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. 5
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 February 6, 1981, as supplemented by letter dated March 19, 1981, 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.

8105130324

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 Appendices A and B, as revised through Amendment No. 5 , 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

John N. Hannon for

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 4, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 5

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Remove the following pages and insert identically numbered pages:

26

82

89

90

213

214

216

217

3.0 LIMITING CONDITIONS FOR OPERATION

3.1. REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate a reactor scram.

Obejective:

To assure the operability of the reactor protection system.

Specification:

A. The setpoints, minimum number of trip systems, and minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as given in Table 3.1.1. The time from initiation of any channel trip to the de-energization of the scram pilot valve solenoids shall not exceed 50 milliseconds.

4.0 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the instrumentation that initiates a scram to verify its operability.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1.1 and 4.1.2, respectively.

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. 3. If the cycle average scram insertion time (τ_{AVE}), based on the de-energization of the scram pilot value solenoids at time zero, of all operable control rods in the reactor power operation condition at the 20% inserted position is larger than the adjusted analysis mean scram time (τ_B), a more restrictive MCPR limit (see section 3.11.C.1) shall be used.	
D. Control Rod Accumlators At all reactor operating pressures, 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 inoperabe accumulator.	D. Control Rod Accumulators Once a shift check the status in the control room of the accumulators pressure and level alarms.

consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of 10% of rated power used in the analyses of transients from cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

5. The consequences of a rod block monitor failure have been evaluated. These evaluations show that during reactor operation with certain limiting control rod patterns, the withdrawal of a designated single control rod could result in one or more fuel rods with MCPR's below the Safety Limit (T.S.2.1.A). During use of such patterns, it is judged that testing of the RBM system prior to withdrawal of such rods to assure its operability will assure that improper withdrawal does not occur. It is the responsibility of the Engineer, Nuclear, to identify these limiting patterns and the designated rods either when the patterns are initially established or as they develop due to the occurrence of inoperable rods in other than limiting patterns.

C. Scram Insertion Times

The control rod system is designed to bring the reactor subcritical at a rate fast enough to prevent fuel damage; i.e., to prevent the MCPR from becoming less than the Safety Limit (T.S.2.1.A). This requires the negative reactivity insertion in any local region of the core and in the overall core to be equivalent to at least the scram reactivity curve used in the transient analysis. The required average scram times for three control rods in all two by two arrays and the required average scram times for all control rods are based on inserting this amount of negative reactivity at the specified rate locally and in the overall core. Under these conditions, the thermal limits are never reached during the transients requiring control rod scram. The limiting operational transient is that resulting from a turbine stop valve closure with failure of the turbine bypass system. Analysis of this transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above Specification, provide the required protection, and MCPR remains above the Safety Limit (T.S.2.1.A).

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

Section 3.3.C.3 allows a lower MCPR limit to be used if the cycle average scram time (τ_{AVE}) is less than the adjusted analysis mean scram time (τ_B) (see Reference 7, of Section 3.11)

τ_{AVE} is the weighted cycle average scram time to the 20% insertion position (~ notch 38) of all the operable. rods measured at any point in the cyce.

$$\tau_{AVE} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = the number of surveillance tests performed to date in this cycle.

i = number of control rods measured in the th test.

τ_i = average scram time to the 20% insertion position of all rods measured in the i th test.

τ_B is the adjusted analysis mean scram time to the 20% insertion position.

$$\tau_B = 0.710 + 0.0875 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{\frac{1}{2}}$$

where: N_1 = total number of active rods measured in the first test following core alterations.

0.710 = the mean scram time used in the analysis.

0.0875 = 1.65x0.053 where 1.65 is the appropriate statistical number to provide a 95% confidence level and, 0.053 is the standard deviation of the distribution for average scram insertion time to the 20% position, that was used in the analysis.

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, 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 condition 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.

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.

*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 Radio (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)							
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*							

*For extended burnup program test bundles

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 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.

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_{AVG}) must be determined (see Bases 3.3.C). If τ_{AVG} is below the adjusted analysis scram time, the $MCPR_B$ Limit can be used. If $\tau_{AVG} > \tau_B$, a linear interpolution must be used to determine the appropriate MCPR. For example:

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

$MCPR_A$ 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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 5 TO FACILITY OPERATING LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

MONTICELLO NUCLEAR GENERATING PLANT

DOCKET NO. 50-263

1.0 Introduction

Northern States Power Company (the licensee) has proposed changes to the Technical Specifications of the Monticello facility in Reference 1. The proposed changes relate to the core for Cycle 9 operation at power levels up to 1670 MWt (100% power). In support of the reload application, the licensee has enclosed proposed Technical Specification changes in Reference 1 and the GE BWR supplemental licensing submittal (Reference 2). The licensee has also proposed changing the allowable RPS delay time from 100 milliseconds to 50 milliseconds.

This reload involves loading of prepressurized GE 8x8 retrofit (P8x8R) fuel having drilled lower tieplates. This is the same type of fuel as was loaded during the last reload. The description of the nuclear and mechanical designs of 8x8 retrofit is contained in References 3 and 4. Reference 3 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of fuel. The use and safety implications of prepressurized fuel have been found acceptable per Reference 4. The conclusions of Reference 5 found that the methods of Reference 3 were generally applicable to prepressurized fuel. Therefore, unless otherwise specified, Reference 3, as supported by Reference 5, is adequate justification for the current application of prepressurized fuel.

2.0 Evaluation

2.1 Reactor Physics

The reload application follows the procedure described in NEDE-24011-P, "Generic Reload Fuel Application." We have reviewed this application and the consequent Technical Specification changes. The transient analysis input parameters are typical for BWRs and are acceptable. Core wide transient analysis results are given for the limiting transients and the required operating limit values for MCPR are given for each fuel type. The revised MCPR limits are required by the reload and they are acceptable.

2.2 Thermal Hydraulics

As stated in Reference 3, for BWR cores which reload with GE's retrofit 8x8R fuel, the safety limit minimum critical power ratio (SLMCPR) resulting from either core-wide or localized abnormal operational transients is equal to 1.07. When meeting this SLMCPR during a transient, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition.

To assure that the fuel cladding integrity SLMCPR will not be violated during any abnormal operational transient or fuel misloading, the most limiting events have been reanalyzed for this reload by the licensee, in order to determine which event results in the largest reduction in the minimum critical power ratio. These events have been analyzed for the exposed fuel and fresh fuel. Addition of the largest reductions in critical power ratio to the SLMCPR was used to establish the operating limits for each fuel type.

We have found the methods used for this analysis consistent with previously approved past practice (Reference 3). We have found the results of this analysis and the corresponding Technical Specification changes acceptable.

2.3 ECCS Appendix K

Input data and results for ECCS analysis have been given in References 1 and 2. The information presented fulfills the requirements for each analysis outlined in Reference 3.

We have reviewed the analyses and information submitted for the reload and conclude that the Monticello plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when it is operated in accordance with the Technical Specifications we are issuing with this amendment.

2.4 RPS Delay Time

The licensee has proposed to change the allowed time between a channel sensing a trip condition and the deenergization of the scram pilot valve solenoids from 100 milliseconds to 50 milliseconds. This change will bring the Technical Specifications into agreement with the value assumed in the licensing analysis. The change is in the more conservative direction; the licensee has indicated that the 50 millisecond RPS delay time has been demonstrated to be attainable in previous tests. We find the licensee's proposed change acceptable.

3.0 Environmental Considerations

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR Section 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations 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.

Dated: May 4, 1981

REFERENCES

1. (a) Letter, NSP to Office of Nuclear Reactor Regulation (USNRC), dated February 9, 1981

   (b) Letter, NSP to Office of Nuclear Reactor Regulation (USNRC), dated March 19, 1981.

2. "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Plant, Reload 8 (Cycle)", dated March 1981.

3. "General Electric Boiling Water Reactor Generic Reload Application", NEDE-24011-P-A, May 1977.

4. Letter, R. E. Engel (GE) to U. S. Nuclear Regulatory Commission, dated January 30, 1979.

5. Letter, T. A. Ippolito (USNRC) to R. Gridley (GE), April 16, 1979, and enclosed SER.