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Docket No. 263

April 18, 1983

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. 17 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The amendment consists of changes to the Technical Specifications in response to your September 24, 1982 application.

The revisions to the Technical Specifications include the following:

1. Title change from AEC to Commission;
2. Correction of Table numbering;
3. Clarification of definition for No;
4. Clarification of the bases section to reflect the removal of two vacuum breakers;
5. Identification of fire detectors that have been installed;
6. Correction of inconsistency between the FSAR and the Technical Specifications on the reactor vessel construction codes and standards;
7. Change from FSAR to USAR as the report to be reviewed by the Operations Committee; and
8. Correction of typographical errors.

Other changes requested in the September 24, 1982 submittal are still under staff review and will be addressed by separate Safety Evaluation and license amendment.

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

Sincerely,

ORIGINAL SIGNED BY

Helen Nicolaras, Project Manager
Operating Reactors Branch #2
Division of Licensing

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Chief M+EB
~~DE:AD:M&QE~~
~~WJohnston~~ BDLiaw
4/5/83

Enclosures:
1. Amendment No. 17 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures
See next page

OFFICE			DL:ORB#2	DL:ORB#2	DL:ORB#2	DL:OR	OELD
SURNAME			S.Norris	H.Nicolaras	D.Vassallo	G.Lainas	
DATE			2/8/83	3/23/83	4/5/83	4/6/83	2/8/83

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. 17
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 September 24, 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 Appendix A as revised through Amendment No. 17 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

D B Vassallo

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 17, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 17

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
4	4
90	90
126	126
150	150
179	179
180	180
227c	227c
230	230
241	241
253a	253a

4. Protective Function - A system protective action which results from the protective action of the channels monitoring a particular plant condition.

R. Rated Neutron Flux - Rated flux is the neutron flux that corresponds to a steady-state power level of 1670 thermal megawatts.

S. Rated Thermal Power - Rated thermal power means a steady-state power level of 1670 thermal megawatts.

T. Reactor Coolant System Pressure or Reactor Vessel Pressure - Unless otherwise indicated, reactor vessel pressures listed in the Technical Specifications are those existing in the vessel steam space.

U. Refueling Operation and Refueling Outage - Refueling Operation is any operation when the reactor water temperature is less than 212°F and movement of fuel or core components is in progress. For the purpose of designating frequency of testing and surveillance, a refueling outage shall mean a regularly scheduled refueling outage; however, where such outages occur within 8 months of the completion of the previous refueling outage, the required surveillance testing need not be performed until the next regularly scheduled outage.

V. Safety Limit - The safety limits are limits below which the maintenance of the cladding and primary system integrity are assured. Exceeding such a limit is cause for plant shutdown and review by the Commission before resumption of plant operation. Operation beyond such a limit may not in itself result in serious consequences but it indicates an operational deficiency subject to regulatory review.

W. Secondary Containment Integrity - Secondary Containment Integrity means that the reactor building is closed and the following conditions are met:

1. At least one door in each access opening is closed.
2. The standby gas treatment system is operable.
3. All reactor building ventilation system automatic isolation valves are operable or are secured in the closed position.

X. Sensor Check - A qualitative determination of operability by observation of sensor behavior during operation. This determination shall include, where possible, comparison with other independent sensors measuring the same variable.

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_{AVG}) is less than the adjusted analysis mean scram time (τ_B) (see Reference 7, of Section 3.11)

τ_{AVG} 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 cycle.

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

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

N_i = number of control rods measured in the ith test.

τ_i = average scram time to the 20% insertion position of all rods measured in the ith 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

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, reactor coolant system leakage, based on sump monitoring, shall be limited to:
   a. 5 gpm Unidentified Leakage
   b. 2 gpm increase in Unidentified Leakage within any 24 hour period
   c. 20 gpm Identified Leakage
   d. no pressure boundary leakage
2. With reactor coolant system leakage greater than 3.6.D.1.a or 3.6.D.1.c above, reduce the leakage rate to within acceptable limits within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
3. With an increase in Unidentified Leakage in excess of the rate specified in 3.6.D.1.b, identify the source of increased leakage within four hours or initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
4. If any Pressure Boundary Leakage is detected when the corrective actions outlined in 3.6.D.2 and 3.6.D.3 above are taken, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.
5. At least one of the leakage measurement instruments associated with each sump shall be operable and the drywell particulate radioactivity monitoring system shall be operable or a sample of the containment atmosphere shall be taken and analyzed at least every four hours. Otherwise, initiate an orderly shutdown of the reactor and reduce reactor water temperature to less than 212°F within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

D. Coolant Leakage

1. Any time irradiated fuel is in the reactor vessel and coolant temperature is above 212°F, the following surveillance program shall be carried out:
   a. Unidentified and Identified Leakage rates shall be recorded at least once every 4 hours using primary containment floor and equipment drain sump monitoring equipment.
   b. Primary containment atmospheric particulate radioactivity shall be recorded at least once every 4 hours.
   c. Drywell pressure and temperature shall be recorded at least once every 12 hours.
2. The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:
   a. Primary containment atmosphere particulate monitoring systems-performance of a sensor check at least once per 12 hours, a channel functional test at least monthly and a channel calibration at least once per cycle.
   b. Primary containment sump leakage measurement system-performance of a sensor check at least once per 4 hours and a channel calibration test at least once per cycle.

Amendment No. ~~75~~, 17

D. Coolant Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached. Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. Systems connected to the reactor coolant system boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

Testing of all required safety/relief valves each refueling outage ensures that any valve deterioration is detected. A tolerance value of 1% for safety/relief valve setpoints is specified in Section III of the ASME Boiler and Pressure Vessel Code. Analyses have been performed with all valves assumed set 1% higher (1108 psig + 1%) than the nominal setpoint; the 1375 psig code limit is not exceeded in any case.

The safety/relief valves are used to limit reactor vessel overpressure and fuel thermal duty.

The required safety/relief valve steam flow capacity is determined by analyzing the transient accompanying the mainsteam flow stoppage resulting from a postulated MSIV closure from a power of 1670 MW_t. The analysis assumes a multiple-failure wherein direct scram (valve position) is neglected. Scram is assumed to be from indirect means (high flux). In this event, the safety/relief valve capacity is assumed to be 83.2% of the full power steam generation rate.

One-inch opening of any one valve or a 1/8-inch opening for all eight valves, measured at the bottom of the disc with the top of the disc at the seat. The position indication system is designed to detect closure within 1/8 inch at the bottom of the disc.

At each refueling outage and following any sigificant maintenance on the vacuum breaker valves, positive seating of the vacuum breakers will be verified by leak test. The leak test is conservatively designed to demonstrate that leakage is less than that equivalent to leakage through a one-inch orifice which is about 3% of the maximum allowable. This test is planned to establish a baseline for valve performance at the start of each operating cycle and to ensure that vacuum breakers are maintained as nearly as possible to their design condition. This test is not planned to serve as a limiting condition for operation.

During reactor operation, an exercise test of the vacuum breakers will be conducted monthly. This test will verify that disc travel is unobstructed and will provide verification that the valves are closing fully through the position indication system. If one or more of the vacuum breakers do not seat fully as determined from the indicating system, a leak test will be conducted to verify that leakage is within the maximum allowable. Since the extreme lower limit of switch detection capability is approximately 1/16", the planned test is designed to strike a balance between the detection switch capability to verify closure and the maximum allowable leak rate. A special test was performed to establish the basis for this limiting condition. During the first refueling outage all ten vacuum breakers were shimmed 1/16" open at the bottom of the disc. The bypass area associated with the shimming corresponded to 63% of the maximum allowable.[1] The results of this test are shown in Figure 3.7.1. Two of the original ten vacuum breakers have since been removed.

When a drywell-suppression chamber vacuum breaker valve is exercised through an opening-closing cycle, the position indicating lights at the remote test panels are designed to function as follows:

Full Closed	2 Green - On 2 Red - Off
Intermediate Position	2 Green - Off 2 Red - Off
Full Open	2 Green - Off 2 Red - On

The remote test panel consists of a push button to actuate the air cylinder for testing, two red lights, and two green lights for each of the eight valves. There are four independent limit switches on each valve. The two switches controlling the green lights are adjusted to provide an indication of disc opening of less than 1/8" at the bottom of the disc. These switches are also used to activate the valve position alarm circuits. The two switches controlling the red lights are adjusted to provide indication of the disc very near the full open position.

The control room alarm circuits are redundant and fail safe. This assures that no simple failure will defeat alarming to the control room when a valve is open beyond allowable and when power to the switches fails. The alarm is needed to alert the operator that action must be taken to correct a malfunction or to investigate possible changes in valve position status, or both. If the alarm cannot be cleared due to the inability to establish indication of closure of one or more valves, additional testing is required. The alarm system allows the operator to make this evaluation on a timely basis. The frequency of the testing of the alarms is the same as that required for the position indication system.

Operability of a vacuum breaker valve and the four associated indicating light circuits shall be established by cycling the valve. The sequence of the indicating lights will be observed to be that previously described. If both green light circuits are inoperable, the valve shall be considered inoperable and a pressure test is required immediately and upon indication of subsequent operation. If both red light circuits are inoperable, the valve shall be considered inoperable, however, no pressure test is required if positive closure indication is present.

The 5% oxygen concentration minimizes the possibility of hydrogen combustion following a loss of coolant accident. Significant quantities of hydrogen could be generated if the core cooling systems failed to sufficiently cool the core. The occurrence of primary system leakage following a major refueling outage or other scheduled shutdown is more probable than the occurrence of the loss of coolant accident upon which the specified oxygen concentration limit is based. Permitting access to the drywell for leak inspections during a startup is judged prudent in terms of the added plant safety offered without significantly reducing the margin of safety. Thus, to preclude the possibility of starting the reactor and operating for extended periods of time with significant leaks in the primary system, leak inspections are scheduled during startup periods, when the primary system is at or near rated operating temperature and pressure. The 24-hour period to provide inerting is judged to be sufficient to perform the leak inspection and establish the required oxygen concentration. The primary containment is normally slightly pressurized during periods of reactor operation. Nitrogen used for inerting could leak out of the containment but air could not leak in to increase oxygen concentration. Once the containment is filled with nitrogen to the required concentration, no monitoring of oxygen concentration is necessary. However, at least once a week the oxygen concentration will be determined as added assurance.

TABLE 3.13.1
SAFETY RELATED FIRE DETECTION INSTRUMENTS

Fire Zone	Location	Minimum Instruments Operable Heat	Flame	Smoke
1A	"B" RHR Room			3
1B	"A" RHR Room			3
1C	RCIC Room			3
1E	HPCI Room			2
1F	Reactor Building-Torus Compartment			11
2A	Reactor Bldg. 935' elev - TIP Drive Area			1
2B	Reactor Bldg. 935' elev - CRD HCU Area East			10
2C	Reactor Bldg. 935' elev - CRD HCU Area West			11
2E	Reactor Bldg. 935' - LPCI Injection Valve Area			1
3B	Reactor Bldg. 962' elev - SBLC Area			2
3C	Reactor Bldg. 962' elev - South			5
3D	Reactor Bldg. 962' elev - RBCCW Pump Area			4
4A	Reactor Bldg. 985' elev - South			4
4B	Reactor Bldg. 985' elev - RBCCW Hx Area			5
4D	SBGT System Room			2
5A	Reactor Bldg. 1001' elev - South			7
5B	Reactor Bldg. 1001' elev - North			3
5C	Reactor Bldg. - Fuel Pool Cooling Pump Area			1
6	Reactor Building 1027' elev			5
7A	Battery Room			1
7B	Battery Room			1
7C	Battery Room			1
8	Cable Spreading Room			7
12A	Turbine Bldg. - 911' - 4.16 KV Switchgear			3
13C	Turbine Bldg. - 911' elev - MCC 133 Area			1
14A	Turbine Bldg. - 931' - 4.16 KV Switchgear			2
15A	#12 DG Room & Day Tank Room		3	
15B	#11 DG Room & Day Tank Room		3	
16	Turbine Bldg. 931' elev - Cable Corridor			3
17	Turbine Bldg. 941' elev - Cable Corridor			3
19A	Turbine Bldg. 931' elev - Water Treatment Area			5
19B	Turbine Bldg. 931' elev - MCC 142-143 Area			1
19C	Turbine Bldg. 931' elev - FW Pipe Chase			1
20	Heating Boiler Room	1		
23A	Intake Structure Pump Room			3

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 follows the northern boundary of the right-of-way for the Burlington Northern Railway.

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. 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.

B. Operations Committee (OC)

1. Membership

The Operations Committee shall consist of at least six (6) members drawn from the key supervisors of the on-site supervisory staff. The Plant Manager shall serve as Chairman of the OC and shall appoint a Vice Chairman from the OC memberaip to act in his absence.

2. Meeting Frequency

The Operations Committee will meet on call by the Chairman or as requested by individual members and at least monthly.

3. Quorum

A quorum shall include a majority of the permanent members, including the Chairman or Vice Chairman

4. Responsibilities - The following subjects shall be reviewed by the Operations Committee:

a. Proposed tests and experiments and their results.

b. Modifications to plant systems or equipment as described in the Updated Safety Analysis Report and having nuclear safety significance or which involve an unreviewed safety question as defined in 10 CFR 50.59.

c. Proposals which would effect permanent changes to normal and emergency operating procedures and any other proposed changes or procedures that are determined by the Plant Manager to affect nuclear safety.

d. Proposed changes to the Technical Specifications or operating license.

e. All reported or suspected violations of Technical Specifications, operating license requirements, administrative procedures, or operating procedures. Results of investigations, including evaluation and recommendations to prevent recurrence, will be reported, in writing, to the General Manager Nuclear Plants and to the Chairman of the Safety Audit Committee.

3. <u>Special Reports</u>

When radioactivity levels in samples exceed limits specified in Table 4.16.3 a Special Report shall be submitted within 30 days from the end of the affected calendar quarter. For certain cases involving long analysis time, determination of quarterly averages may extend beyond the 30 day period. In these cases the potential for exceeding the quarterly limits will be reported within the 30 day period to be followed by the Special Report as soon as practicable.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 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 September 24, 1982, Northern States Power Company (the licensee) proposed changes to the Technical Specifications (TS) of Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The revisions to the Technical Specifications addressed in this Safety Evaluation include the following:

1. Title change from AEC to Commission;
2. Correction of Table numbering;
3. Clarification of definition for Ni;
4. Clarification of the bases section to reflect the removal of two vacuum breakers;
5. Identification of fire detectors that have been installed;
6. Correction of inconsistency between the FSAR and the Technical Specifications on the reactor vessel construction codes and standards;
7. Change from FSAR to USAR as the report to be reviewed by the Operations Committee; and
8. Correction of typographical errors.

Other changes requested in the September 24, 1982 submittal are still under staff review and will be addressed by separate Safety Evaluation and license amendment.

2.0 Evaluation

2.1 Vacuum Breakers

The licensee has proposed to change the bases of section 3.7 to reflect the removal of two vacuum breakers. These revisions supplement Amendment 8 to DPR-22 issued on November 5, 1981. By Amendment 8, the staff approved the licensee's determination that eight (rather than ten) vacuum breakers be operable under normal conditions with six vacuum breakers required to keep the torus to drywell differential pressure below the two psid design limit. Since we have previously evaluated the reduction from ten to eight vacuum breakers and because this revision supplements the previous amendment, we have determined that the level of safety provided by the current Technical Specifications is not diminished. Therefore, the proposed changes to the bases is acceptable.

2.2 Fire Detectors

The licensee proposed changes to Table 3.13.1, "Safety Related Fire Detection Instruments" to reflect the actual number of installed smoke detectors and corresponding locations. The identification of smoke detectors is required to be included in the Technical Specifications and therefore, the proposed changes are acceptable.

2.3 Reactor Vessel Construction Codes and Standards

The licensee has proposed to revise the description of Section 5.3 on the reactor vessel construction codes and standards. An inconsistancy exists between the Final Safety Analysis Report (FSAR) and the Technical Specifications. Since the FSAR is correct, the licensee proposes to reference the information in the FSAR. To further maintain consistency, the design temperature in the Technical Specifications was changed from 575°F to 562°F. This does not alter the actual design of the vessel but references the information in the FSAR. This change was discussed with and agreed to by the licensee. Since these changes do not diminish the level of safety, we find them acceptable.

2.4 Administrative Changes

The licensee has proposed the following changes:

1. Title change from AEC to Commission (TS definition "V" and 6.2.A.5.a)
2. Clarification of definition of Ni (p. 90)
3. Change from FSAR to USAR as the report to be reviewed by the Operations. Committee (TS 6.2.B.4.b)
4. Table renumbering (Table 3.2.7), and
5. Typographical errors (TS 3.6.E/Bases and 6.7.C.3)

Item 4, renumbering the Table to 3.2.7, has been previously amended and therefore, this change is unnecessary. Item 5 (typographical errors) - during the staff's review of the requested license amendment, a couple of typographical errors were observed by the staff. These corrections were discussed with and agreed to by the licensee.

A typographical error was observed by the staff on TS 3.6-D.1.b and the limiting condition was corrected to 2gpm increase in unidentified leakage within any 24-hour period. The staff's intent was to approve an LCO of 2 gpm increase within 24-hour period because it is more conservative than an LCO of 2 gpm increase within any 4-hour period. This correction was discussed with and agreed to by the licensee.

The changes proposed above are administrative in nature and since they do not diminish the level of safety provided by the existing Technical Specifications, we have found them acceptable.

3.0 Environmental Consideration

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 this 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 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.

Date: April 18, 1983

Principal Contributor: H. Nicolaras

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. 17 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 include:

1. Title change from AEC to Commission
2. Correction of Table numbering
3. Clarification of definition for Ni
4. Clarification of the bases section to reflect the removal of two vacuum breakers
5. Identification of fire detectors that have been installed
6. Correction of inconsistency between the FSAR and the Technical Specifications on the reactor vessel construction codes and standards
7. Change from FSAR to USAR as the report to be reviewed by the Operations Committee; and
8. Correction of typographical errors

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 the 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 September 24, 1982, (2) Amendment No. 17 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 18th day of April, 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

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