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December 6, 1982

Docket No. 50-263

Mr. D. M. Musolf
Nuclear Support Services Department
Northern States, Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Musolf:

The Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment authorizes changes to the Technical Specifications in response to your June 25, 1982 application, supplemented by letters dated August 3, 1982 and August 24, 1982 and subsequent discussions between the NRC staff and your staff. During these conversations, changes to the Technical Specifications were discussed with and agreed to by your staff.

The amendment changes the Technical Specifications to incorporate revised safety and operating limits associated with the operation of Monticello during the upcoming fuel Cycle 10.

Copies of the Safety Evaluation and Notice of Issuance are enclosed.

Sincerely,
Original signed by

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 13 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

cc; w/enclosures
See next page

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F.R. NOTICE & AMENDMENT

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DATE	12/5/82	12/2/82	12/3/82	12/2/82	12/03/82	11/ /82	

NRC FORM 318 (10-80) NRCM 0240 OFFICIAL RECORD COPY USGPO: 1981—335-960

Mr. D. M. Musolf
Northern States Power Company

cc:

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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. 13
License No. DPR-22

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

   A. This application for amendment by Northern States Power Company (the licensee) dated June 25, 1982 and supplements dated August 3, and 24th, 1982 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 Appendices A and B as revised through Amendment No. 13 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

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 13, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 13

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Remove the following pages and insert identically numbered pages:

INSERT

82
212
213
214
215
216

3.0 LIMITING CONDITIONS FOR OPERATION

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_{AVG}), based on the de-energization of the scram pilot valve 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) shall be used.

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.

4.0 SURVEILLANCE REQUIREMENTS

D. Control Rod Accumulators

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

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 (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

B. Linear Heat Generation Rate (LHGR)

During power operation, the LHGR shall be limited to:

LHGR $\leq$ 13.4 kw/ft

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 two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.

B. Linear Heat Generation Rate (LHGR)

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

C. Minimum Critical Power Ratio (MCPR)

During power operation the Operating MCPR Limit shall be ≥ 1.36 for 8x8, ≥ 1.37 8x8R fuel, ≥ 1.39 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.

*If $\tau_{ave} > \tau_{B}$, the operating MCPR Limit shall be a linear interpolation between the limits in 3.11.C and 1.41 for 8x8, 1.42 for 8x8R fuel and 1.44 for P8x8R fuel.

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)							
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	10.5	10.6	10.2	10.7	10.7	11.1	11.1	10.8
35,000	9.8	9.6	9.7	10.2	10.2	10.4	10.4	10.2
40,000	8.9	9.0	9.1	9.6	9.6	9.8	9.8	9.5

Bases 3.11

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 a automatic reactor scram are not considered a violation of the LCO. Exceeding APLHGR 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 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 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_{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.

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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 13 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 June 25, 1982 (Reference 1), the Northern States Power Company (the licensee) proposed changes to the Technical Specifications of Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. To support the reload application, the licensee attached a supplemental licensing submittal (Reference 2) prepared by General Electric that summarized the results of the core performance analysis for Cycle 10. By letters dated August 3, and 24, 1982 (References 3 and 4), the licensee submitted additional information relevant to this evaluation. The licensee proposes a revision to the Technical Specifications to incorporate revised limiting conditions for operation: 1) as a result of the analysis for Cycle 10 operation, 2) removal of fission gas release restrictions on MAPLHGR limits; and 3) removal of the power spiking penalty. During subsequent discussions with the licensee's staff, changes to the Technical Specifications were discussed with and agreed to by the licensee's staff.

2.0 EVALUATION

2.1 Generic Reload Analysis

Generic information associated with the reload analysis of BWR fuel is presented in the General Electric (GE) Licensing Topical Report, "General Electric Boiling Water Reactor Generic Reload Fuel Applications," (Reference 5). This generic topical report has been reviewed and approved by the staff (see Reference 6). To supplement the generic information with the information associated with Monticello's Cycle 10 operation, the licensee submitted Reference 2. Since we have previously reviewed a large number of generic considerations associated with this type of core, and on the basis of the evaluations presented in References 5 and 6, we conclude that additional staff review of those portions of Reference 5 concerning the standard fuel design is unnecessary for Cycle 10 operation. Only a limited number of areas need to be addressed in this Safety Evaluation. For those areas not addressed, the reader is referred to References 5 and 6.

2.2 Transient Analysis

2.2.1 Safety Limit Minimum Critical Power Ratio

The safety limit minimum critical power ratio (MCPR) has been imposed to assure that 99.9 percent of the fuel rods in the core do not experience boiling transition during normal operation or worst anticipated operational transient. As stated in Reference 5, for BWR cores with GE retrofit 8x8R fuel, the safety limit MCPR is 1.07. There has been no change in the safety limit MCPR for Monticello Nuclear Generating Plant from Cycle 9 to Cycle 10.

2.2.2 Operating Limit Minimum Critical Power Ratio

Various transients could reduce the MCPR below the intended safety limit MCPR during Cycle 10 operation. The most limiting events have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial critical power ratio (delta CPR). The transient events analyzed for Cycle 10 included the reactor coolant system pressurization (load rejection without bypass and feedwater controller failure), feedwater temperature reduction (loss of 100°F feedwater heating) and local reactivity insertion (control rod withdrawal error). These events have been analyzed for exposed and fresh fuel. In the above categories the licensee reports load rejection without bypass as the most limiting event for exposed unpressurized 8x8, 8x8R, and prepressurized 8x8R fuel assemblies.

The delta CPR values given in Section 11 of Reference 2 are plant-specific values that are calculated by using the ODYN methods. The maximum delta CPRs for the non-pressurized and pressurized fuels (8x8, 8x8R and P8x8R) for Cycle 10 are 0.28, 0.29 and 0.31 as compared to 0.35, 0.35 and 0.39 for Cycle 9 (the previous cycle, see Reference 7). The large difference of delta CPR is due to a faster effective scram insertion rate (and thus, a lower peak heat flux during the transient) for Cycle 10 as compared to Cycle 9. The calculated delta CPRs were adjusted to reflect either Option A or Option B delta CPRs by employing the conversion method described in Reference 8. The initial MCPR for the transient is determined by adding the delta CPR to the safety limit MCPR. Section 11 of Reference 2 presents both the MCPRs for the non-pressurization events and the adjusted MCPRs (Option A and Option B) for pressurization events. Section 3.11.C of Reference 1 (Exhibit B) shows the operating limit MCPR for 8x8, 8x8R, and P8x8R types of fuel.

We have reviewed the operating limit MCPRs results discussed above. We find that the approved methods were used and the results are consistent with the previous Cycle 9 analysis. Therefore, we conclude that these results are acceptable.

The licensee has proposed changes to the Technical Specifications, Section 3.11.C to include the operating limit MCPR for Cycle 10 operation. Using the linear interpolation method specified on page 216 of the Technical Specifications, the operating limit MCPRs shall be a value between 1.36 and 1.41 for 8x8 fuel, 1.37 and 1.42 for 8x8R fuel, and 1.39 and 1.44 for P8x8R fuel.

2.2.3 Corrective Action for Reload MCPR Errors

General Electric identified and notified the staff of a programming error in the axial power distribution calculations used in the ODYN transient computer code. In Reference 3, the licensee states that the corrected version was used for the reload analysis.

On June 8, 1982 the staff was notified by General Electric of a generic fuel length error. In Reference 4, the licensee states that this error affected the Monticello analysis in the conservative direction. Each of these changes (more bottom peaked power shape and corrected fuel length) resulted in an improved scram reactivity as used in ODYN for Cycle 10. We have reviewed the submittal that has accounted for these errors and since the analysis is in the conservative direction, we find the corrective actions to be acceptable.

2.2.4 Reactor Vessel Overpressure Protection

Closure of the main steam isolation valves (MSIV) is the limiting event for vessel overpressurization. For Cycle 10, the licensee analyzed MSIV closure and verified that the ASME Boiler and Pressure Vessel Code requirements will continue to be met. The methods used for this analysis, when modified to account for one failed safety valve, have been approved by the staff (Reference 6). For this event, the calculated peak transient pressure must not exceed 110% design pressure, or 1375 psig. For worst case End-of-Cycle 10 conditions, the peak pressure at the bottom of the vessel was predicted not to exceed 1222 psig; even when assuming the effects of one failed safety valve. Since this value falls below the peak allowable ASME overpressure of 1375 psig, we find the protection provided for the reactor vessel overpressure to be acceptable.

2.2.5 Feedwater Controller Failure During Maximum Demand-Transient Analysis

In analyzing anticipated operational transients, the licensee has taken credit for plant operating equipment which is not normally reviewed by the staff because this equipment is not considered essential to safety. On a generic basis, we have discussed, with General Electric, the application of this equipment. Based on these discussions, it is our understanding that the most limiting transient, aside from generator trip without bypass, that takes credit for this equipment is the excess feedwater event.

This event postulates the failure of the feedwater controller during maximum demand; thereby, directly causing an increase in coolant inventory. The influx of excess feedwater flow results in an increase in core subcooling which reduces the void fraction and thus induces an increase in reactor power. To mitigate the consequences of this transient, the high water level, main turbine and feedwater systems will trip and the turbine bypass valves will actuate.

To assure an acceptable level of performance, our position is that this equipment (the turbine bypass system and the level 8 high water level trip) should be identified in the Technical Specifications with limiting conditions for operation, surveillance requirements and with adequate degree of power reduction, in case of inoperability. The licensee has agreed to submit, within 60 days, proposed changes to the Technical Specifications that will incorporate the LCO and surveillance requirements for the turbine bypass and high water level trip systems. We find this acceptable, since we only recently informed the licensee of our position.

2.3 Thermal Hydraulic Stability

The results of the thermal-hydraulic analysis (Reference 2) show that the maximum reactor core stability ratio is 0.62 for Cycle 10 as compared to 0.63 for Cycle 9. Based on the evaluation results that (1) the calculated decay ratio for Cycle 10 is less than that of Cycle 9, and (2) the decay ratio compares favorably to the calculated value for several operating reactors that have been previously approved, we conclude that the thermal-hydraulic stability results are acceptable for Cycle 10.

2.4 Maximum Average Planar Linear Heat Generation Rate Limits

The licensee has submitted revised Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for each type of fuel that will be used during Cycle 10 operation. These limits were generated by methods (Reference 9) submitted as part of this application. Although the methodology is generally applicable for these limits, we believe that the effects of enhanced fission gas release in high burnup fuel (above 20,000 MWd/MTU) were not adequately considered in the generic analysis. In response to this concern, GE requested (References 10 and 11) that credit for approved, but unapplied, emergency core cooling system (ECCS) evaluation model changes be used to avoid MAPLHGR penalties at higher burnup. We found this proposal acceptable (Reference 12) provided that certain plant-specific conditions were met. In a letter dated August 24, 1982 (Reference 4), the licensee found the GE proposal applicable to the Monticello Nuclear Generating Plant.

Therefore, since the licensee has endorsed the GE position with respect to enhanced fission gas release in high burnup fuel (which we found acceptable), we conclude that the MAPLHGR limits (proposed on page 214) for Monticello, Cycle 10 are acceptable.

2.5 Linear Heat Generation Rate

Fuel densification affects fuel rods by increasing the stored energy, linear thermal output, and probability of local power spikes from axial gaps. The power spiking penalty, by assuming a linearly increasing variation and axial gaps between core bottom and top, assures with a 95% confidence that no more than one fuel rod exceeds the design linear heat generation rate. The licensee has proposed changes to the Technical Specifications (pages 212 and 215) that delete the height dependent power spiking penalty from the linear heat generation rate (LHGR) limiting condition for operation. Instead, the power spiking penalty is included in the transient analysis, i.e., this factor is added to the results of the analysis before being compared to the limit. We have reviewed this method on a generic basis and have found it acceptable. We have reviewed the licensee's proposed changes (pages 212 and 215) to the Technical Specifications to delete the core height dependency on LHGR, and find them acceptable based on the discussion above.

In the rod withdrawal error analysis, a fully inserted high worth rod is assumed to be withdrawn continuously. At the time of withdrawal, an assembly near the withdrawn rod is assumed to be operating at the Technical Specifications limit. The response of the Rod Block Monitor is then calculated as a function of the distance the rod is withdrawn. When the rod block setpoint is reached, the rod is assumed to travel an additional two inches and then stop at the next notch. In Reference 2, the licensee used the generic rod withdrawal analysis, with a Rod Block Monitor setting of 108. At this setting, the rod withdrawal error is not a limiting transient. We have reviewed the generic analysis and approved it in the interim while receipt of additional information is pending from General Electric. Since we have accepted the generic analysis used by the licensee, we find the licensee's analysis to be acceptable.

In the generic analysis of misloaded fuel events, two types of fuel misloadings are analyzed; misorienting an assembly in its proper location and mislocating a properly oriented bundle. In the misoriented (former) event, the bundle may be rotated by 90° or 180° from its normal orientation.The presence of higher enrichment rods near the wide water gap increase the linear heat generation rate. The analysis of the mislocated bundle (latter) event, uses procedures that substitute higher enrichment bundles for various high burnup bundles throughout the core. A generic analysis of the mislocated bundle event has shown that this event is never limiting. A cycle specific misoriented bundle analysis (without credit for the bundle tilt) has shown that this event is not limiting for Monticello during the Cycle 10 operation and therefore, find the analysis to be acceptable. Since we have accepted GE's analysis that the fuel loading error is no longer expected to be limiting, then it is acceptable to delete TS (Section 3.11.C.2 and revise the associated bases on page 216) for adjusting operating MCPR limits according to high air ejector activity.

3.0 SUMMARY

We have reviewed the analyses used by the licensee for Cycle 10 operation and have found them acceptable. Based on the results of these analyses, the licensee has proposed changes to the Technical Specifications. We have reviewed the proposed changes to the Technical Specifications, that incorporate revised safety and operating limits for the operation of Monticello during the upcoming Cycle 10, and have found them acceptable.

4.0 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 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.

5.0 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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: December 6, 1982

Principal Contributors: Helen Nicolaras
Summer Sun
John Voglewede
Walter Brooks
Michael McCoy

REFERENCES

1. Letter, D. M. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (USNRC), June 25, 1982 (with attachments).

2. "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Plant, Reload 9 (Cycle 10)," Y1003J01A39, May 1982.

3. Letter, D. M. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (USNRC), August 3, 1982.

4. Letter, D. M. Musolf (NSP) to Director, Office of Nuclear Reactor Regulation (USNRC), August 24, 1982.

5. "General Electric Boiling Water Reactor Generic Reload Fuel Applications," General Electric Company Report NEDE-24011-P-A-4 (Proprietary) and NEDO-24011-A-4 (Non-Proprietary), January 1982.

6. Letter, D. G. Eisenhut (USNRC) to R.Gridley (GE), May 12, 1978.

7. "Supplemental Reload Licensing Submittal for Monticello Nuclear Generating Plant Reload 8 (Revision 1)," Y1003J01A16, March 1981.

8. Letter, R. Buchholz (GE) to P. Check (USNRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980.

9. "Loss-of-Coolant Accident Analysis Report for Monticello Nuclear Generating Plant," General Electric Company Report NEDO-24050-1, December 1980.

10. Letter, R. E. Engel (GE) to T. A. Ippolito (USNRC) dated May 6, 1981.

11. Letter, R. E. Engel (GE) to T. A. Ippolito (USNRC) dated May 28, 1981.

12. Memorandum, L. S. Rubenstein (USNRC) to T. M. Novak (USNRC), "Extension of General Electric Emergency Core Cooling Systems Performance Limits," June 25, 1981.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY

OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised the Technical Specifications for operation of the Monticello Nuclear Generating Plant (the facility) located in Wright County, Minnesota. The amendment is effective as of its date of issuance.

The amendment changes the Technical Specifications to establish revised safety and operating limits for operation of the facility during fuel Cycle 10.

The application for amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of the amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated June 25, 1982 as supplemented by letters dated August 3 and 24, 1982, (2) Amendment No. 13 to License No. DPR-22, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Environmental Conservation Library, Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 6th day of December 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

DBVassallo

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing