In the case of the September 16 submittal, the licensee provided a revised description and safety evaluation supporting changes that were submitted on July 27 and September 3, 1987. This supporting information does not substantially change the action notice or affect the proposed determination of no significant hazards consideration published in the Federal Register on September 23, 1987.

The reload for Cycle 13 is generally a normal reload with no unusual core features and characteristics. Technical Specification changes are few and primarily related to Maximum Average Planar Linear Heat Generation (MAPLHGR) limits for the new fuel, Linear Heat Generating Rate (LHGR) limit for the new fuel, and Minimum Critical Power Ratio (MCPR) limits for all of the fuel using Cycle 13 core and transient parameters and extended operating regions and conditions. The new fuel is the extended burnup type which has been used in several recent BWR reloads (see, for example, Reference 3).

The Cycle 13 reload submittal includes a number of operating flexibility options: single loop operation (SLO), load line limit analysis (LLLA), extended load line limit analysis (ELLLA), and the Average Power Range Monitor (APRM)/Rod Block Monitor/Technical Specification improvement program (ARTS). The effects of these operating flexibility options have been included in the Cycle 13 reload safety analysis.

2.0 EVALUATION

2.1 Reload Description

The Monticello Cycle 13 reload will retain 240 P8x8R and 124 BP8x8R GE fuel assemblies from the previous cycle and add 120 new GE8x8EB fuel assemblies.

The reload safety analysis is based on a previous cycle core nominal average exposure of 21.8 GWd/MTU and Cycle 13 end of cycle core nominal average exposure of 22.3 GWd/MTU. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This loading is acceptable.

2.2 Fuel Design

The new fuel for Cycle 13 is the GE extended burnup fuel GE8x8EB. The fuel designation is BD319B. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR-II (see Refs. 4 and 5). The specific description of this fuel has been submitted in Amendment 18 to GESTAR-II but, since this amendment has not yet been accepted, the fuel description has also been presented for Monticello Cycle 13 in Reference 6. This fuel description is acceptable.

In operation, the GE8x8EB fuel will be assigned a number of axial lattice regions. Appropriate MAPLHGR limits to a maximum value of 45,000 MWD/STU, which have been determined by approved thermal-mechanical and loss-of-coolant accident (LOCA) analyses, will be applied to each of these regions. There was extensive interaction among the staff, GE, and a number of utilities in deciding on an acceptable format for presentation of this information, suitable for plant use and staff requirements for Technical Specifications. Reference 7 provides an example of the Technical Specification for multiple lattice fuel bundles. Reference 16 provides the NSP version of the Technical Specification. The Technical Specification agreed to by the staff, GE and certain utilities presents the least and most limiting lattice MAPLHGR as a function of burnup. However, the process computer used by the licensee contains, and acts on, full details of the MAPLHGR information. When hand calculations of MAPLHGR are required (process computer is inoperative), the most limiting MAPLHGR values as a function of burnup are used as limits for all the lattices of that bundle type. The proposed Technical Specification is acceptable although NSP does not include the least limiting lattice MAPLHGR as a function of burnup in its version of the Technical Specification. A proprietary report (Ref. 6), reviewed by the staff and available to the NSP staff, provides complete details of the lattice definitions and MAPLHGR limits.

The proposed LHGR limit for the GE8x8EB fuel is 14.4 kW/ft rather than the 13.4 kW/ft for other GE fuel. This LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review and meets the criteria for fuel material design set forth in SRP 4.2 (Ref. 4). (See the Reference 9 referrals to References 18 and 19. These references are responses to questions and presentations relating to the GE8x8EB fuel which provide information on the 14.4 kW/ft LHGR). This LHGR is acceptable for the GE8x8EB fuel in Monticello Cycle 13.

Reference 2 states that not all the fuel channels to be used in Cycle 13 were supplied by GE but that GE, at the direction of the licensee, assumed that the performance characteristics of the non-GE fuel channels are identical to the characteristics of the channels supplied by GE. The staff has previously approved the use of non-GE fuel channels for Cycle 11 and these channels have been used at Monticello with no adverse effects. The staff concludes, therefore, that the use of non-GE fuel channels is acceptable.

2.3 Nuclear Design

The nuclear design for Monticello Cycle 13 has been performed by GE with the approved methodology described in GESTAR-II (Ref. 5). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR-II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 1.6% and 1.0% delta K_{eff} at beginning-of-cycle (BOC) and at the exposure of minimum shutdown margin, respectively, thus fully meeting the Technical Specification required amount of 0.25% delta K_{eff}. The Standby Liquid Control System (SLCS) also meets shutdown requirements with a shutdown margin of 4.3% delta K_{eff}. Since these and other Monticello Cycle 13 nuclear design parameters have been obtained with previously approved methods and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for Monticello Cycle 13 has been performed by GE with the approved methodology described in GESTAR-II (Ref. 5) and the results are given in the GE reload report (Ref. 2). The parameters used for the analyses are those approved in Reference 5 for the Monticello class BWR/3 unless otherwise indicated in Reference 2. The GEMINI system of methods (approved in Ref. 9) was used for the relevant transient analyses.

The Operating Limits MCPR (OLMCPR) values are determined by the limiting transients, which are usually the local Rod Withdrawal Error (RWE) and the core-wide transients Feedwater Controller Failure (FWCF), Loss of Feedwater Heating (LFWH) and turbine trip without bypass (TTWOBP). The analyses of the FWCF and TTWOBP events for Monticello Cycle 13 used the standard, approved (Ref. 5) ODYN Options A and B approaches for pressurization transients. However, the RWE is limiting with a minimum CPR of 1.30 (Ref. 10) for Rod Block Monitor (RBM) upscale setpoints of 120, 115 and 110 for the low, intermediate and high trip setpoints, respectively. The Monticello Cycle 13 Technical Specifications will not require OLMCPR's, as a function of average scram time, for operation in both standard and extended operating regions. The Technical Specification OLMCPR is 1.30 with other changes made to remove the dependency on scram speed. Approved methods (Ref. 5) were used to analyze these events and the analyses and results are acceptable and fall within expected ranges.

The results of thermal-hydraulic analyses for two recirculation loop operation show that the maximum core stability decay ratio was 0.63 for Monticello Cycle 11 and Cycle 10. Since Monticello is a BWR/3 with a conventional fuel design, the staff concluded in its Safety Evaluation (SE) that no additional stability analysis was required for Cycle 12. NSP states that the new GE8x8EB fuel has only a small impact on stability performance. The staff agrees with this assessment because of the similarities in the nuclear parameters of the new fuel (e.g., gap conductance, void coefficient) as compared to the previously used fuel. Therefore, Monticello Cycle 13 is typical of previous reload cores which have acceptable stability margin. The staff concludes that Monticello Cycle 13 is acceptable for two recirculation loop operation since it conforms to the staff position of Generic Letter 86-02 (Ref. 11) for BWR/3's.

Thermal-hydraulic stability for single loop operation for Monticello has been addressed and found acceptable in a staff SE for a previous amendment (Ref. 12). Monticello has Technical Specifications which set forth the limiting conditions of operation and surveillance requirements in conformance with the guidance proposed by GE in Service Information Letter (SIL) No. 380, Revision 1 (Ref. 13). Therefore, no additional analyses are required for establishing thermal-hydraulic stability for single loop operation.

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for Monticello Cycle 13 are described in the NRC approved GESTAR-II (Ref. 5). The GEMINI system of methods (Ref. 9) option was used for the transient analyses. The limiting MCPR events for Monticello Cycle 13 are indicated in Section 2.4 above. The core wide transient analysis methodologies and results are acceptable and fall within expected ranges.

The rod withdrawal error is analyzed in the Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvements (ARTS) program topical report (Ref. 14), which has been approved by the staff. A recently approved submittal supports the upscale setpoint changes for a RWE MCPR limit of 1.30 (Ref. 15). The mislocated assembly event was not analyzed for Monticello Cycle 13 since the event is less limiting than for an initial core. This is acceptable since this position was approved by the staff in Reference 5. The misorientation event was analyzed with standard methods for the Monticello Cycle 13 D lattice (non-symmetric water gaps) fuel, giving a non-limiting value of MCPR. The local transient event analyses are thus acceptable.

The limiting pressurization event, the main steam isolation valve (MSIV) closure with flux scram, analyzed with standard GESTAR-II methods, gave results for peak steam dome and vessel pressures well under the limits required by ASME Code Section III for upset conditions (i.e., 110% of design pressure - 1375 psi). These are acceptable methodologies and results.

The licensee's submittal indicates that LOCA analyses, using approved methodologies (SAFE/REFLOOD/CHASTE) and parameters, were performed using MAPLHGR values for the new reload fuel bundles (GE8x8EB). These results were within the limits of 10 CFR 50.46 and are, therefore, acceptable.

Since banked position withdrawal sequence rod patterns are used for Monticello, a cycle specific control rod drop accident analysis is not required. The basis for this position and NRC approval is presented in Amendment 9 to Reference 5.

2.6 Technical Specifications

The Technical Specification (TS) changes for Monticello Cycle 13 are to provide for:

(a) The 14.4 kW/ft LHGR limit for the new (GE8x8EB) fuel.

The change is to TS 3.11.B. The time to initiate corrective action in the ACTION statement has been changed to correspond to Standard Technical Specifications (STS). These changes are acceptable.

(b) MAPLHGR limits for the fuel.

The changes, which were revised, in part, in References 8 and 16, are to TS 3.11.A, Table 3.11.1, and MAPLHGR's for fuel that is no longer used have been deleted. MAPLHGR's for the new fuel and future fuel have been included in Table 3.11.1. The time to initiate corrective action in the ACTION statement has been changed to correspond to STS. These changes are acceptable.

(c) The new MCPR limits for Cycle 13.

Since the proposed TS MCPR limit of 1.30 is higher than the Option A and B limits, all references to MCPR varying as a function of scram time are deleted. The changes are to Table of Contents Item 3.11.2, TS 3.3.C.3, Bases 3.3 and 4.3, TS 3.11.C, Table 3.11.2, and Bases 3.11.C. Bases 3.11.C will now state that the LOCA analyses assumes a MCPR of 1.24, thus correcting an error. The time to initiate corrective action in the ACTION statement has been changed to correspond to STS. All of these changes are acceptable.

(d) Single loop operation surveillance power/flow curve.

Figure 3.5.1 has been redrawn to more clearly define for the operators the permissible operating regimes. This change is acceptable.

(e) Reactor Design Features.

TS 5.2.B has been rewritten in a more general manner so that control rods whose design has been reviewed and approved by the NRC may be used by Monticello. This change is acceptable.

Each of the above changes has been previously discussed and approved in this review except for items (d) and (e), which are acceptable for the reasons stated above.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to 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. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that 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.

4.0 CONCLUSIONS

The staff has reviewed the reports submitted for the Cycle 13 operation of Monticello with extended operating regions. Based on this review, it is concluded that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design and transient and accident analyses are acceptable. The Technical Specification changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations, and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Dan Fieno

Dated: November 25, 1987

REFERENCES

1. Letter and enclosure from David Musolf (NSP) to NRC, dated July 27, 1987. Application requesting changes to the Monticello Technical Specifications for Cycle 13 operation.

2. GE Report 23A5827, Revision 0, dated June 1987, "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Station, Cycle 13."

3. Letter (and enclosure) from R. Clark (NRC) to E. Bauer (PEC), June 1987 (Cycle 8 core reload for Peach Bottom Unit 2).

4. Letter (and attachment) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10."

5. GESTAR-II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

6. Letter and enclosure from David Musolf (NSP) to NRC, dated August 28, 1987. The enclosure dated August 1987 is NEDE-24050-2, Supplement 2, "Supplement 2 to Loss-of-Coolant Accident Analysis for Monticello Nuclear Generating Plant."

7. Letter from J. Charnley (GE) to M. W. Hodges (NRC) dated March 4, 1987, "Recommended MAPLHGR Technical Specifications for Multiple Lattice Fuel Designs."

8. Letter and enclosure from David Musolf (NSP) to NRC, dated September 3, 1987. The enclosure presents a revised wording for the MAPLHGR Technical Specification.

9. Letter (and attachment) from G. Lainas (NRC) to J. Charnley (GE) dated March 22, 1986, "Acceptance for Referencing of Licensing Topical Report, NEDE-24011-P-A, 'GE Generic Licensing Reload Report,' Supplement to Amendment 11."

10. Letter and enclosure from David Musolf (NSP) to NRC, dated February 4, 1987. The enclosure provides RBM setpoints for a CPR of 1.30 for the rod withdrawal error event.

11. Generic Letter No. 86-02, "Technical Resolution of Generic Issue B-19-Thermal-Hydraulic Stability," January 23, 1986.

12. Letter from John Zwolinski (NRC) to David Musolf (NSP) dated October 22, 1986. The letter transmitted the staff SE on Amendment 47 for Single Loop Operation at Monticello.

13. General Electric Service Information Letter No. 380, Revision 1, February 10, 1984.

14. "General Electric BWR Licensing Report: Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for Monticello Nuclear Generating Plant," NEDC-30492-P, April 1984.

15. Letter from Dino Scaletti (NRC) to David Musolf (NSP) dated August 26, 1987. The letter transmitted the staff SE on Amendment 49 for Rod Block Monitor setpoint changes.

16. Letter and enclosure from David Musolf (NSP) to NRC, dated September 16, 1987. The enclosure presents revised wording for the MAPLHGR Technical Specification, the LHGR Technical Specification revision to accommodate the new reload fuel, and other changes.