October 31, 1984

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. 27 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 application dated May 29, 1984, supplemented by clarifying information provided in your letter dated August 16, 1984.

The amendment changes the Technical Specifications to add surveillance and operability requirements for the residual heat removal (RHR) intertie line valves and add limitations on use of the intertie line. This amendment also deletes Technical Specifications pertaining to the recirculation system crosstie lines, which have been removed.

A copy of the Safety Evaluation is enclosed.

Sincerely,

Original signed by/

Vernon L. Rooney, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 27 to
   License No. DPR-22
2. Safety Evaluation

cc w/enclosures:
See next page

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Mr. D. M. Musolf
Northern States Power Company
Monticello Nuclear Generating Plant

cc:

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and
Trowbridge
1800 M Street, N. W.
Washington, D. C. 20036

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Box 1200
Monticello, Minnesota 55362

Plant Manager
Monticello Nuclear Generating Plant
Northern States Power Company
Monticello, Minnesota 55362

Russell J. Hatling
Minnesota Environmental Control
Citizens Association (MECCA)
Energy Task Force
144 Melbourne Avenue, S. E.
Minneapolis, Minnesota 55113

Executive Director
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Mr. Steve Gadler
2120 Carter Avenue
St. Paul, Minnesota 55108

John W. Ferman, Ph.D.
Nuclear Engineer
Minnesota Pollution Control Agency
1935 W. County Road B2
Roseville, Minnesota 55113

Commissioner of Health
Minnesota Department of Health
717 Delaware Street, S. E.
Minneapolis, Minnesota 55440

Auditor
Wright County Board of
Commissioners
Buffalo, Minnesota 55313

U. S. Environmental Protection
Agency
Region V Office
Regional Radiation Representative
230 South Dearborn Street
Chicago, Illinois 60604

James G. Keppler
Regional Administrator
Region III Office
U. S. Nuclear Regulatory Commission
799 Roosevelt Road
Glen Ellyn, Illinois 60137

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. 27
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 May 29, 1984, as supplemented August 16, 1984, 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. 27, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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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: October 31, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 27

FACILITY OPERATING LICENSE NO. DPR-22

DOCKET NO. 50-263

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
17	17
103	103
104	104
---	105a
106	106
107	107
114	114
---	116a
119	119

backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

B. APRM Control Rod Block Trips Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculate flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit operation is permissible only during the succeeding seven days unless at least one of such systems is sooner made operable, provided that during such seven days all active components of the LPCI mode of RHR system and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

4. Each core spray system shall be capable of delivering 3,020 gpm against a reactor pressure of 130 psig. If this rate of delivery requirement cannot be met, the system shall be considered inoperable.

5. If the requirements of 3.5.A.1-3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

diesel generators required for operation of such components (if no external source of power were available) shall be demonstrated to be operable immediately and daily thereafter.

B. Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System)

1. Except as specified in 3.5.B.2 and 3.5.B.3 below, the LPCI shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the LPCI pumps or admission valves is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump or admission valve is sooner made operable, provided that during such thirty days the remaining active components of the LPCI and containment cooling subsystem and all active components of both core spray systems and the diesel generators required for operation of such components (if no external source of power were available) shall be operable.

B. Surveillance of the Low Pressure Coolant Injection (LPCI) Subsystem (LPCI Mode of RHR System) shall be performed as follows:

1. Testing

Item	Frequency
Pump Operability	Once/month
Motor operated valve operability	Once/month
Cycling of RHR Intertie Line Valves	Once/Quarter
Flow rate test (recirculate to torus)	After major pump maintenance and every three months
Simulated automatic actuation test	Every refueling outage

2. When it is determined that one of the LPCI pumps is inoperable, the remaining active components of the LPCI and containment cooling subsystem, both core spray systems and the diesel generators required for operation of such components (if no external source of power were available) shall be demonstrated to be operable immediately and the operable LPCI pumps daily thereafter.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
6. Both RHR Intertie return line isolation valves shall be operable. To be considered operable, each valve must be capable of automatic closure on a LPCI initiation signal or be in the closed position. If one valve is made or found to be inoperable for any reason, the other return line isolation valve and the RHR suction line isolation valve shall be closed, otherwise the actions specified in 3.5.B.3 shall be taken. 7. Flow shall not be established in the RHR intertie line with the reactor in the Run Mode. 8. If the requirements of 3.5.B.1 through 3.5.B.4 cannot be met, an orderly shutdown of the reactor shall be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.	

Containment Cooling Capability

C. Residual Heat Removal (RHR) Service Water System

1. Except as specified in 3.5.C.2 and 3.5.C.3 below, both RHR service water system loops shall be operable whenever irradiated fuel is in the reactor vessel and reactor coolant temperature is greater than 212°F.

2. From and after the date that one of the RHR service water system pumps is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding thirty days unless such pump is sooner made operable, provided that during such thirty days all other active components of the RHR service water system are operable.

Containment Cooling Capability

C. Surveillance of the RHR service water system shall be performed as follows:

1. Testing

Item	Frequency
Pump and valve operability	Once/3 months
Flow rate test	After major pump maintenance and every three months

2. When it is determined that one RHR service water pump is inoperable, the redundant components of the remaining subsystem shall be demonstrated to be operable immediately and daily thereafter.

3. From and after the date that one of the RHR service water systems is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such system is sooner made operable, provided that during such seven days all active components of the operable RHR service water system shall be demonstrated to be operable at least once each day.

4. To be considered operable, a RHR service water pump shall be capable of delivering 3500 gpm against a head of 500 feet.

5. If the requirements of 3.5.C.1-3 cannot be met, an orderly shutdown of the reactor will be initiated and the reactor water temperature shall be reduced to less than 212°F within 24 hours.

3. When one RHR service water system becomes inoperable, the operable system shall be demonstrated to be operable immediately and daily thereafter.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
I. Recirculation System 1. Reactor operation with one loop shall be limited to 24 hours.	I. Recirculation System See Specification 4.6.G.

Bases Continued:

An intertie line is provided to connect the RHR suction line with the two RHR loop return lines. This four-inch line is equipped with three isolation valves. The RHR loop return line isolation valves receive a closure signal on LPCI initiation. In the event of an inoperable return line isolation valve, there is a potential for some of the LPCI flow to be diverted to the broken loop during a loss of coolant accident. Surveillance requirements have been established to periodically cycle the RHR intertie line isolation valves. In the event of an inoperable RHR loop return line isolation valve, the other two isolation valves are closed to prevent diversion of LPCI flow.

The RHR intertie line is not used when the reactor is in the Run Mode to eliminate the need to compensate for the small change in jet pump drive flow or for a potential reduction in core flow during a loss of coolant accident.

G. Emergency Cooling Availability

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

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

H. Deleted

I. Recirculation System

Extended operation with one reactor recirculation loop inoperable is prohibited until the NRC has completed an evaluation of single loop operation.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 27 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 May 29, 1984 Northern States Power Company (the licensee) proposed to change the Technical Specifications for the Monticello Nuclear Generating Plant to add surveillance and operability requirements for intertie line valves, add limitations on the use of the intertie line, and delete Technical Specifications (TSs) pertaining to the crosstie line. These Technical Specification changes stem from changes in piping configuration during the current pipe replacement outage, in which an intertie line has been added and the crosstie line has been removed. Additional clarifying information was presented by the licensee by letter dated August 16, 1984. According to the licensee there have been incidents of waterhammer in the recirculation system when the system was operating during shutdown cooling. The General Electric Company conducted an analysis to determine the best method of preventing waterhammer and providing for warmup of an idle recirculation loop. The analysis indicated that a residual heat removal (RHR) intertie line as shown in Figure-1, was the best alternative. A four-inch intertie line was shown to provide sufficient flow to minimize collection of steam bubbles in the high points of the recirculation system piping and RHR suction and discharge pipes as long as the flow is circulated continuously during depressurization. This is a design enhancement intended to minimize the potential for waterhammer in the circulation system. Three motor operated valves, as shown in Figure-1, will be installed for isolation purposes. The two isolation valves which are normally closed during normal plant operation will receive a closure signal on low pressure coolant injection (LPCI) actuation to prevent a reduction in LPCI flow delivered to the reactor under accident conditions.

The new intertie line would be normally isolated during plant operation with MO-4085A and MO-4085B closed and MO-4086 open. The line would be used:

1. When the plant is proceeding to cold shutdown, the valves would be opened after the reactor is out of the run mode and prior to depressurizing the reactor.

2. To prevent a recirculation loop waterhammer in a hot isolated loop during primary system depressurization. In this case, MO-4985A and MO-4085B would be opened and MO-4086 would be either open or closed, depending on which loop is isolated.

2.0 EVALUATION

2.1 RHR Intertie Line

The RHR intertie line has been evaluated for potential adverse effects on plant safety. The evaluation, performed by the General Electric Company, is described in NEDO-30477, Rev. 1, June 1984.

The following potential adverse effects were identified:

1. An impact of the 10 CFR Part 50, Appendix K, analysis due to an increase in design basis accident (DBA) break area equal to a four-inch line (0.08 square feet).

2. An additional flow path between the broken and unbroken recirculation loops affects core flow and may cause early boiling transition during a loss of coolant accident (LOCA).

3. An increase in containment peak pressure and temperature due to the larger DBA break size.

4. An increase in containment suppression pool loads (Mark I Long Term Program considerations) due to the larger DBA break size.

5. When the intertie line is in use, measured recirculation drive flow will be slightly greater than drive flow delivered to the jet pumps. Flow biased scrams and rod blocks may be affected.

Since the intertie valves are assumed to be open at the time of a postulated accident, the effect of the intertie on the DBA break size was evaluated. NRC-approved evaluation models were used in the analysis. Assumptions and models used in the latest Monticello LOCA analysis (Ref. 1) were also used in the new analysis.

The intertie is a four-inch line with a flow area of 0.08 ft^2. If this flow area is added to the former maximum flow area, the new 100% DBA break size becomes 4.09 ft^2. The addition of the intertie flow area increases the maximum flow area by about 2%. The analysis performed for the new maximum flow area indicates that the difference in the time of core uncovery is less than one second. There is no change in the limiting break size (1.36 ft^2) with or without the intertie. There is no significant impact on the Appendix K analysis.

The open intertie line will split off some flow from the unbroken recirculation loop during a LOCA. Since use of the intertie line is prohibited during the run mode the reduced flow conditions is not a safety concern.

The licensee's analysis indicates that the effects of the increased DBA break size of 4.09 ft^2 on the peak containment pressure and temperature are slight increases of 0.9 psig and 1°F, respectively. The licensee's analysis also showed that the pool swell loads on the torus shell and internals would increase by less than 1% due to the increase in the drywell pressure.

The licensee stated that the new analyses are consistent with the previously established method in the Plant Unique Load Definition Report (PULDR). The PULDR methodology was reviewed by the staff as part of the Mark I long term program. We concluded that the review of the Monticello program has been completed with no outstanding issues.

The recent revisions were also performed in accordance with NUREG-0661. Therefore, we find the methodology acceptable.

Due to the inherent conservatism associated with the acceptance criteria reported in NUREG-0661, we find that a 1% increase in the pool swell load will not affect the capability of the Monticello containment to accommodate the DBA LOCA loads.

With respect to the adequacy of the containment design temperature and pressure, we find that the revised increases in the calculated maximum temperature and pressure values are small. They remain well below the design values. Therefore, we find the analyses acceptable.

Technical Specifications prohibit the use of intertie line with the reactor in the run mode. Therefore, since the flow-biased scram and rod blocks are not required to be operable in shutdown mode, there is no need to adjust the flow-biased scram and rod blocks.

The following Limiting Conditions and Surveillance Requirements are proposed for additions to the Technical Specifications:

1. Quarterly cycling of MO-4085A, MO-4085B and MO-4086. In addition, these valves will be included in the Monticello ASME Code, Section XI, Pump and Valve Testing Program.

2. Requirement for operability of MO-4085A and MO-4085B at all times. In the event of an inoperable valve, actions are specified to assure full LPCI flow.

3. Limitations on use of the intertie line. Flow will not be allowed in the line if the reactor is in the run mode.

The limiting conditions and surveillance requirements proposed by the licensee are acceptable to the staff.

2.2 Recirculation System Crosstie Line

The licensee amendment request also revises the Technical Specifications to reflect the removal of the recirculation loop crosstie line. The recirculation system crosstie line and associated valve are being removed as part of the recirculation piping replacement program. Because of emergency core cooling system (ECCS) break area considerations, use of this line was prohibited by the Technical Specifications. The requested changes delete restrictions related to this line.

The proposed change is acceptable to the staff.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes in surveillance requirements. 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

We have 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.

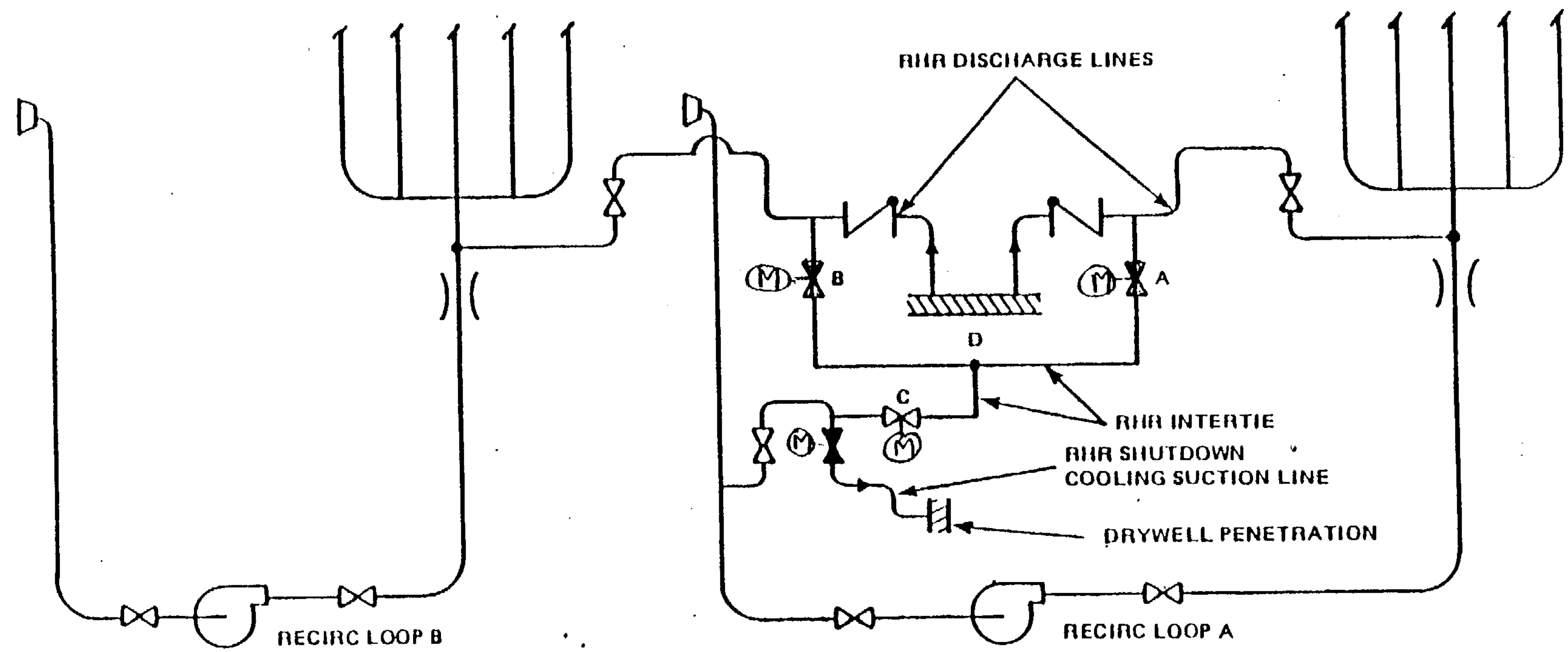
5.0 REFERENCES

1. "LOCA Analysis Report for Monticello Nuclear Generating Plant," General Electric Company, NEDO-24050, Rev. 1, December 1980.

2. "Mark I Containment Program Plant Unique Load Definition, Monticello Nuclear Generating Plant," General Electric Company, NEDO-24576-1, October 1981.

Principal Contributor: G. Thomas

Dated: October 31, 1984



VALVE NUMBERS

A: MO – 4085A – NORMALLY CLOSED – CLOSE ON LOCA SIGNAL IF OPEN
B: MO – 4085B – NORMALLY CLOSED – CLOSE ON LOCA SIGNAL IF OPEN
C: MO – 4086 - NORMALLY OPEN

Figure 1 RHR Intertie Modification