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and 50-306
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Docket Nos. 50-282
and 50-306

Northern States Power Company
ATTN: Mr. L. O. Mayer, Manager
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Gentlemen:

In response to your applications dated December 9, 1977 and September 8, 1978, the Commission has issued the enclosed Amendment Nos. 32 and 26 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments change the Technical Specifications that relate to the pressurizer heatup rate, the reactor vessel pressure-temperature operating limits, and the part length rods. The Table of Contents of the Technical Specifications has also been updated.

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

Sincerely,

ORIGINAL SIGNED

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 32 and 26 to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice of Issuance

7811170256

OELD concurs with changes discussed with Mr. Grotenhuis
ESF Const. I
GD

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-42

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

   A. The applications for amendment by Northern States Power Company (the licensee) dated December 9, 1977 and September 8, 1978, comply 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.

7811170259

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 License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 1, 1978

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. DPR-60

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

   A. The applications for amendment by Northern States Power Company (the licensee) dated December 9, 1977 and September 8, 1978, comply 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 License No. DPR-60 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 26 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: November 1, 1978

ATTACHMENT TO LICENSE AMENDMENT NOS. 32 AND 26

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated licenses with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
ii	ii
iii	iii
iv	iv
TABLE TS.3.1-1	TABLE TS.3.1-1
TABLE TS.3.1-2	TABLE TS.3.1-2
FIGURE TS.3.1-1	FIGURE TS.3.1-1
FIGURE TS.3.1-2	FIGURE TS.3.1-2
FIGURE TS.3.1-3	FIGURE TS.3.1-3
FIGURE TS.3.1-4	FIGURE TS.3.1-4
TS 3.10-4	TS 3.10-4
TS 3.10-5	TS 3.10-5
TS 5.3-1	TS 5.3-1

APPENDIX A TECHNICAL SPECIFICATIONS

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APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF TABLES

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

B. HEATUP AND COOLDOWN

Specification:

1. The Unit 1 and Unit 2 reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS.3.1-1 and TS.3.1-2 for the first full power service period.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures TS.3.1-1 and TS.3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figures TS.3.1-1, TS.3.1-2 shall be recalculated periodically using methods discussed in the Basis section.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

4. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis

The reactor coolant system heatup and cooldown rates in Figures TS.3.1-1 and TS.3.1-2 are applicable to both Unit 1 and Unit 2. The curves are based on Unit 1 toughness data and are conservative for the Unit 2 vessel. Toughness data is included in Tables TS.3.1-1 and TS.3.1-2.

Table TS 3.1-1

UNIT NO. 1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	Transverse[a] 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RT_{NDT} (°F)	Average Transverse[a] Upper Shelf (ft lb)
Closure Head Dome	A533 Gr. B, Cl. 1			-4	64[c]	4[c]	75 [c]
Head Flange	A508 Cl. 3			-4	12[c]	-4[c]	84 [c]
Vessel Flange	A508 Cl. 3			-4	41[c]	-4[c]	77.5 [c]
Injection Nozzles	A508 Cl. 3			-22	-114[c]	-22 [c]	97 [c]
Inlet and Outlet Nozzle	A508 Cl. 3			+5	39[c]	5 [c]	92 [c]
Upper Shell	A508 Cl. 3			-4	39[c]	-4 [c]	85 [c]
Inter. Shell[b]	A508 Cl. 3	0.06	0.013	+14	14	14	143
Lower Shell[b]	A508 Cl. 3	0.07	0.014	-4	45	-4	134
Trans. Ring	A508 Cl. 3			+5	63[c]	5	79 [c]
Bottom Head	A533 Gr. B, Cl. 1			-4	57[c]	-3	68.5 [c]
Weldment[b]	Weld	0.13	0.017	0	10	0	78.5
HAZ[b]	HAZ			0 [c]	<-100	0	211

a. Specimen oriented normal to the major working direction

b. Based on actual transverse data through the surveillance program

c. Estimated using "Pressure-Temperature Limits," Section 5.3.2 of *Standard Review Plan*, NUREG-75/087, 1975, from longitudinal data.

TABLE TS.3.1-1

TABLE TS.3.1-2

UNIT NO. 2

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	Transverse[a] 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RT_{NDT} (°F)	Average Transverse[a] Upper Shelf (ft lb)
Closure Head Dome	A533 Gr. B, Cl. 1			5	52[c]	5	64[c]
Head Flange	A508 Cl. 3			-31	18[c]	-31	87[c]
Vessel Flange	A508 Cl. 3			-22	18[c]	-22	88[c]
Injection Nozzles	A508 Cl. 3			-22	-114[c]	-22	97[c]
Inlet and Outlet Nozzle	A508 Cl. 3			-13	50[c]	-10	89[c]
Upper Shell	A508 Cl. 3			-13	41[c]	-13	85[c]
Inter. Shell[b]	A508 Cl. 3	0.075	0.010	-4	56	-4	112
Lower Shell[b]	A508 Cl. 3	0.085	0.011	-13	54	-6	108
Trans. Ring	A508 Cl. 3			10	50	10	76[c]
Bottom Head	A533 Gr. B, Cl. 1			-13	56	-4	68[c]
Weldment[b]	Weld	0.082	0.019	-31	-6	-31	103
HAZ[b]	HAZ			-31	-35	-31	117

a. Specimen oriented normal to the major working direction

b. Based on actual transverse data through the surveillance program

c. Estimated using "Pressure-Temperature Limits," Section 5.3.2 of *Standard Review Plan*, NUREG-75/087, 1975, from longitudinal data.

FIGURE TS.3.1-1

UNIT 1 AND UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
(Applicable for First 10 EFPY of Operation)

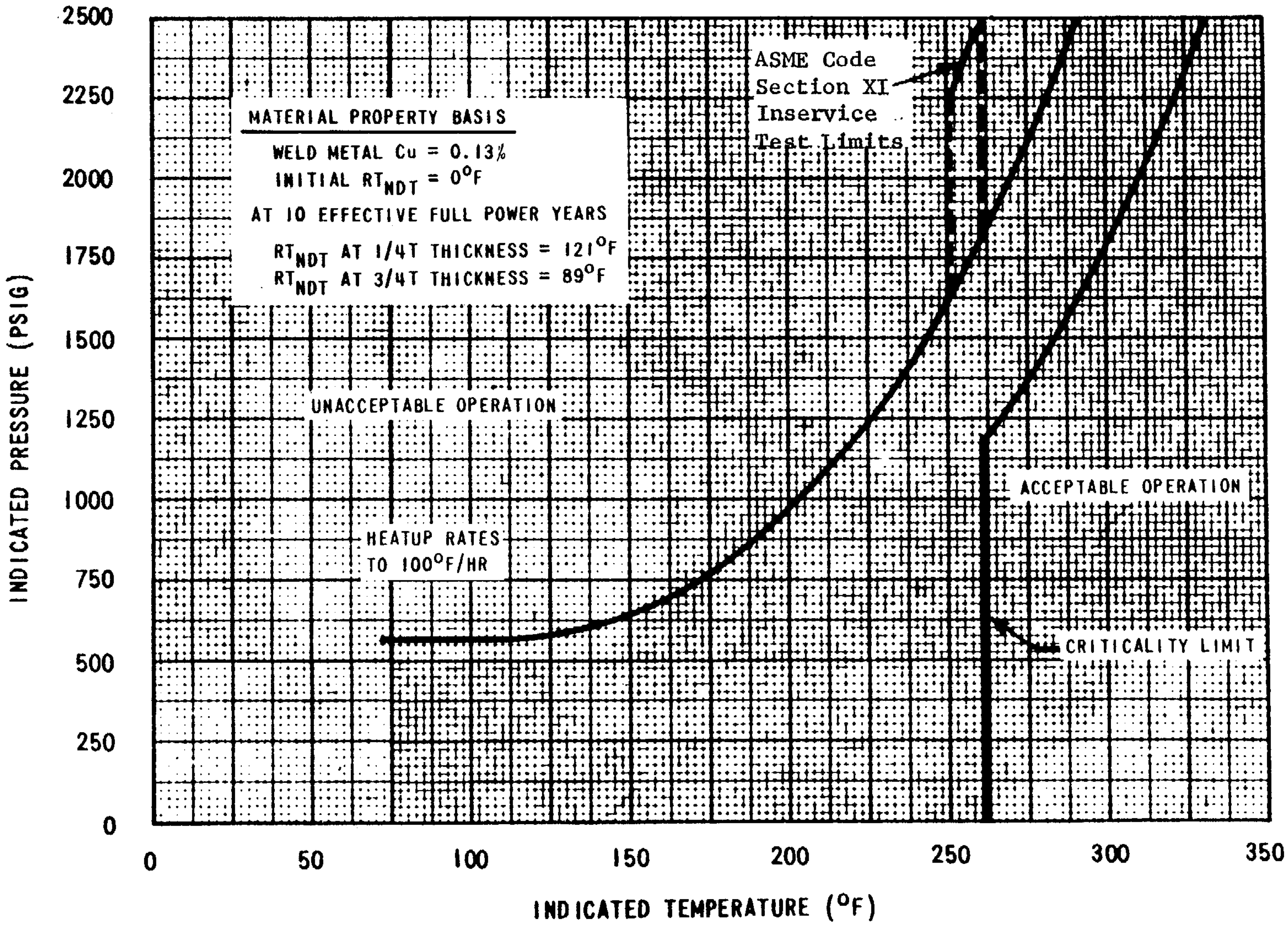
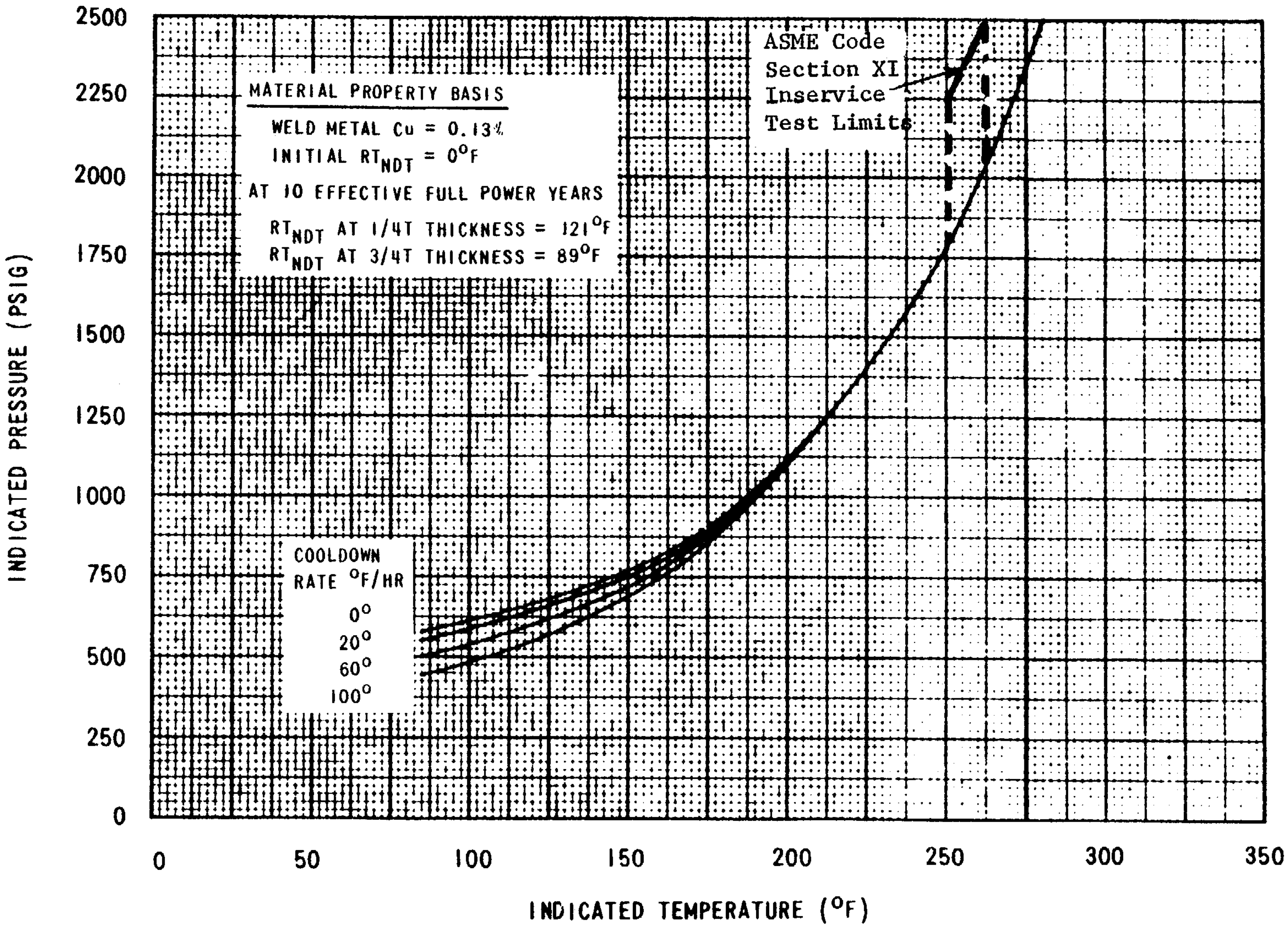
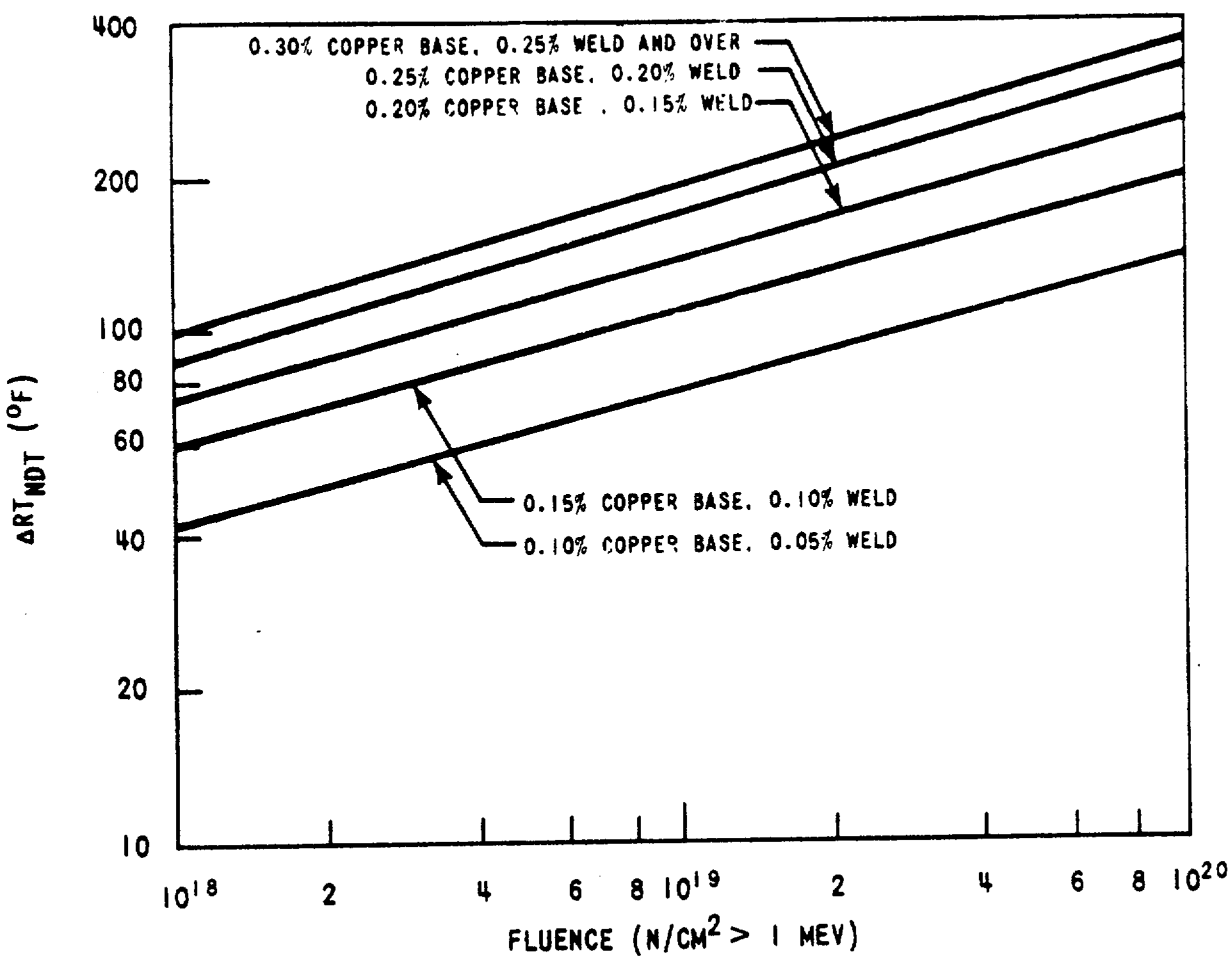


FIGURE TS.3.1-2

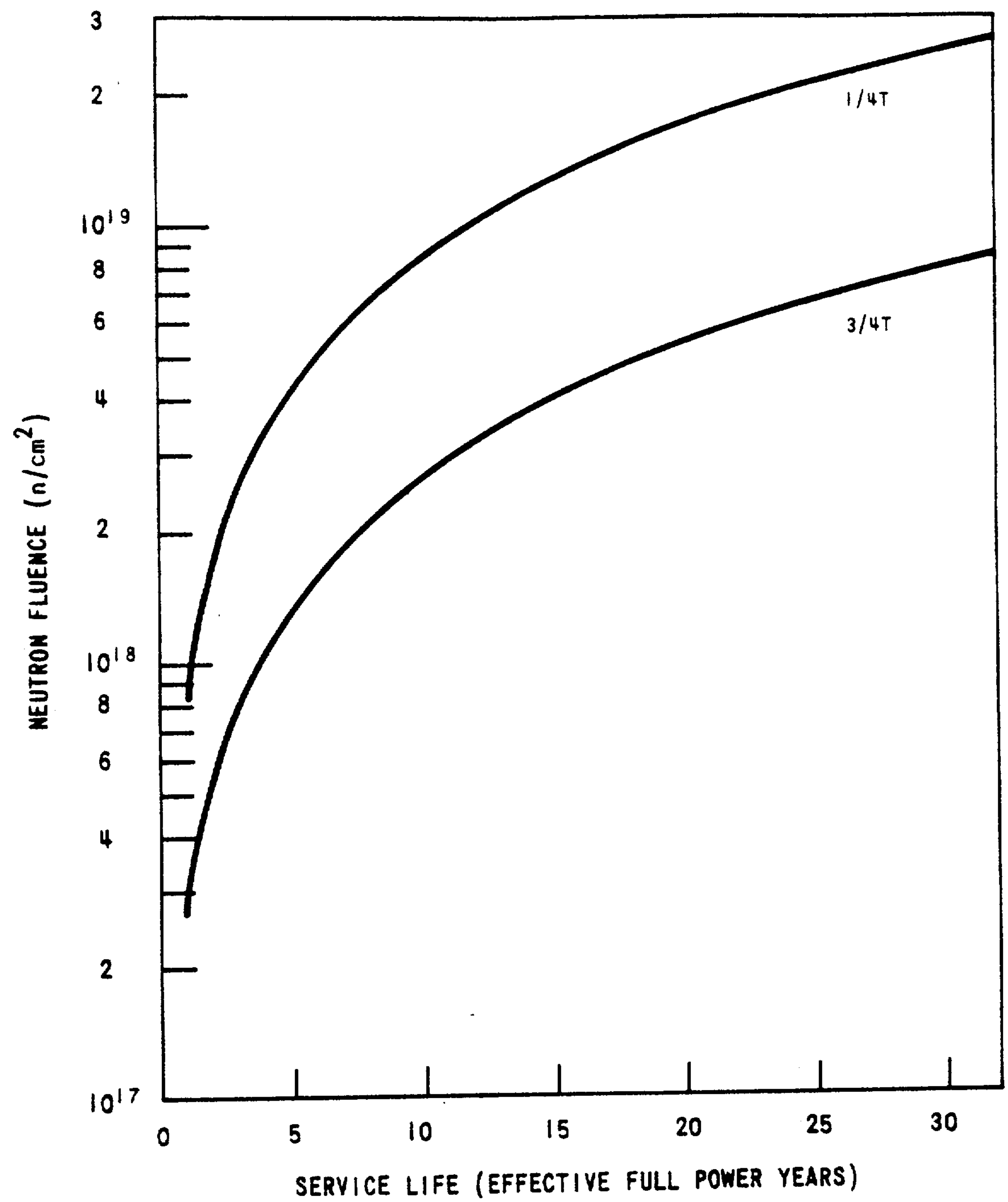
UNIT 1 AND UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(Applicable for First 10 EFPY of Operation)





EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO 550°F TEMPERATURE

FIGURE TS.3.1-3



FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION
OF FULL POWER SERVICE LIFE

FIGURE TS.3.1-4

2. If the percentage quandrant power tilt exceeds 2% but is less than 7% for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for physics tests if the quadrant power tilt ratio exceeds 1.07, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel out of service, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. Except during low power physics testing, operation with part length rods shall be restricted such that the part length rod bank is not inserted in the reactor core at any time the reactor is critical.[2]

3. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion; insertion limits are shown in Figure TS.3.10-2, 3 and 4 for normal and abnormal operating conditions.

4. Control bank insertion may be further restricted by Specification 3.10.A if, (1) the measured control rod worth of all rods, less the worth of the worst stuck rod, is less than 5.52% reactivity at the beginning of the first cycle or the equivalent value if measured at any other time, or (2) if a rod is inoperable (Specification 3.10.G).

5. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for the low power margin. For this test the reactor may be critical with all but one high worth full-length control rod inserted and all part-length rods fully withdrawn for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power physics test.[3]

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[2] This limiting condition for operation as pertains to the part length rods is no longer applicable for a particular Prairie Island unit following removal of the part length rods from that unit.

[3] See footnote 2.

E. Rod Misalignment Limitations

1. If a full-length or part-length rod cluster control (RCC) assembly is misaligned from its bank by more than 15 inches, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.[4]

2. If the misaligned rod cluster control is not realigned within a total of 8 hours, the rod shall be declared inoperable.

F. Inoperable Rod Position Indicator Channels

1. If a rod position indicator (RPI) channel is out of service then

   a. For operation between 50% and 100% of rating, the position of the RCC shall be checked directly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to rod motion exceeding a total of 24 steps, whichever occurs first.

   b. During operation below 50% of rating, no special monitoring is required.

2. The plant shall be brought to the hot shutdown condition should more than one RPI channel per sub-bank or more than two RPI channels per bank be found to be inoperable during power operation.

3. If a full length or part length rod having a rod position indicator channel out of service is found to be misaligned from 1.a. above, then Specification 3.10 E. will be applied.[5]

G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under Specification 3.10 E. or 3.10 H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

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[4] See footnote 2, page TS.3.10-4.

[5] See footnote 2, page TS.3.10-4.

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods.(1)

2. The average enrichment of the initial core is a nominal 2.90 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight per cent of U-235.(2)

3. Reload fuel will be similar in design to the initial core.

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12, and 16-rod clusters, which are located in vacant rod cluster control guide tubes.(3) The burnable poison rods consist of borosilicate glass clad with stainless steel.(4)

5. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel.(5)

6. Up to 10 grams of enriched fissionable material may either be used in the core or be available on the plant site in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements.(6)

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 32 AND 26 TO FACILITY

LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

By letters dated December 9, 1977 and September 8, 1978 Northern States Power Company (the licensee) requested amendment of the Technical Specifications appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (PINGP). The proposed amendments would reduce the maximum pressurizer heatup rate from 200°F per hour to 100°F per hour, change the pressure-temperature operating limits and permit removal of the part length rods.

I. PRESSURIZER HEATUP RATE AND PRESSURE-TEMPERATURE LIMITS

Background

In August 1977, Mitsubishi Heavy Industries, Ltd., of Japan noted an inconsistency in the pressurizer heatup rate stated in their Technical Specifications. Specification 3.4.9 required a heatup rate of 200°F/hr; Specification 5.7.1, however, required a heatup rate of 100°F/hr. This discrepancy was reported to the Westinghouse Electric Corporation (Westinghouse), who then reviewed their analysis of the pressurizer heatup rate and determined that the correct heatup rate is 100°F/hr, and that the correct cooldown rate is 200°F/hr; the Technical Specifications for the PINGP stated that pressurizer heatup and cooldown rates were 200°F/hr. Westinghouse then notified the Nuclear Regulatory Commission (the Commission) and the licensee of this problem. The requested amendment would correct the error in the pressurizer heatup rate limit.

Evaluation

In designing the pressurizer, Westinghouse performed a thermal stress analysis which analyzed the fatigue resulting from a heatup rate of 100°F/hr and a cooldown rate of 200°F/hr. This analysis meets the standards of the ASME Code, Section III, which requires that the analysis be based on a usage factor. The usage factor represents the fraction of the fatigue life (the total amount of stress that a particular component is designed to handle), with a usage factor of zero implying that no stress has been exerted on the component, and a usage factor of one implying that the stress exerted on the component is equal to the amount of stress that the component is designed to handle. For any piece of equipment, certain components receive more stress than others. For the pressurizer, this component is the surge nozzle, which has a usage factor of 0.9 for the design numbers listed above. This usage factor is such that if the heatup and cooldown rates used in the analysis were exceeded more than a few times, the actual usage factor for the surge nozzle would exceed 1.0, which is not allowable under the ASME Code. Thus, we conclude that reducing the heatup rate limit from 200°F/hr to 100°F/hr is necessary to maintain thermal stresses in the pressurizer to allowable levels. For the same reasons, we further conclude that the cooldown rate limit presently listed in the Technical Specifications is adequate.

Because the current Technical Specifications provision authorized higher rates of pressurizer heatup than the correct limit, the question arose as to whether the correct limit of 100°F/hr has been exceeded in the past. Discussions with Westinghouse indicate that this is unlikely. This is because system capabilities and Technical Specification limits on the rate of reactor coolant system heatup and pressurization effectively preclude pressurizer heatup rates in excess of 50°F to 75°F/hr. Furthermore, the licensee reviewed operating records for the PINGP and found that the 100°F limit had never been exceeded. System design precludes heatup rates in excess of 70°F/hr. Accordingly, we conclude that the only action required by the licensee is modification of the Technical Specifications to reduce the limiting pressurizer heatup rate of 200°F/hr to 100°F/hr.

The Staff has corresponded with Westinghouse on this matter and has received confirmation from the company that it is performing a review of the stress analyses for components of the reactor coolant pressure boundary to assure that no similar inadvertent error appears in any other portion of the applicable Technical Specifications. This action will be confirmed by Westinghouse.

II. REACTOR PRESSURE VESSEL

In its letter dated December 9, 1977, the licensee also submitted a request to change the Technical Specifications of PINGP regarding the reactor pressure vessel pressure-temperature operating limits. The proposed operating limit curves are based on the test results obtained from material surveillance Capsule V that was removed from the Unit 1 reactor vessel.* These revised operating limits are proposed for operation through 10 effective full power years (EFPY).

We have reviewed the proposed operating limits and the test results from Capsule V. The capsule received an average neutron fluence of 5.21 x 10^{18} n/cm^2. Weld metal and base metal showed increases in RT_{NDT} at this fluence of 32 and 15°F respectively. There was no measurable drop in the upper shelf energy.

The reactor vessels for Unit 1 and Unit 2 are both made from A508 Class 3 forging material having very low copper content; below 0.1%. As indicated by the test results from the surveillance program, this material has excellent resistance to radiation damage. The limiting material is the weld metal from Unit 1 having a copper content of 0.14% and a phosphorus content of 0.016%. Using the above material as the limiting material for both units, we made independent calculations for pressure-temperature limits through 10 EFPY. We found that the operating limits proposed by the licensee conform to those calculated by us and are in accordance with Appendix G, 10 CFR Part 50. Such compliance with Appendix G in establishing safety operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Based on the above, we conclude that the proposed operating limits are acceptable for operation through 10 EFPY.

---

*Analysis of Capsule V from NSP Prairie Island Unit No. 1 Reactor Vessel Surveillance Program WCAP-8916, transmitted by NSP letter dated September 12, 1977

III. PART LENGTH RODS

Discussion and Evaluation

Part length control rods were initially installed to suppress xenon induced power oscillations in the axial direction, should such oscillations occur. They were also intended for use in axial offset calibration tests or low power physics tests.

The Technical Specifications, as now written, require that these part length rod cluster control assemblies (PLRCCAs) be withdrawn and excluded from the core at all times during reactor operations. The PLRCCAs are not needed, used or assumed to be available to achieve required reactor shutdown conditions. The proposed removal, therefore, will not cause any change in required reactivity characteristics, or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part length rods dropping into the core during operation. Such an event could cause abnormal flux distribution or reactor shutdown.

In addition, in order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCAs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the springs, are construced from type 304 stainless steel. The springs are wound from Inconel x-750 for corrosion resistance and high strength.

These thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCAs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that (1) the PLRCCAs are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution reactor shutdown, and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 1, 1978

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 32 and 26 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

These amendments change the Technical Specifications that relate to the pressurizer heatup rate, the reactor vessel pressure-temperature operating limits, and the part length rods.

The application for amendments 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the applications for amendments dated December 9, 1977 and September 8, 1978, (2) Amendment Nos. 32 and 26 to License Nos. DPR-42 and DPR-60, respectively, 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 of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single 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 Operating Reactors.

Dated at Bethesda, Maryland, this 1st day of November, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

NOV 1 1978

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Docket Files 50-282
and 50-306
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Clare Miles, O
R. Diggs
J. Carter

Docket Nos. 50-282
and 50-306

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

Gentlemen:

In response to your applications dated December 9, 1977 and September 8, 1978, the Commission has issued the enclosed Amendment Nos. 32 and 26 to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant Unit Nos. 1 and 2, respectively.

The amendments change the Technical Specifications that relate to the pressurizer heatup rate, the reactor vessel pressure-temperature operating limits, and the part length rods. The Table of Contents of the Technical Specifications has also been updated.

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

Sincerely,

ORIGINAL SIGNED

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment Nos. 32 and 26 to License Nos. DPR-42 and DPR-60
2. Safety Evaluation
3. Notice of Issuance

SEP 2+3

OELD concurs with changes discussed with M. Grotenhuis
ESFriedel

OFFICE	DOR:ORB1	DOR:ORB1	DOR:ORB1	DOR:RSB	OELD	DOR:ORB1
SURNAME	CParrish:jd	MGrotenhuis	DNeighbors	JGianelli		ASchwencer
DATE	10/11/78	10/11/78	10/11/78	10/11/78	10/27/78	11/1/78

481-12

NRC FORM 318 (9-76) NRCM 0240 ★ U.S. GOVERNMENT PRINTING OFFICE: 1976 - 265 - 769

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 32
License No. DPR-42

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

   A. The applications for amendment by Northern States Power Company (the licensee) dated December 9, 1977 and September 8, 1978, comply 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 License No. DPR-42 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 32 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical Specifications

Date of Issuance: November 1, 1978

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 26
License No. DPR-60

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

   A. The applications for amendment by Northern States Power Company (the licensee) dated December 9, 1977 and September 8, 1978, comply 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 License No. DPR-60 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 26 , 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

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 1, 1978

ATTACHMENT TO LICENSE AMENDMENT NOS. 32 AND 26

FACILITY OPERATING LICENSE NOS. DPR-42 AND DPR-60

DOCKET NOS. 50-282 AND 50-306

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated licenses with the attached pages bearing the same numbers, except as otherwise indicated. The changed areas on the revised pages are reflected by a marginal line.

Remove	Insert
ii	ii
iii	iii
iv	iv
TABLE TS.3.1-1	TABLE TS.3.1-1
TABLE TS.3.1-2	TABLE TS.3.1-2
FIGURE TS.3.1-1	FIGURE TS.3.1-1
FIGURE TS.3.1-2	FIGURE TS.3.1-2
FIGURE TS.3.1-3	FIGURE TS.3.1-3
FIGURE TS.3.1-4	FIGURE TS.3.1-4
TS 3.10-4	TS 3.10-4
TS 3.10-5	TS 3.10-5
TS 5.3-1	TS 5.3-1

APPENDIX A TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS (Continued)

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF TABLES

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF FIGURES

B. HEATUP AND COOLDOWN

Specification:

1. The Unit 1 and Unit 2 reactor coolant temperature and pressure and system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figures TS.3.1-1 and TS.3.1-2 for the first full power service period.

   a. Allowable combinations of pressure and temperature for specific temperature change rates are below and to the right of the limit lines shown. Limit lines for cooldown rates between those presented may be obtained by interpolation.

   b. Figures TS.3.1-1 and TS.3.1-2 define limits to assure prevention of non-ductile failure only. For normal operation other inherent plant characteristics, e.g., pump heat addition and pressurizer heater capacity may limit the heatup and cooldown rates that can be achieved over certain pressure-temperature ranges.

2. The limit lines shown in Figures TS.3.1-1, TS.3.1-2 shall be recalculated periodically using methods discussed in the Basis section.

3. The secondary side of the steam generator must not be pressurized above 200 psig if the temperature of the vessel is below 70°F.

4. The pressurizer heatup rate shall not exceed 100°F/hr and the pressurizer cooldown rate shall not exceed 200°F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

Basis

The reactor coolant system heatup and cooldown rates in Figures TS.3.1-1 and TS.3.1-2 are applicable to both Unit 1 and Unit 2. The curves are based on Unit 1 toughness data and are conservative for the Unit 2 vessel. Toughness data is included in Tables TS.3.1-1 and TS.3.1-2.

Table TS 3.1-1

UNIT NO. 1

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	Transverse[a] 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RT_{NDT} (°F)	Average Transverse[a] Upper Shelf (ft lb)
Closure Head Dome	A533 Gr. B, Cl. 1			-4	64[c]	4[c]	75 [c]
Head Flange	A508 Cl. 3			-4	12[c]	-4[c]	84 [c]
Vessel Flange	A508 Cl. 3			-4	41[c]	-4[c]	77.5 [c]
Injection Nozzles	A508 Cl. 3			-22	-114[c]	-22 [c]	97 [c]
Inlet and Outlet Nozzle	A508 Cl. 3			+5	39[c]	5 [c]	92 [c]
Upper Shell	A508 Cl. 3			-4	39[c]	-4 [c]	85 [c]
Inter. Shell[b]	A508 Cl. 3	0.06	0.013	+14	14	14	143
Lower Shell[b]	A508 Cl. 3	0.07	0.014	-4	45	-4	134
Trans. Ring	A508 Cl. 3			+5	63[c]	5	79 [c]
Bottom Head	A533 Gr. B, Cl. 1			-4	57[c]	-3	68.5 [c]
Weldment[b]	Weld	0.13	0.017	0	10	0	78.5
HAZ[b]	HAZ			0 [c]	<-100	0	211

a. Specimen oriented normal to the major working direction

b. Based on actual transverse data through the surveillance program

c. Estimated using "Pressure-Temperature Limits," Section 5.3.2 of *Standard Review Plan*, NUREG-75/087, 1975, from longitudinal data.

TABLE TS.3.1-1

TABLE TS.3.1-2

UNIT NO. 2

REACTOR VESSEL TOUGHNESS DATA (UNIRRADIATED)

Component	Material Type	Cu (%)	P (%)	NDTT (°F)	Transverse[a] 50 ft lb/35 mils Lateral Expansion Temp. (°F)	RT_{NDT} (°F)	Average Transverse[a] Upper Shelf (ft lb)
Closure Head Dome	A533 Gr. B, Cl. 1			5	52[c]	5	64[c]
Head Flange	A508 Cl. 3			-31	18[c]	-31	87[c]
Vessel Flange	A508 Cl. 3			-22	18[c]	-22	88[c]
Injection Nozzles	A508 Cl. 3			-22	-114[c]	-22	97[c]
Inlet and Outlet Nozzle	A508 Cl. 3			-13	50[c]	-10	89[c]
Upper Shell	A508 Cl. 3			-13	41[c]	-13	85[c]
Inter. Shell[b]	A508 Cl. 3	0.075	0.010	-4	56	-4	112
Lower Shell[b]	A508 Cl. 3	0.085	0.011	-13	54	-6	108
Trans. Ring	A508 Cl. 3			10	50	10	76[c]
Bottom Head	A533 Gr. B, Cl. 1			-13	56	-4	68[c]
Weldment[b]	Weld	0.082	0.019	-31	-6	-31	103
HAZ[b]	HAZ			-31	-35	-31	117

a. Specimen oriented normal to the major working direction

b. Based on actual transverse data through the surveillance program

c. Estimated using "Pressure-Temperature Limits," Section 5.3.2 of *Standard Review Plan*, NUREG-75/087, 1975, from longitudinal data.

FIGURE TS.3.1-1

UNIT 1 AND UNIT 2 REACTOR COOLANT SYSTEM HEATUP LIMITATIONS
(Applicable for First 10 EFPY of Operation)

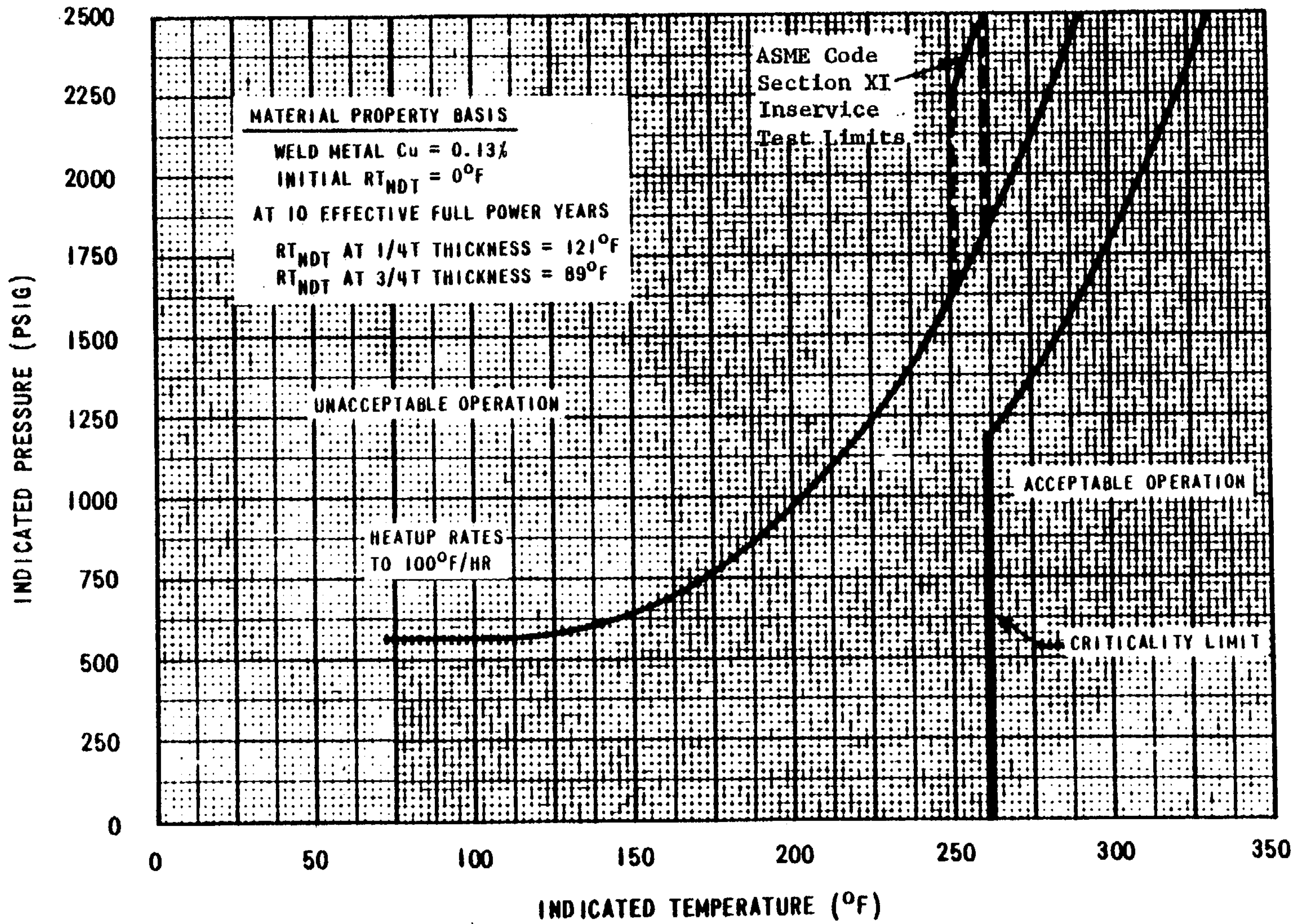
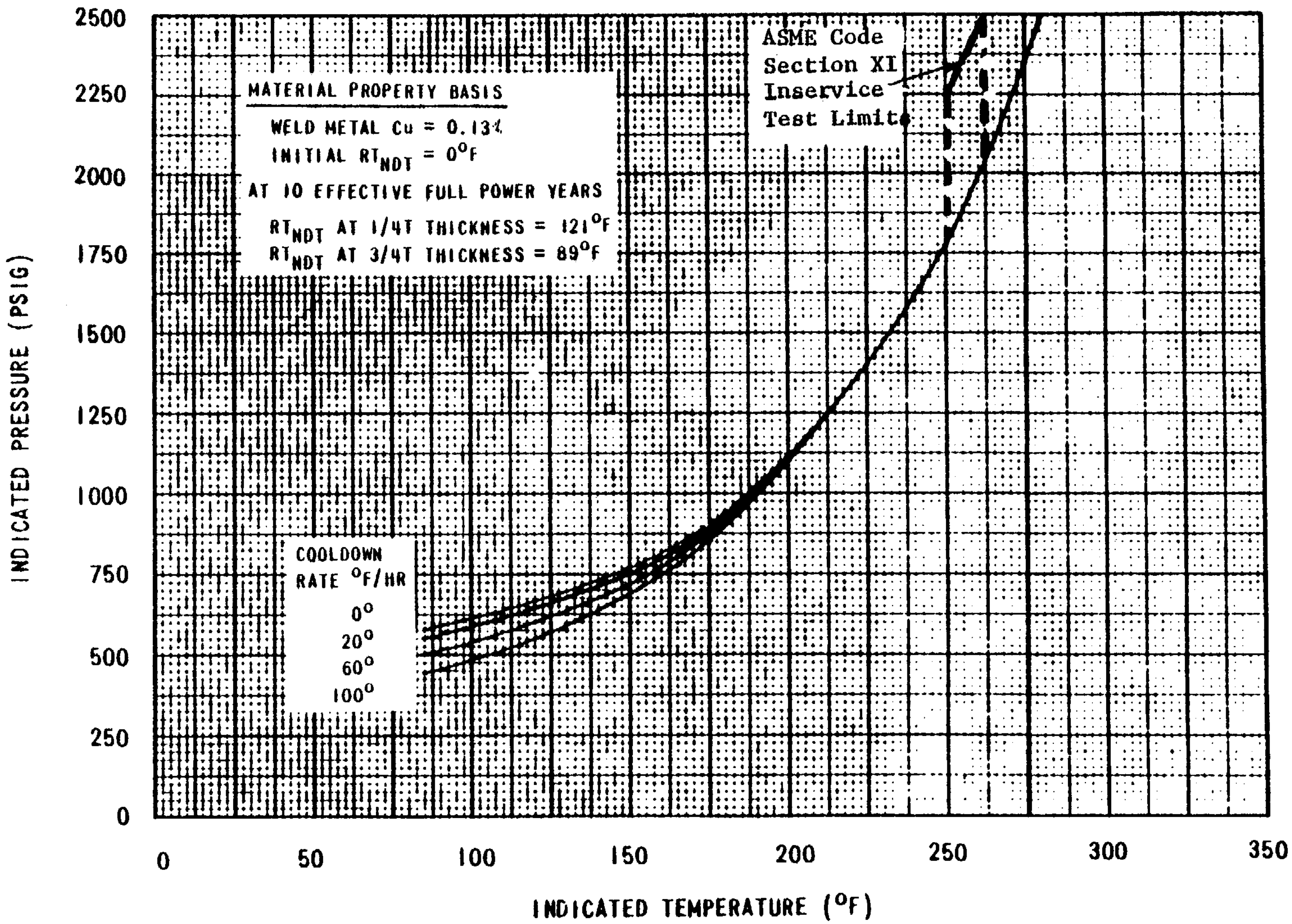


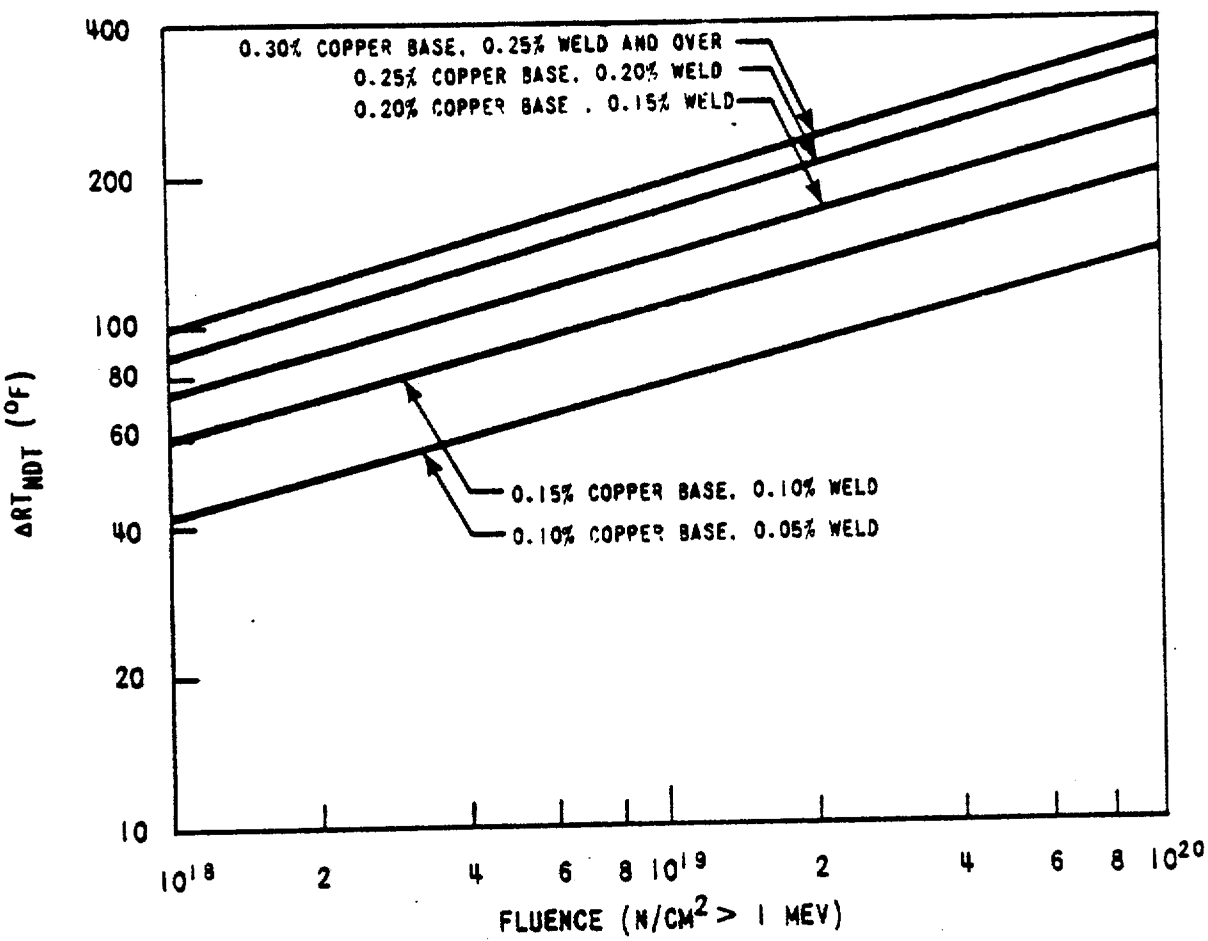
FIGURE TS.3.1-2

UNIT 1 AND UNIT 2 REACTOR COOLANT SYSTEM COOLDOWN LIMITATIONS
(Applicable for First 10 EFPY of Operation)



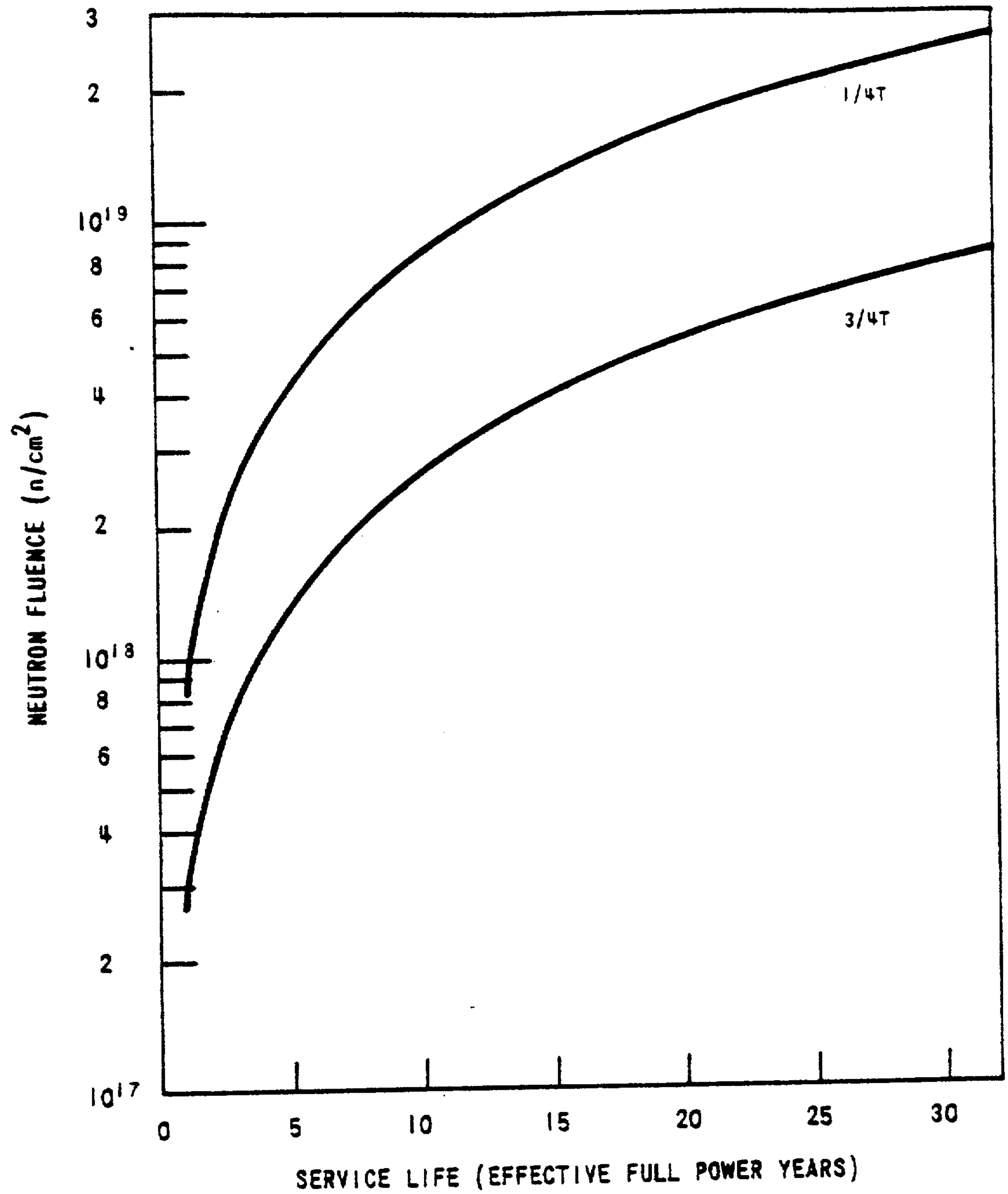
Amendment No. 32 & 26

FIGURE TS.3.1-2



EFFECT OF FLUENCE AND COPPER CONTENT ON SHIFT OF RT_{NDT} FOR REACTOR VESSEL STEELS EXPOSED TO 550°F TEMPERATURE

FIGURE TS.3.1-3



FAST NEUTRON FLUENCE (E>1MeV) AS A FUNCTION
OF FULL POWER SERVICE LIFE

FIGURE TS.3.1-4

2. If the percentage quadrant power tilt exceeds 2% but is less than 7% for a sustained period of more than 24 hours, or if such a tilt recurs intermittently, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

3. Except for physics tests if the quadrant power tilt ratio exceeds 1.07, the reactor shall be brought to the hot shutdown condition. Subsequent operation below 50% of rating, for testing, shall be permitted.

4. If the core is operating above 85% power with one excore nuclear channel out of service, then the core quadrant power balance shall be determined daily and after a 10% power change using either 2 movable detectors or 4 core thermocouples per quadrant, per Specification 3.11.

D. Rod Insertion Limits

1. The shutdown rods shall be fully withdrawn when the reactor is critical or approaching criticality.

2. Except during low power physics testing, operation with part length rods shall be restricted such that the part length rod bank is not inserted in the reactor core at any time the reactor is critical.[2]

3. When the reactor is critical or approaching criticality, the control banks shall be limited in physical insertion; insertion limits are shown in Figure TS.3.10-2, 3 and 4 for normal and abnormal operating conditions.

4. Control bank insertion may be further restricted by Specification 3.10.A if, (1) the measured control rod worth of all rods, less the worth of the worst stuck rod, is less than 5.52% reactivity at the beginning of the first cycle or the equivalent value if measured at any other time, or (2) if a rod is inoperable (Specification 3.10.G).

5. Insertion limits do not apply during physics tests or during periodic exercise of individual rods. The shutdown margin shown in Figure TS.3.10-1 must be maintained except for the low power margin. For this test the reactor may be critical with all but one high worth full-length control rod inserted and all part-length rods fully withdrawn for a period not to exceed 2 hours per year provided a rod drop test is run on the high worth full-length rod prior to this particular low power physics test.[3]

---

[2] This limiting condition for operation as pertains to the part length rods is no longer applicable for a particular Prairie Island unit following removal of the part length rods from that unit.

[3] See footnote 2.

E. Rod Misalignment Limitations

1. If a full-length or part-length rod cluster control (RCC) assembly is misaligned from its bank by more than 15 inches, the rod will be realigned or the core power peaking factors shall be determined within 2 hours, and Specification 3.10.B applied. If peaking factors are not determined within 2 hours, the high neutron flux trip setpoint shall be reduced to 85 percent of rating.[4]

2. If the misaligned rod cluster control is not realigned within a total of 8 hours, the rod shall be declared inoperable.

F. Inoperable Rod Position Indicator Channels

1. If a rod position indicator (RPI) channel is out of service then

   a. For operation between 50% and 100% of rating, the position of the RCC shall be checked directly by core instrumentation (excore detector and/or thermocouples and/or movable incore detectors) every shift or subsequent to rod motion exceeding a total of 24 steps, whichever occurs first.

   b. During operation below 50% of rating, no special monitoring is required.

2. The plant shall be brought to the hot shutdown condition should more than one RPI channel per sub-bank or more than two RPI channels per bank be found to be inoperable during power operation.

3. If a full length or part length rod having a rod position indicator channel out of service is found to be misaligned from 1.a. above, then Specification 3.10 E. will be applied.[5]

G. Inoperable Rod Limitations

1. An inoperable rod is a rod which (a) does not trip, (b) is declared inoperable under Specification 3.10 E. or 3.10 H. or (c) cannot be moved by its drive mechanism and cannot be corrected within 8 hours.

---

[4] See footnote 2, page TS.3.10-4.

[5] See footnote 2, page TS.3.10-4.

5.3 REACTOR

A. Reactor Core

1. The reactor core contains approximately 48 metric tons of uranium in the form of slightly enriched uranium dioxide pellets. The pellets are encapsulated in Zircaloy-4 tubing to form fuel rods. The reactor core is made up of 121 fuel assemblies. Each fuel assembly contains 179 fuel rods.[1]

2. The average enrichment of the initial core is a nominal 2.90 weight per cent of U-235. Three fuel enrichments are used in the initial core. The highest enrichment is a nominal 3.40 weight per cent of U-235.[2]

3. Reload fuel will be similar in design to the initial core.

4. Burnable poison rods are incorporated in the initial core. There are 704 poison rods in the form of 8, 12, and 16-rod clusters, which are located in vacant rod cluster control guide tubes.[3] The burnable poison rods consist of borosilicate glass clad with stainless steel.[4]

5. In the reactor core, there are 29 full-length RCC assemblies that contain a 142-inch length of silver-indium-cadmium alloy clad with stainless steel.[5]

6. Up to 10 grams of enriched fissionable material may either be used in the core or be available on the plant site in the form of fabricated neutron flux detectors for the purposes of monitoring core neutron flux.

B. Reactor Coolant System

1. The design of the reactor coolant system complies with all applicable code requirements.[6]

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NOS. 32 AND 26 TO FACILITY

LICENSE NOS. DPR-42 AND DPR-60

NORTHERN STATES POWER COMPANY

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NOS. 1 AND 2

DOCKET NOS. 50-282 AND 50-306

Introduction

By letters dated December 9, 1977 and September 8, 1978 Northern States Power Company (the licensee) requested amendment of the Technical Specifications appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (PINGP). The proposed amendments would reduce the maximum pressurizer heatup rate from 200°F per hour to 100°F per hour, change the pressure-temperature operating limits and permit removal of the part length rods.

I. PRESSURIZER HEATUP RATE AND PRESSURE-TEMPERATURE LIMITS

Background

In August 1977, Mitsubishi Heavy Industries, Ltd., of Japan noted an inconsistency in the pressurizer heatup rate stated in their Technical Specifications. Specification 3.4.9 required a heatup rate of 200°F/hr; Specification 5.7.1, however, required a heatup rate of 100°F/hr. This discrepancy was reported to the Westinghouse Electric Corporation (Westinghouse), who then reviewed their analysis of the pressurizer heatup rate and determined that the correct heatup rate is 100°F/hr, and that the correct cooldown rate is 200°F/hr; the Technical Specifications for the PINGP stated that pressurizer heatup and cool-down rates were 200°F/hr. Westinghouse then notified the Nuclear Regulatory Commission (the Commission) and the licensee of this problem. The requested amendment would correct the error in the pressurizer heatup rate limit.

Evaluation

In designing the pressurizer, Westinghouse performed a thermal stress analysis which analyzed the fatigue resulting from a heatup rate of 100°F/hr and a cooldown rate of 200°F/hr. This analysis meets the standards of the ASME Code, Section III, which requires that the analysis be based on a usage factor. The usage factor represents the fraction of the fatigue life (the total amount of stress that a particular component is designed to handle), with a usage factor of zero implying that no stress has been exerted on the component, and a usage factor of one implying that the stress exerted on the component is equal to the amount of stress that the component is designed to handle. For any piece of equipment, certain components receive more stress than others. For the pressurizer, this component is the surge nozzle, which has a usage factor of 0.9 for the design numbers listed above. This usage factor is such that if the heatup and cooldown rates used in the analysis were exceeded more than a few times, the actual usage factor for the surge nozzle would exceed 1.0, which is not allowable under the ASME Code. Thus, we conclude that reducing the heatup rate limit from 200°F/hr to 100°F/hr is necessary to maintain thermal stresses in the pressurizer to allowable levels. For the same reasons, we further conclude that the cooldown rate limit presently listed in the Technical Specifications is adequate.

Because the current Technical Specifications provision authorized higher rates of pressurizer heatup than the correct limit, the question arose as to whether the correct limit of 100°F/hr has been exceeded in the past. Discussions with Westinghouse indicate that this is unlikely. This is because system capabilities and Technical Specification limits on the rate of reactor coolant system heatup and pressurization effectively preclude pressurizer heatup rates in excess of 50°F to 75°F/hr. Furthermore, the licensee reviewed operating records for the PINGP and found that the 100°F limit had never been exceeded. System design precludes heatup rates in excess of 70°F/hr. Accordingly, we conclude that the only action required by the licensee is modification of the Technical Specifications to reduce the limiting pressurizer heatup rate of 200°F/hr to 100°F/hr.

The Staff has corresponded with Westinghouse on this matter and has received confirmation from the company that it is performing a review of the stress analyses for components of the reactor coolant pressure boundary to assure that no similar inadvertent error appears in any other portion of the applicable Technical Specifications. This action will be confirmed by Westinghouse.

II. REACTOR PRESSURE VESSEL

In its letter dated December 9, 1977, the licensee also submitted a request to change the Technical Specifications of PINGP regarding the reactor pressure vessel pressure-temperature operating limits. The proposed operating limit curves are based on the test results obtained from material surveillance Capsule V that was removed from the Unit 1 reactor vessel.* These revised operating limits are proposed for operation through 10 effective full power years (EFPY).

We have reviewed the proposed operating limits and the test results from Capsule V. The capsule received an average neutron fluence of 5.21 x 10^{18} n/cm^2. Weld metal and base metal showed increases in RT_{NDT} at this fluence of 32 and 15°F respectively. There was no measurable drop in the upper shelf energy.

The reactor vessels for Unit 1 and Unit 2 are both made from A508 Class 3 forging material having very low copper content; below 0.1%. As indicated by the test results from the surveillance program, this material has excellent resistance to radiation damage. The limiting material is the weld metal from Unit 1 having a copper content of 0.14% and a phosphorus content of 0.016%. Using the above material as the limiting material for both units, we made independent calculations for pressure-temperature limits through 10 EFPY. We found that the operating limits proposed by the licensee conform to those calculated by us and are in accordance with Appendix G, 10 CFR Part 50. Such compliance with Appendix G in establishing safety operating limitations will ensure adequate safety margins during operation, testing, maintenance and postulated accident conditions and constitutes an acceptable basis for satisfying the requirements of NRC General Design Criterion 31, Appendix A, 10 CFR Part 50.

Based on the above, we conclude that the proposed operating limits are acceptable for operation through 10 EFPY.

*Analysis of Capsule V from NSP Prairie Island Unit No. 1 Reactor Vessel Surveillance Program WCAP-8916, transmitted by NSP letter dated September 12, 1977

III. PART LENGTH RODS

Discussion and Evaluation

Part length control rods were initially installed to suppress xenon induced power oscillations in the axial direction, should such oscillations occur. They were also intended for use in axial offset calibration tests or low power physics tests.

The Technical Specifications, as now written, require that these part length rod cluster control assemblies (PLRCCAs) be withdrawn and excluded from the core at all times during reactor operations. The PLRCCAs are not needed, used or assumed to be available to achieve required reactor shutdown conditions. The proposed removal, therefore, will not cause any change in required reactivity characteristics, or safety margins at full power, low power or shutdown. To the contrary, removal will eliminate the potential for part length rods dropping into the core during operation. Such an event could cause abnormal flux distribution or reactor shutdown.

In addition, in order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRCCAs were to be performed, the licensee proposes to install thimble plug assemblies in the spaces previously occupied by PLRCCAs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a pin welded to the nut.

All components in the thimble plug assembly, except for the springs, are construced from type 304 stainless steel. The springs are wound from Inconel x-750 for corrosion resistance and high strength.

These thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRCCAs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that (1) the PLRCCAs are not needed for reactor operation, (2) that removal of these assemblies will remove the chance for an abnormal flux distribution reactor shutdown, and (3) that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

Environmental Consideration

We have determined that the amendments do 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 amendments involve 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 these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 1, 1978

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment Nos. 32 and 26 to Facility Operating License Nos. DPR-42 and DPR-60, issued to the Northern States Power Company (the licensee), which revised Technical Specifications for operation of Unit Nos. 1 and 2 of the Prairie Island Nuclear Generating Plant (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of their date of issuance.

These amendments change the Technical Specifications that relate to the pressurizer heatup rate, the reactor vessel pressure-temperature operating limits, and the part length rods.

The application for amendments 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 amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

The Commission has determined that the issuance of these amendments 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 these amendments.

For further details with respect to this action, see (1) the applications for amendments dated December 9, 1977 and September 8, 1978, (2) Amendment Nos. 32 and 26 to License Nos. DPR-42 and DPR-60, respectively, 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 of the Minneapolis Public Library, 300 Nicollet Mall, Minneapolis, Minnesota 55401. A single 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 Operating Reactors.

Dated at Bethesda, Maryland, this 1st day of November, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors