Docket
DMB-016

June 3, 1981

Docket No. 50-263

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U.S. NUCLEAR REGULATORY COMMISSION

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

The Commission has issued the enclosed Amendment No. 6 to Facility Operating License No. DPR-22 for the Monticello Nuclear Generating Plant. The Amendment consists of changes to the Technical Specifications in partial response to your application dated August 30, 1977 as supplemented by subsequent filings and discussions between the NRC staff and your staff. Portions of your August 30, 1977 application and subsequent filings relating to your inservice testing (IST) program are still under review by the staff and will be the subject of a future license amendment.

This amendment revises the Technical Specifications for Monticello by replacing the current inservice inspection Technical Specifications with an inservice inspection program for the period 1978-1981 that meets the requirements of 10 CFR 50.55a. The issuance of this amendment and revision of the Technical Specifications supersedes the relief from ASME code inservice inspection requirements provided in our letter of February 28, 1978.

Relief from certain inservice inspection requirements is hereby granted as discussed in the enclosed Safety Evaluation. We have determined that the granting of this relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest. This relief is granted, except for certain requirements as discussed in the Safety Evaluation, in response to your request of December 14, 1977 as revised March 15, 1978, August 28, 1978, January 5, 1979, February 26, 1979, July 27, 1979, March 5, 1980, and July 16, 1980. Additional information was transmitted by letters dated January 20 and February 20, 1981. In addition to changes pertaining to the inservice inspection program, several editorial changes of an administrative nature were made which improve the quality of the Monticello Technical Specifications.

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A copy of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

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

Enclosures:
1. Amendment No. 6 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

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

June 3, 1981

Docket No. 50-263

Mr. L. O. Mayer, Manager
Nuclear Support Services
Northern States Power Company
414 Nicollet Mall - 8th Floor
Minneapolis, Minnesota 55401

Dear Mr. Mayer:

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A copy of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

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

Enclosures:
1. Amendment No. 6 to DPR-22
2. Safety Evaluation
3. Notice of Issuance

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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. 6
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 August 30, 1977, as supplemented by letter dated March 15, 1978 and revised August 28, 1978, January 5, 1979, February 26, 1979, July 27, 1979, March 5, 1980, and July 16, 1980, 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. 6 , 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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 6, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 6

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.

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Table 3.2.7
Trip Functions And Deviations

	Trip Function	Deviation
Reactor Building Ventilation Isolation and Standby Gas Treatment System Initiation Specification 3.2.E.3 and Table 3.2.4	Ventilation Plenum Radiation Monitors	+0.2 Mr/Hr
	Refueling Floor Radiation Monitors	+5 Mr/Hr
	Low Reactor Water Level	-6 inches
	High Drywell Pressure	+ 1 psi
Primary Containment Isolation Functions Table 3.2.1	Low Low Water Level	-3 inches
	High Flow in Main Steam Line	+2 %
	High Temp. in Main Steam Line Tunnel	+2°F
	Low Pressure in Main Steam Line	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Water Level	-6 inches
	HPCI High Steam Flow	+7,500 lb/hr
	HPCI Steam Line Area High Temp.	+2°F
	RCIC High Steam Flow	+2250 lb/hr
	RCIC Steam Line Area High Temp	+2°F

Table 3.2.7 - Continued
Trip Function and Deviations

	Trip Function	Deviation
Instrumentation That Initiates Emergency Core Cooling Systems Table 3.2.2	Low-Low Reactor Water Level	-3 Inches
	Reactor Low Pressure (Pump Start) Permissive	-10 psi
	High Drywell Pressure	+1 psi
	Low Reactor Pressure (Valve Permissive	-10 psi
Instrumentation That Initiates Rod Block Table 3.2.3	IRM Downscale	-2/125 of Scale
	IRM Upscale	+2/125 of Scale
	APRM Downscale	-2/125 of Scale
	APRM Upscale	See Basis 2.3
	RBM Downscale	-2/125 of Scale
	RBM Upscale	Same as APRM Upscale
Instrumentation That Initiates Recirculation Pump Trip	High Reactor Pressure	+ 12 psi
	Low Reactor Water Level	- 3 Inches

A violation of this specification is assumed to occur only when a device is knowingly set outside of the limiting trip settings, or, when a sufficient number of devices have been affected by any means such that the automatic function is incapable of operating within the allowable deviation while in a reactor mode in which the specified function must be operable or when actions specified are not initiated as specified.

Amendment No. 6

3.0 LIMITING CONDITIONS FOR OPERATION

F. Deleted

G. Jet Pumps

Whenever the reactor is in the Startup or Run modes, all jet pumps shall be operable. If it is determined that a jet pump is inoperable, the plant shall be placed in a cold shutdown condition within 24 hours.

4.0 SURVEILLANCE REQUIREMENTS

F. Deleted

G. Jet Pumps

Whenever there is recirculation flow with the reactor in the Startup or Run modes, jet pump operability shall be checked daily by verifying that all the following conditions do not occur simultaneously:

1. The two recirculation loop flows are unbalanced by 15% or more when the recirculation pumps are operating at the same speed.

2. The indicated value of core flow rate is 10% or more less than the value derived from loop flow measurements.

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Bases Continued 3.6 and 4.6:

The safety/relief valves have two functions; i.e. power relief or self-actuated by high pressure. The solenoid actuated function (Automatic Pressure Relief) in which external instrumentation signals of coincident high drywell pressure and low-low water level initiate opening of the valves. This function is discussed in Specification 3.5.E. In addition, the valves can be operated manually.

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9 of the ASME Pressure Vessel Code Section III Nuclear Vessels requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

It is realized that there is no way to repair or replace the bellows during operation and the plant must be shut down to do this. The thirty-day period to do this allows the operator flexibility to choose his time for shutdown; meanwhile, because of the redundancy present in the design and the continuing monitoring of the integrity of the other valves, the overpressure pressure protection has not been compromised. The auto-relief function would not be impaired by a failure of the bellows. However, the self-actuated overpressure safety function would be impaired by such a failure.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system quarterly provides assurance of bellows integrity.

When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

F. Deleted

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G. Jet pumps

Failure of a jet pump nozzle assembly hold down mechanism, nozzle assembly and/or riser, would increase the cross-sectional flow area for blowdown following the design basis double-ended line break. Therefore, if a failure occurred, repairs must be made.

The detection technique is as follows. With the two recirculation pumps balanced in speed to within ± 5%, the flow rates in both recirculation loops will be verified by Control Room monitoring instruments. If the two flow rate values do not differ by more than 10%, riser and nozzle assembly integrity has been verified. If they do differ by 10% or more, the core flow rate measured by the jet pump diffuser differential pressure system must be checked against the core flow rate derived from the measured values of loop flow to core flow correlation. If the difference between measured and derived core flow rate is 10% or more (with the derived value higher) diffuser measurements will be taken to define the location within the vessel of failed jet pump nozzle (or riser) and the plant shut down for repairs. If the potential blowdown flow area is increased, the system resistance to the recirculation pump is also reduced; hence, the affected drive pump will "run out" to a substantially higher flow rate (approximately 115% to 120% for a single nozzle failure). If the two loops are balanced in flow at the same pump speed, the resistance characteristics cannot have changed. Any imbalance between drive loop flow rates would be indicated by the plant process instrumentation. In addition, the affected jet pump would provide a leakage path past the core thus reducing the core flow rate. The reverse flow through the inactive jet pump would still be indicated by a positive differential pressure but the net effect would be a slight decrease (3% to 6%) in the total core flow measured. This decrease, together with the loop flow increase, would result in a lack of correlation between measured and derived core flow rate. Finally, the affected jet pump diffuser differential pressure signal would be reduced because the backflow would be less than the normal forward flow.

3.0 LIMITING CONDITIONS FOR OPERATION

3.15 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To assure the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. To be considered operable, Quality Group A, B, and C components shall satisfy the requirements contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda for continued service of ASME Code Class 1, 2, and 3 components, respectively, except where relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

4.0 SURVEILLANCE REQUIREMENTS

4.15 INSERVICE INSPECTION AND TESTING

Applicability:

Applies to the periodic inspection and testing of components which are part of the reactor coolant pressure boundary and their supports and other safety-related pressure vessels, piping, pumps, and valves.

Objective:

To verify the integrity of the reactor coolant pressure boundary and the operational readiness of safety-related pressure vessels, piping, pumps, and valves.

Specification:

A. Inservice Inspection

1. Inservice inspection of Quality Group A, B, and C components shall be performed in accordance with the requirements for ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

Amendment No. 6

Bases 3.15 and 4.15

The inservice inspection program for the Monticello plant conforms to the requirements of 10 CFR 50, Section 50.55a(g). Where practical, the inspection of components classified into NRC Quality Groups A, B, and C conforms to the requirements of ASME Code Class 1, 2, and 3 components, respectively, contained in Section XI of the ASME Boiler and Pressure Vessel Code. If a Code required inspection is impractical for the Monticello facility, a request for a deviation from that requirement is submitted to the Commission in accordance with 10 CFR 50, Section 50.55a(g)(6)(i).

Deviations which are needed from the procedures prescribed in Section XI of the ASME Code and applicable Addenda will be reported to the Commission prior to the beginning of each 10-year inspection period if they are known to be required at that time. Deviations which are identified during the course of inspection will be reported quarterly throughout the inspection period.

A program of inservice testing of Quality Group A, B, and C pumps and valves is also in effect at the Monticello plant. Technical Specifications related to this program will be issued following NRC review and approval of the pump and valve testing program.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 6 TO LICENSE NO. DPR-22

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-263

MONTICELLO NUCLEAR GENERATING PLANT

The Commission has reviewed and evaluated the Inservice Inspection Program (excluding testing of pumps and valves) for the Monticello Nuclear Generating Plant, and finds it in compliance with Paragraph (g) of 10 CFR 50.55a, "Inservice Inspection Requirements." The Inservice Inspection Program was submitted by Northern States Power Company (licensee) on March 15, 1978, and revised August 28, 1978, January 5, 1979, February 26, 1979, July 27, 1979, March 5, 1980 and July 16, 1980. Additional information was supplied in letters dated January 20, 1981, and February 20, 1981.

Pursuant to 10 CFR Part 50.55a(g)(6)(i), relief may be granted from specific requirements stated in the ASME Boiler and Pressure Vessel Code, Section XI, 1974 Edition including Addenda through Summer 1975, which we have concluded to be impractical for the facility because of component or system design, geometry, or materials of construction. In some cases, relief may be granted only after performing the alternative inspection requirements which the staff deems necessary. Such relief and alternative requirements are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. In some cases, relief should not be granted because of the factors stated in the evaluation of the specific request.

I. Relief Request Evaluations

A. Reactor Vessel Welds in Core Region (Examination Category B-A, Item B1.1); in Shell & Bottom Head (Examination Category B-B, Item B1.2); Standby Liquid Control Nozzle to Vessel Welds (Examination Category B-D, Item B1.4); Integrally Welded Vessel Stabilizer Lugs (Examination Category B-H, Item B1.12); and Standby Liquid Control Nozzle to Safe End Welds (Examination Category B-F, Item B1.6); (Relief Requests 15, 16, and 19).

Code Requirement

For core, shell, and bottom head welds, volumetric examination of 10% of each longitudinal weld and 5% of each circumferential weld. For the nozzle to vessel welds, volumetric examination of weld, base metal and inside radius. For the vessel stabilizer lugs, volumetric examination of 100% of the welding to the vessel. For the nozzle to safe end welds, volumetric and surface examination of the circumference of 100% of the welds.

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Code Deviation Request

Delete volumetric examination from the following welds:

Examination Category	Weld Identification
B-A	VLAA-1,2; VLBA-1,2; VCBA-2
B-B	VLBA-1,2; VCBB-3; HMAB-9,10
B-D	CPAD-1
B-F	CPAF-2
B-H	4 lugs

Reason for Request

The design of the reactor internals and the external, biological shield and vessel insulation prevents access to these welds. The Monticello RPV was constructed with a 2'3½" thick biological shield wall surrounding it, with the exception of the top eight or so feet. Between this wall and the reactor vessel shell is a space of approximately 1 foot that houses the thermal insulation. The only access areas to the reactor vessel is: 1. at the top eight feet above the biological shield wall, 2. through openings in the wall at each nozzle location and two inspection ports below the skirt weld, 3. by the control rod drives under the reactor head, and 4. from the vessel inside diameter. There are no nozzles in the core region for access to Category B-A welds.

The area above the biological shield wall and at the nozzle opening is further obstructed by non-removable insulation. A good portion of the vessel insulation was not designed to be removed and therefore it was installed prior to the installation of the piping, electrical conduits, vessel stabilizers, duct work, etc.

A very thorough review was performed, using drawings, sketches, and previous examination reports, to try and locate weld areas that possibly could be inspected. It was concluded that some of the vessel welds appear to be close enough to nozzle openings for performing the examinations provided the insulation can be removed. An attempt will be made to remove or modify the insulation and to examine some of these welds during the 1981 refueling outage. In addition, during the upcoming outage we are hopeful that radiation levels will permit access to the bottom of the vessel (between the control rod drives, skirt, and vessel). This would allow accessibility to a larger portion of the skirt to vessel weld and to the bottom head meridional and dollar plate welds.

Each of the welds that can be examined will be sketched to show the examination amount, extent and location. In addition, each area will be scheduled for examination during the next ten year interval.

The design of the Monticello reactor vessel internals does not allow internal access to the vessel welds either in or below the core region, but does give internal access to the top 20 ft. of the reactor vessel. This access allows one circumferential and two

longitudinal welds to be examined that were thought to be inaccessible from the outside of the vessel.

The Standby Liquid Control nozzle and safe end protrudes through a hole in the vessel thermal insulation with a minimum of clearance. It appears, from an early sketch, that the bottom of the weld is next to and just outside the insulation. The insulation around this nozzle was not designed with removable panels. With the present design the nozzle to safe end weld is 100% accessible for ultrasonic examination from one side, and approximately 50-75% accessible for a liquid penetrant examination.

The volumetric and surface examinations required on this weld will be performed to the extent possible during the 1981 refueling outage. During this outage an attempt will be made to modify the insulation to allow 100% coverage of the weld for both examination methods. In addition, a sketch will be made that will identify any areas that remain inaccessible.

It should be noted that the Monticello reactor vessel was fabricated and subject to as-built inspection under very demanding specifications. Because the site was inaccessible to a river barge of the capacity necessary to transport a fully assembled vessel, the vessel was assembled at the site from shop-fabricated subassemblies. All requirements of Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, including Addenda through Summer 1966, were satisified just as if the vessel were shop fabricated. Additional requirements more stringent than those required by the Code were applied by General Electric due to the unique circumstances surrounding the vessel fabrication. Refer to Volume VII of the Monticello Final Safety Analysis Report, "Pressure Vessel Design Report" for details concerning vessel fabrication and inspection. In addition, it should be noted that based on analysis of the dosimeter removed from the reactor vessel, the maximum neutron fluence level at T/4 of the reactor vessel was estimated to be only 1.23×10^{18} nvt at the end of designed life (40 years). Based on the stringent construction requirements, the inability to examine these welds during service is not considered to significantly reduce the vessel integrity.

Staff Evaluation

The design of the biological shield, vessel insulation, and reactor internals prevents access from all of the code required examinations of the reactor vessel welds. Imposition of the exact code requirements would subject the licensee to extreme hardships in necessitating the removal of portions of the concrete biological shield and the permanently installed insulation to perform the required examinations of the welds from the vessel's outside surface.

The code requires that 10% of each longitudinal weld and 5% of each circumferential weld be examined during the inspection interval. To compensate for those welds which are inaccessible for volumetric examination, the licensee has agreed to examine all accessible areas

of circumferential or longitudinal welds. Table I is a summary of each Category B-A and B-B weld, the length of each weld, and the length believed to be accessible for examination. The licensee has indicated that in excess of 150 feet of longitudinal and circumferential welds may be examined by the end of the interval. The ASME code requires that only 40 feet be examined during the same time period.

To help ensure the integrity of the vessel, the vessel materials are presently evaluated for radiation damage by use of the requirements in Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." To ensure that stresses in the reactor vessel remain within acceptable limits, the Monticello Nuclear Generating Plant is subject to pressure/temperature limitations for reactor coolant system heatup and cooldown operations, and inservice leak and hydrostatic tests in the plant Technical Specifications.

The licensee has indicated that Category B-H welds may be accessible for examination pending the results of the next refueling outage. We will review and evaluate this request for relief after the examinations have been attempted and the extent and methods of examination possible are known.

We conclude that the augmented examinations for certain welds, the ongoing surveillance program of the reactor vessel materials in the beltline region, and the vessel design are adequate for providing an acceptable level of safety and assurance that the vessel's structural integrity will not be compromised during the inspection period by granting relief from the examinations of welds discussed above.

B. Control Rod Drive and Bottom Head Drain Vessel Penetrations (Examination Category B-E, Item B1.5). Relief Request 17.

Code Requirement

The area surrounding each penetration shall be visually examined for evidence of leakage during the pressure test.

Code Deviation Request

Visually inspect areas below penetrations during pressure test.

Reason for Request

The design of the vessel, the biological shield, and vessel insulation prevents access to these areas that are directly adjacent to the vessel penetrations.

Staff Evaluation

The relief request evaluation in I.a. describes in detail the design of the vessel and insulation which makes these examinations

TABLE I
MONTICELLO REACTOR VESSEL WELD SUMMARY

EXAM CATEGORY	WELD NUMBER	LOCATION	LENGTH	REQ'D EXAM LENGTH	ACCESSIBLE* LENGTH	REMARKS
B1.1 B-A LONG	VLAA-1	TOP 27"	11'	1.1'	6"	ATTEMPT 2ND INTERVAL
	VLAA-2	TOP 27"	11'	1.1'	6"	ATTEMPT '81 OUTAGE
	VLBA-1	BOT 117"	11'	1.1	NONE	-
	VLBA-2	BOT 117"	11'	1.1'	NONE	-
CIRC	VCBA-2	-	57'	2.85'	NONE	-
B1.2 B-B LONG	VLAA-1	BOT 8'8"	11'	1.1'	4'	ATTEMPT 2ND INTERVAL
	VLAA-2	BOT 8'8"	11'	1.1'	4'	ATTEMPT '81 OUTAGE
	VLBA-1	TOP 15"	11'	1.1'	NONE	-
	VLBA-2	TOP 15"	11'	1.1'	NONE	-
	VLCB-1	-	11'	1.1'	4'	ATTEMPT '81 OUTAGE
	VLCB-2	-	11'	1.1'	4'	ATTEMPT '81 OUTAGE
	VLDB-1	-	11'	1.1'	5'	SCHEDULED '81
	VLDB-2	-	11'	1.1'	5'	57" EXAMINED
CIRC	VCBB-1		57'	2.85'	6'	ATTEMPT '81 OUTAGE
					6'	ATTEMPT 2ND INTERVAL
	VCBB-3	-	57'	2.85'	NONE	-
	VCBB-4	-	57'	2.85'	16'	ATTEMPT '81 OUTAGE
CLOSURE HD MERIDONAL	HMCB-1	-	7'	.7'	ALL	7' EXAMINED
	HMCB-2	-	7'	.7'	ALL	2.5' EXAMINED
	HMCB-3	-	7'	.7'	ALL	7' EXAMINED
	HMCB-4	-	7'	.7'	ALL	7' SCHEDULED '81
	HMCB-5	-	7'	.7'	ALL	7' EXAMINED
	HMCB-6	-	7'	.7'	ALL	7' SCHEDULED '81
CIRC	HCCB-2	-	25'	1.25'	ALL	25' EXAMINED
BOTTOM HD MERIDONAL	HMAB-1	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-2	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-3	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-4	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-5	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-6	-	6'2"	.6'	2'8"	ATTEMPT 2ND INTERVAL
	HMAB-7	-	6'2"	.6'	2'8"	ATTEMPT '81 OUTAGE
	HMAB-8	-	6'2"	.6'	2'8"	ATTEMPT '81 OUTAGE
	HMAB-9	-	14'5"	1.5'	NONE	-
	HMAB-10	-	13'8"	1.4'	NONE	-
CIRC	HCAB-1	-	44'	2.2'	ALL	ATTEMPT '81 OUTAGE
B1.12 B-H LUGS	4 LUGS	-	-	4 LUGS	4 LUGS	ATTEMPT '81 OUTAGE

*Length depends on the results obtained during the 1981 outage.

impractical. However, since the vessel pressure test hold time is 4 hours, and the licensee has committed to inspection of the areas below these penetrations for evidence of leakage, any loss of leak-tight integrity should be detectable. We agree that the alternate test and inspection procedure proposed by the licensee provides an acceptable level of safety and therefore grant relief from the visual inspection requirement of Category B-E.

C. Closure Head Flange Leakage Sensors (Examination Category B-E, Item B1.5). Relief Request 18.

Code Requirement

The area surrounding each penetration shall be visually examined for evidence of leakage during the pressure test.

Code Deviation Request

Nozzles will be visually inspected only if area insulation is removed for other maintenance or inspection activities. This relief request applies to nozzles N-13 and N-14.

Reason for Request

There are two seal leak detector nozzles; N13 and N14. Nozzle 13 has an opening on the vessel flange face between the inner and outer "O" rings, and nozzle 14 has an opening between the outer "O" ring and the outer gasket. The inner "O" ring seal would have to leak for N13 to see liquid level and pressure buildup, and both "O" ring seals would have to leak for N14. The seal leakage is monitored through N13, and the line from N14 is run to a point inside the drywell and capped off.

The reactor vessel head seal leak detection system is designed to monitor reactor vessel head seal integrity. This is accomplished by detecting liquid level buildup and pressure buildup in the drain line between two metal "O" rings which comprise the reactor vessel head seal. The reactor vessel head seal leak detection system consists of a closed chamber located in the head seal drain line. A float type level switch is mounted in this chamber. A pressure switch and a pressure indicator are also included as part of the system. The only control included in the vessel head seal leak detection system is the reactor flange drain valve switch, which operates to energize and de-energize valves CV-2369 and CV-2370. With the switch in the CLOSE position, SV-2369 is energized to open CV-2369 and SV-2370 is de-energized to close CV-2370. Local contacts light the CV-2369 red indicator lamp and the CV-2370 green indicator lamp, indicating the seal leak detection system is in service. To drain the system, the handswitch is placed in the OPEN position de-energizing SV-2369 to close CV-2369 and energizing SV-2370 to open CV-2370 allowing any leakage collected in the float cage to drain to the Drywell Equipment Drain Sump. The facility to drain the reactor vessel head-seal leak detection system provides a convenient method of determining the approximate leak rate. Draining and

returning the system to service permits recording the time required to refill the closed chamber and subsequent calculation of the leak rate.

The reactor vessel insulation prohibits access to the N13 and N14 nozzle area, and was not designed to be removed. The insulation would receive considerable damage by removal and replacement. Removable insulation has not been considered due to the fact that the seal leak detectors do not see reactor pressure and it would therefore serve no purpose.

Staff Evaluation

The Northern States Power Company has a system designed to monitor the reactor vessel head seal integrity and leakage. Nozzle 14 is outside the vessel head O-rings and nozzle 13 is located between the vessel head O-rings. Any leakage of the vessel flange O-rings would be detected by the vessel head seal leak detection system.

The licensee has agreed to inspect the area around each nozzle if the insulation is removed for maintenance or inspection activities. In addition, the staff requires that the vessel head seal be hydrostatically tested once per inspection interval. The staff concludes that the combination of the leak detection system design and hydrostatic testing provide reasonable assurance that the integrity of these nozzles will be maintained. Therefore, relief from the visual inspection requirements of nozzles 13 and 14 may be granted.

D. Piping Socket Welds (Examination Category B-1, Item B4.8) and Valve Bonnet Bolting (Examination Category B-G-2, Item B6.9). Relief Request No. 22.

Code Requirement

Category B-J: Surface examinations performed during each inspection interval shall include all of the area of 25% of the circumferential joints including the adjoining one-foot sections of longitudinal joints and 25% of the pipe branch connection joints of welds six inches in diameter and smaller.

Category B-G-2: A visual examination shall be conducted on all pressure retaining bolting two-inch diameter or smaller.

Reason for Request

These two, 2-inch drain lines are reading in excess of 2 R/hr. The location of these lines prevents the use of shielding or distance to provide any significant reduction in radiation exposure to personnel. We have estimated that exposure to inspection and insulating personnel would be in excess of 1 man-rem for the examination of approximately four socket welds and the bolting of four valves.

Staff Evaluation

The licensee has not demonstrated that the performance of the code required examinations is impractical due to design, geometry, or materials of construction and therefore relief cannot be granted at this time.

E. Visual Examination of Recirculation Pumps Casings (Examination Category B-L-2, Item B5.7). Relief Request 41.

Code Requirement

The internal pressure boundary surfaces of one pump in each group of pumps performing similar functions in the system shall be visually examined once each interval.

Code Deviation Request

The visual examination of the recirculation pumps internal pressure surfaces will not be scheduled as required by code.

Reason for Request

Disassembly of the recirculation pumps for the sole purpose of visual examination of the casing internal pressure surfaces requires many man-hours of skilled maintenance personnel. Increased radiation exposures result from this activity. The probability of pump failure is increased by unnecessarily disassembling the units. Deferring the examination has no effect on integrity of the pumps. The internal pressure surfaces of these pumps will be visually examined when the pumps are disassembled for maintenance.

Staff Evaluation

The disassembly of the reactor recirculation pumps to inspect the internal pressure retaining surfaces is a major effort in terms of man-hours and personnel exposure. An estimated 614 man-hours and 30-60 man-rem would be required for a visual examination. The licensee has committed to a visual examination if the pump is disassembled for maintenance. In addition the following parameters are monitored by the licensee for each of the recirculation pumps:

1. Pump flow

2. Differential pressure across pump

3. Pump motor vibration

4. Pump oil level

5. Pump discharge temperature

6. Pump seal circulation pressure (first and second stage)

7. Temperature of the motor thrust bearing (upper and lower face)

8. Temperature of the motor guide bearing (upper and lower)

9. Temperature of the motor windings

10. Temperature of the first and second stage seal cavity

11. Temperature of the seal closed cooling water

12. Temperature of the motor closed cooling water

Deviations in some of the above monitored parameters may indicate degradation of pump components. These degraded components could affect pump performance to the point that maintenance would be required long before degradation of the internal pressure boundary surface is noticeable.

We conclude that the effort required for dissasembly of the pumps and the associated radiation exposure solely for a visual examination is impractical and imposes an undue burden on the licensee. Visual examinations will be conducted if the pumps are disassembled for other reasons. Other parameters of pump operation will aid in detecting degradation of pump components. Relief is therefore granted from the visual examination requirement.

F. Visual Examination of Recirculation Valves (Examination Category B-M-2, Item B6.7), Relief Request 42.

Code Requirement

Visual examination of internal pressure boundary surfaces on valves exceeding 4 in. nominal pipe size, once each interval. One valve in each group of valves of the same constructional design (e.g., globe, gate, or check valve), manufacturing method, and manufacturer that performs similar functions in the system shall be examined.

Code Deviation Request

The visual examination of the recirculation valve internal pressure surfaces will not be scheduled as required by the code.

Reason for Request

Dissassembly of the recirculation valves for the sole purpose of visual examination of the internal pressure surfaces requires many man-hours from skilled maintenance personnel and increased radiation exposures result from this activity. The probability of valve failure is increased by unnecessarily disassembling the units. Deferring the examination has no effect on the integrity of the valves. The internal pressure surfaces will be visually examined when the pumps are disassembled for maintenance.

Staff Evaluation

The disassembly of these valves to visually inspect the pressure retaining surfaces would require approximately 266 man-hours of labor and 13-26 man-rem in radiation exposure. The licensee has committed to a visual examination if the valves are disassembled for maintenance.

In addition, periodic hydrostatic testing as required by the code provides further assurance that the structural integrity of the valve bodies will be maintained. Relief is therefore granted from the visual examination requirement of the code.

G. Pressure Testing of all Class 1, 2, and 3 Components (Relief Requests 30 and 31).

Code Requirement

IWB-5000: Class 1 components shall be subject to a hydrostatic pressure test at or near the end of each inspection interval. The pressure shall be 1.10 X P_o at 100°F.

IWC-5000: Class 2 components shall be subjected to a hydrostatic pressure test at or near the end of each inspection interval. The pressure shall be 1.25 X P_D at 100°F.

IWD-5000: Class 3 components shall be subjected to a system pressure test of 1.10 X P_D.

Code Deviation Request

The test pressure requirements will not be met on certain components.

Reason for Request

The code does not recognize that non-isolable junctions of components with different design pressures or different ASME classes exist (i.e., pump suction and discharge lines, piping upstream and downstream of restricting orifices, etc.). Pressurizing components to the requirements of the code may result in over-pressurizing the non-isolable components.

Staff Evaluation

Because of the design of some systems, isolation of different ASME classes cannot be made at the class boundary. The licensee has committed to hydrostatically test the non-isolable portions of these systems at the lower design pressure and to conduct pressure tests during system inservice tests.

The staff concludes that the alternate testing program proposed by the licensee is adequate to provide evidence of leakage and, therefore, grants relief from code pressure test requirements for these junctions.

H. Reactor Vessel Safe End Welds and Piping Welds (Examination Categories B-F, B-J, B-K, B-F, C-F, C-G). Relief Request No. 21.

Code Requirement

The ultrasonic examination requirements of Appendix I of Section XI apply to Class 1 and 2 ferritic vessels 2-1/2 inches and greater in thickness. Where Appendix I is not applicable, the provisions of Article 5 of Section V shall apply.

Code Deviation Request

The rules of Appendix III, including Supplement 7, of the Winter 1975 and Summer 1976 Addenda to ASME Section XI will govern the ultrasonic examination method for the inspection of pipe welds and welds of components fabricated from pipe components.

Reason for Request

The use of side drill holes (instead of slots) to establish a distance amplitude correction curve (DAC) for pipe weld inspections, as required by Appendix I of Section XI and Article 5 of Section V, results in an excessive instrument gain setting which greatly impairs the inspector's ability to detect and to interpret indications by producing a lower signal-to-noise-ratio and reduces the range of useable DAC.

Staff Evaluation

Appendix III, including Supplement 7 of the Winter 1975 Addenda and Summer 1976 Addenda, is not approved by the NRC in 50.55a(b) of 10 CFR 50. However, the 1977 Edition through Summer 1978 Addenda are essentially the same as the earlier addenda. Hence, the provisions of Appendix III specified by the licensee may be utilized. For ferritic vessels greater than 2 inches in thickness, the requirements of Article 4 of Section V in the Winter 1975 Addenda or Appendix I in the 1974 Edition of Section XI must be met since Appendix III applies only to piping welds.

The staff requires the following regarding DAC recording levels for piping welds:

1) All indications at or above 50% DAC shall be recorded.

2) All indications 100% DAC or greater shall be recorded and evaluated in accordance with the rules of Section XI.

3) Indications 20% of DAC or greater which are interpreted to be a crack must be identified and evaluated to the rules of Section XI.

4) The owner shall evaluate and take corrective action for the disposition of any indication investigated and found to be other than geometric in nature.

I. Class 1 and 2 Bolts and Studs (Examination Categories B-G-1 and C-D), Relief Request 24.

Code Requirement

Article 5 of Section V requires that calibration be established by a test bar of the same nominal composition and diameter as the production part and minimum of one-half of the length. A 3/8-inch (10mm) diameter X 3-inch (76mm) deep flat bottom hole shall be drilled in one end of the bar and plugged with similar material to full depth. A distance amplitude curve shall be established by scanning from both ends of the best bar.

Code Deviation Request

Use the alternate procedure for ultrasonic examination in Paragraph T525.1 for bolts and studs prior to threading (Back Reflection Procedure).

Reason for Request

The variation in ultrasonic attenuation between bolts and/or studs diminishes the usefulness of a DAC generated from a particular specimen.

In addition, a procedure qualification test was performed comparing the ASME Section V requirements with the NSP Back Reflection Bolt and Stud Examination Procedure. The results showed that the NSP procedure was slightly more sensitive than the ASME Section V requirements.

The comparison was performed using a 2" dia. X 15" long test bar machined as a calibration standard in accordance with ASME Section V, and another 15" long section of the same bar used to simulate a stud. The back reflection of the stud was set at 80% FBH in accordance with the NSP procedure. The amplitude of the 3/8" dia. FBH (at 12") in the calibration standard was then noted to be about 6 dB greater than the reporting level for the NSP procedure.

This result should be generally true regardless of bar length for diameters greater than 3/4" because the ratio between the area of the FBH and the 3/4" dia. transducer remains constant at 1/4. Therefore, the FBH should produce an echo of at least 25% of the back return, depending on the relative finish of the reflecting surfaces. In addition, the poorer the end reflecting surfaces the more sensitive the NSP procedure would be with respect to the Section V method. This tends to make the NSP procedure a more conservative approach to bolt and stud examination.

Staff Evaluation

As an alternate procedure the licensee proposes examination of studs/bolts using the first back reflection. The procedures

requires that the back reflection be set at 80% of screen height. Any discontinuity producing an indication 20% of the back reflection or greater is recorded. In addition, any indication producing 50% loss of back reflection that is not caused by bolt or stud geometry is recorded and evaluated: The licensee's procedure will be used where the code requires ultrasonic examination of bolts/studs except for reactor vessel studs which will be examined per code.

The licensee provided data showing that this alternate procedure is as sensitive for flaw detection as the alternate code procedure.

Paragraph IWA-2240 of Section XI permits the use of alternative examinations provided the results yield demonstrated equivalence or superiority. Relief may therefore be granted for this deviation from the code.

J. Non-Welded Piping and Valve Supports (Examination Categories B-K-2 and C-E-2). Relief Request No. 23.

Code Requirements

The examination performed during each inspection interval shall cover all support components. The areas of examination shall include support components that extend from the piping, valve, and pump attachment to and including the attachment to the supporting structure.

Code Deviation Request

The Code requires all areas of the support component from the piping, valve, and pump attachment to and including the attachment to the supporting structure. Insulation will not be removed for visual examination of these support components.

Reason for Request

The general radiation background field for the inspection of Class 1 systems located within containment ranges from 30 to 400 mr and the Class 2 systems have permanent type of insulation (insulation not designed for removal and replacement). Approximately 35 man-rem in personnel exposure would result from Class 1 examinations in the third inspection period.

It has been our experience that any loss of support capability or inadequate restraint can usually be detected through the inspection of the uninsulated portion of the support and the surrounding insulation. It is our contention that the removal and replacement of insulation for the sole purpose of inspecting Class 1 supports would result in undue radiation exposure to personnel without providing a significant increase in safety. The governing codes and regulations used in the design and construction of those systems that are now classified as Class 2 and 3 did not require provisions for inspection access for these systems. Thus, it would be an undue burden without compensating increase in safety to require insulation removal for support inspection.

Staff Evaluation

The mechanical connections to pipe strap are exposed and the licensee has committed to visual examination of these connections.

The estimated exposure is 35 man-rem just to remove insulation to perform visual inspection of the pipe strap. Possible damage to the pipe strap can be detected by noting damage to insulation and the licensee has committed to remove insulation around pipe support in the event that visual examination of the insulation indicates pipe strap damage or loss of support.

The staff finds the design of the supports and insulation make the exact code required examinations impractical, agrees that the above procedure will provide an adequate level of safety and, therefore, grants relief from the visual inspection criteria of Category B-K-2 and C-E-2; Items 1.4, 2.2, 2.3 and 2.4.

II. Additional Relief Requests

In addition to the relief requests evaluated in Section I, the licensee submitted three requests for relief which involved updating to examination requirements of the 1977 Edition through Summer 1978 Addenda of Section XI of the ASME Code. Updating to the requirements of later NRC approved editions and addenda is permitted by 50.55a(g)(4)(iv), provided all of the related requirements of the respective editions or addenda are met. We have evaluated the following relief requests submitted by the licensee and find them to be acceptable and in accordance with 50.55a(g)(4)(iv):

Relief Request Identification	Examination Category	Component Examination
No. 20	B-I-1, Item 81.13	Visual exam of closure head cladding.
No. 36	All Class 1, 2 & 3	Holding time during hydrotest.
No. 37	All Class 2 & 3 Air & Nitrogen Systems	Pressure testing with contained fluid.
No. 38	All Class 1 & 2	Austenitic stainless steel temperature during hydrotest.

III. Administrative Changes

Table 3.2.G on pages 70 and 71 of the Technical Specifications was renumbered to Table 3.2.7 to correct duplication of numbering with Table 3.2.6 on page 60A. A minor editorial change was also made to Table 3.2.7. We find these changes acceptable as administrative changes.

IV. Summary

Based on the evaluations of the requested relief from ASME Code requirements, we conclude that the Monticello Nuclear Generating Plant Inservice Inspection Program for last period in the first ten year interval meets the requirements of the 1974 Edition through Summer 1975 Addenda of the ASME Code to the extent practical and thus complies with 50.55a(g) in 10 CFR 50.

V. Environmental Consideration

We have determined that this 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.

VI. 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 the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: June 3, 1981

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-263

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND GRANTING OF RELIEF

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 6 to Facility Operating License No. DPR-22, issued to Northern States Power Company, which revised 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 revises the Technical Specifications to replace the current inservice inspection technical specifications with an inservice inspection program that meets the requirements of 10 CFR 50.55a and to incorporate several changes of an administrative nature.

By letter dated April 10, 1981, as supported by the related safety evaluation, the Commission has also granted relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" to the licensee. The relief relates to the inservice inspection program for the facility. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR 50. The relief is effective as of its date of issuance.

The application for the 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 this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment and the granting of this relief 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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated August 30, 1977, the licensee's submittals dated March 15, 1978, August 28, 1978, January 5, 1979, February 26, 1979, July 27, 1979, March 5, 1980, and July 16, 1980, (2) Amendment No. 6 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, NW., 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 3rd day of June 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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