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Docket Nos. 50-282
and 50-306

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 Amendments Nos. 50 and to 44 Facility Operating Licenses Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments are in response to your application dated August 14, 1981.

The amendments revise the Appendix A Technical Specifications by clarifying snubber surveillance requirements. The amendments also correct some administrative errors which you called to our attention in your letter of August 24, 1981.

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

Sincerely,

Original signed by:

Dominic DiIanni, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 50 to DPR-42
2. Amendment No. 44 to DPR-60
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

RECEIVED OCT 29 1981 U.S. NUCLEAR REGULATORY COMMISSION

MB - no objection on legal basis to FR notice or to amendment

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OFFICE	ORB#3:DL	ORB#3:DL	ORB#3:DL	ORAB	ORB#3:DL	AD/OR:DL	OELD
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DATE	10/22/81	10/20/81	10/23/81	10/22/81	10/ /81	10/27/81	10/26/81

Northern States Power Company

cc:

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 50
License No. DPR-42

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

   A. The application for amendment by Northern States Power Company (the licensee) dated August 14, 1981, 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-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. 50 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: October 28, 1981

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. 44
License No. DPR-60

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

   A. The application for amendment by Northern States Power Company (the licensee) dated August 14, 1981, 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-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. 44 , 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

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 28, 1981

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-42

AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NOS. 50-282 AND 50-306

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

TS-i
TS-iii
Table TS.3.5-6
TS.3.12-1
Table TS.3.12-1, (pg 1 of 8)
Table TS.3.12-1, (pg 2 of 8)
Table TS.3.12-1, (pg 3 of 8)
Table TS.3.12-1, (pg 4 of 8)
Table TS.3.12-1, (pg 5 of 8)
Table TS.3.12-1, (pg 6 of 8)
Table TS.3.12-1, (pg 7 of 8)
TS.4.13-1
TS.4.13-2
TS.4.13-3
TS.4.13-4 (new)
Table TS.4.1-1 (pg 1 of 5)
Table TS.4.1-1 (pg 5 of 5)
TS.6.6-2

APPENDIX A TECHNICAL SPECIFICATIONS

TABLE OF CONTENTS

APPENDIX A TECHNICAL SPECIFICATIONS

LIST OF TABLES

TABLE TS.3.5-6

INSTRUMENT OPERATING CONDITIONS FOR AUXILIARY ELECTRICAL SYSTEM

FUNCTIONAL UNIT	1 MINIMUM OPERABLE CHANNELS	2 MINIMUM DEGREE OF REDUNDANCY	3 PERMISSIBLE BYPASS CONDITIONS	4 OPERATOR ACTION IF CONDITIONS OF COLUMN 1 OR 2 CANNOT BE MET
1. Degraded Voltage 4KV Safeguards Busses	1/Bus	1/Bus	---	Place inoperable channel in the tripped condition within one hour or be in hot shutdown.***
2. a. Loss of voltage 4KV Safeguard Bus (90%)	1/Bus	1/Bus	---	Place inoperable channel in the tripped condition within one hour or be in hot shutdown.***
b. Loss of voltage 4KV Safeguard Bus (55%)	1/Bus	1/Bus	---	Place inoperable channel in the tripped condition within one hour or be in hot shutdown.***

***If minimum conditions are not met within 24 hours, steps shall be taken to place the unit in cold shutdown conditions.

3.12 SNUBBERS

Applicability

Applies to the operability of safety related snubbers.

Objective

To define those conditions of snubber operability necessary to assure safe reactor operation.

Specification

A. Except as permitted below, all snubbers listed in Table TS.3.12-1 shall be operable above Cold Shutdown. Snubbers may be inoperable in Cold Shutdown and Refueling Shutdown whenever the supported system is not required to be Operable.

B. With one or more snubbers made or found to be inoperable for any reason when Operability is required, within 72 hours:

   1. Replace or restore the inoperable snubbers to Operable status and perform an engineering evaluation per Specification 4.13.E on the supported component(s), or

   2. Declare the supported system inoperable and take the action required by the Technical Specifications for inoperability of that system.

C. Snubber modifications may be made to safety related systems without prior License Amendment to Table TS.3.12-1 provided that a revision to the Table is included with the next License Amendment Request.

Basis

All snubbers are required to be Operable above Cold Shutdown to ensure that the structural integrity of the reactor coolant system and all other safety related systems is maintained during and following a seismic or other event initiating dynamic loads.

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Area During Shutdown
	UNIT I				
AFSH-22 A&B	Main and Aux-	773'-4-1/4"	A		
AFSH-36	iliary Steam	745'-7-1/4"	A		
AFSH-39		699'-10-1/4"	A		
AFSH-48		699'-6-1/4""	A		
MSDH-25 A&B		736'-6-7/16"	A		
MSDH-26 A&B		756'-7-1/4"	A		
MSDH-29		756'-7-1/4"	A		
MSDH-30		736-6-7/16"	A		
MSH-48 A&B		739'-1-11/16"	A		
MSH-62 A&B		735'-6"	A		
MSH-68 A&B		755'-8"	A		
	UNIT II				
AFSH-2	Main and Auxiliary	749'-4"	A		
AFSH-19	Steam	745'-7-1/4"	A		
AFSH-20		745'-7-1/4"	A		
AFSH-24		745'-6"	A		
AFSH-29 A&B		721'-1-9/16"	A		
AFSH-33		707'-5"	A		
AFSH-39		696'-6-1/4"	A		
AFSH-40		696'-6-1/4"	A		
AFSH-44		750'-7-1/2"	A		
AFSH-46		750'-7"	A		
MSDH-17		739'-0"	A		
MSDH-18		759'-0"	A		
MSDH-19		739'-0"	A		
MSDH-20		759'-0"	A		

SAFETY RELATED SNUBBERS

Snubber No	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT II				
MSH-23 A&B	Main and Auxiliary	739'-1-3/16"	A		
MSH-54 A&B	Steam	756'-0-1/16"	I		
MSH-81 A&B		735'-9"	A		
MSH-82 A&B		755'-8"	A		
MSH-83		761'-13/16""	I		
	UNIT I				
RHRRH-5	Safety Injection	723'-4-1/4"	I		
RHRRH-41		698'-11"	I		
RHRRH-58		670'-0"	A		
RHRRH-60		670'-0"	A		
RPCH-160		718'-1/2"	I		
RSIH-92		714'-11"	I		
RSIH-93		714'-11"	I		
RSIH-95		711'-2"	I		
RSIH-96		711'-2"	I		
RSIH-98		701'-2"	I		
RSIH-163		717'-9"	I		
RSIH-167		717'-9"	I		
RSIH-413 A&B		722'-8"	A		
RSIH-414		716'-10"	I		
RSIH-442		717'-9-1/2"	I		
RSIH-469		707'-6-1/2"	I		
RSIH-469		707'-6-1/2"	I		
RSIH-476		707'-1-3/4"	I		
SIRH-9		737'-0"	I		
SIRH-11		718'-6""	I		
SIRH-17		730'-0"	I		
SIRH-18		730'-0"	I		
SIRH-22		711'-4"	I		
SIRH-23 A&B		711'-4"	I		
SIRH-26		705'-0"	I		

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT II				
RHRH-13	Safety Injection	673'-9"	A		
RHRH-14		674'-0"	A		
RHRH-52		670'-6"	A		
RHRH-54		670'-6"	A		
RHRRH-19		700'-11"	I		
RHRRH-23		711'-2"	I		
RHRRH-28		707'-4"	I		
RSIH-265		699'-9"	I		
RSIH-268		713'-9-3/16"	I		
RSIH-343		719'-8-11/16"	I		
RSIH-349		703'-11"	I		
RSIH-350		703'-11"	I		
RSIH-353 A&B		701'-9"	I		
SIH-53		710'-3"	A		
SIRH-4A		711'-6-1/8"	I		
SIRH-4B		711'-3"	I		
SIRH-7		716'-3-1/16"	I		
SIRH-18		722'-6"	I		

SAFETY RELATED SHOCK SUPPRESSORS (SNUBBERS)

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT I				
RCRH-5 A&B	Reactor Coolant	732'-6"	I		
RCRH-12 A&B		720'-7"	I		
RCRH-26		762'-8"	I		
RCRH-27 A&B		761'-7"	I		
RCRH-34		764'-7"	I		
RCRH-45		765'-1"	I		
RERH-46		765'-1"	I		
RCRH-47		745'-10"	I		
RHRRH-15		705'-6"	I		
RHRRH-27		705'-6"	I		
RHRRH-29 A&B		705'-6"	I		
	UNIT II				
RCRH-5	Reactor Coolant	731'-6"	I		
RCRH-8		717'-6"	I		
RCRH-9		712'-0"	I		
RCRH-14		705'-9"	I		
RCRH-25		732'-2"	I		
RCRH-26		757'-7"	I		
RCRH-31		764'-1"	I		
RCRH-45		724'-6"	I		
RCRH-46		758'-3"	I		
RCRH-47		760'-3"	I		
RCRH-48		765'-1"	I		
RCRH-49		765'-1"	I		
RRCH-279 A&B		724'-9"	I		
RRCH-282		723'-2"	I		
RRCH-284 A&B		725'-8"	I		
RHRRH-2		699'-0"	I		
RHRRH-4		705'-11"	I		
RHRRH-9		705'-11"	I		
RHRRH-15		699'-0"	I		

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT I				
CWH-359	Cooling Water	705'-8"	A		
CWH-380		706'-11"	A		
CWH-385		709'-0"	A		
CWH-394		731'-0"	A		
CWH-395		746'-6"	A		
CWH-405		707'-10"	A		
CWH-429		722'-11"	A		
CWH-432		722'-11"	A		
CWH-433		735'-11"	A		
CWH-434		735'-11"	A		
CWH-436		737'-11"	A		
CWRH-80		730'-0"	I		
CWRH-81		729'-0"	I		
CWRH-82		730'-0"	I		
	UNIT II				
CWH-34	Cooling Water	709'-3"	A		
CWH-35		746'-8"	A		
CWH-39		710'-6"	A		
CWH-40		710'-6"	A		
CWH-44		730'-11"	A		
CWH-45		709'-0"	A		
CWH-49		723'-0"	A		
CWH-50		723'-10"	A		
CWH-52		736'-0"	A		
CWH-54		738'-0"	A		

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT I				
AFWH-72	Feedwater	752'-0"	I		
AFWH-82		728'-11"	A		
AFWH-84		728'-11"	A		
	UNIT II				
AFWH-72 A&B	Feedwater	706'-3/4"	A		
FWH-72 A&B		751'-0"	I		
	UNIT I				
*25.12620.003 3	Steam Generator	760'-9-1/2"	I	X	
*25.12620.003 - 4		760'-9-1/2"	I	X	
*25.12620.003 - 5		760'-9-1/2"	I	X	
*25.12620.003 - 6		760'-9-1/2"	I	X	
*25.12620.003 - 7		760'-9-1/2"	I	X	
*25.12620.003 - 8		760'-9-1/2"	I	X	
*25.12620-003 - 10		760'-9-1/2"	I	X	
*25.12620.003 - 15		760'-9-1/2"	I	X	
	UNIT II				
*25.12620.003 - 1		760'-9-1/2"	I	X	
*25.12620.003 - 2		760'-9-1/2"	I	X	
*25.12620.003 - 9		760'-9-1/2"	I	X	
*25.12620.003 - 11		760'-9-1/2"	I	X	
*25.12620.003 - 12		760'-9-1/2"	I	X	
*25.12620.003 - 13		760'-9-1/2"	I	X	
*25.12620.003 - 14		760'-9-1/2"	I	X	
*25.12620.003 - 16		760'-9-1/2"	I	X	
	UNIT I				
CVCH-182	Chemical & Vol Control	707'-6"	A		
RCRH-16 A&B		705'-2"	I		
RCRH-19		705'-2"	I		
RCRH-21		705'-7"	I		
RCRH-23 A&B		715'-11"	I		
RCVCH-907 A&B		717'-11"	I		
RCVCH-1293		712'-0"	I		
RPCH-22		703'-1"	I		
RPCH-23		703'-1"	I		
RPCH-121		707'-9"	I		
RPCH-139		704'-4"	I		
RPCH-140		707'-7"	I		
RPCH-146		714'-7"	I		
RPCH-147		714'-10"	I		
WDRH-24		707'-9"	I		

Notes

* Preservice documented testing of 900K Anker Holth snubbers has qualified their operability for all design conditions. Functional testing specified in 4.13 C is not required.

Unit 1 - Amendment No. ~~14~~, ~~49~~, 50
Unit 2 - Amendment No. ~~8~~, ~~43~~, 44

SAFETY RELATED SNUBBERS

Snubber No.	Location	Elevation	Accessible or Inaccessible (A or I)	Snubbers Especially Difficult to Remove	In High Radiation Areas During Shutdown
	UNIT II				
RCVCH-1396	Chemical & Vol	702'-10"	I		
RCVCH-1505	Control	708'-6"	I		
RCVCH-1513		710'-1"	I		
RCVCH-1524		719'-1"	I		
RCVCH-1574		721'-0"	I		
RCVCH-1668		705'-5"	I		
RCVCH-1373		722'-11"	I		
RCVCH-1389		706'-1"	I		
RRCH-253		704'-4"	I		
RRCH-255		704'-8"	I		
RRCH-261		707'-2"	I		
RRCH-288		707'-2"	I		
RRCH-291		704'-6"	I		
RRCH-292		704'-7"	I		
CVCH-166		708'-0"	A		
	UNIT I				
CCH-304	Comp Cooling	717'-7"	A		
CCH-373		712'-4"	A		
CCH-376 A&B		700'-5"	A		
CCH-377		703'-0"	A		
CCH-378		708'-4"	A		
CCH-380		670'-8"	A		
CCH-381 A&B		671'-4"	A		
CCH-397		699'-3"	A		
CCH-398 A&B		671'-4"	A		
	UNIT II				
CCH-161	Comp Cooling	717'-7"	A		
CCH-166		719'-11"	A		
CCH-167		720'-0"	A		
CCH-172		720'-0"	A		
CCH-173		708'-5"	A		
CCH-176		705'-3"	A		
CCH-179 A&B		671'-4"	A		
CCH-180		670'-8"	A		
CCH-181		708'-4"	A		
CCH-182		704'-2"	A		
CCH-185 A&B		671'-4"	A		
CCH-186		670'-10"	A		
	UNIT I				
RCSH-81	Containment Spray	76"'-9"	I		
RCSH-82		760'-8"	I		
RSCH-83 A&B		732'-1"	I		
	UNIT II				
CSH-75 A&B	Containment Spray	731'-10"	I		
CSH-76		752'-7"	I		
CSH-79		751'-9"	I		
CSH-82 A&B		731'-11"	I		
CSH-83		767'-2"	I		
CSH-84		767'-2"	I		
CSH-210		698'-0"	I		
CSH-215		698'-0"	A		
CSH-224		710'-6"	A		

4.13 SNUBBERS

Applicability

Applies to periodic testing and surveillance requirements of safety related hydraulic snubbers.

Objective

To verify the integrity and operability of hydraulic snubbers.

Specification

The following surveillance requirements apply to all snubbers listed in Table TS.3.12-1. These requirements augment the inspections required by Section XI of the ASME Code.

A. Visual Inspection of snubbers shall be conducted in accordance with the following schedule:

No. of Snubbers Found Inoperable per Inspection Period	Next Required Inspection Period
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%
3,4	124 days ± 25%
5,6,7	62 days ± 25%
8 or more	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

B. Visual inspections shall verify (1) that there are no visible indications of damage or impaired operability, (2) attachments to the supporting structure are secure, and (3) in those locations where snubber movement can be manually induced without disconnecting the snubber, that the snubber has freedom of movement and is not frozen up. Snubbers which appear inoperable as a result of visual inspection may be determined Operable for the purpose of establishing the next visual inspection interval by:

a. Clearly establishing the cause of the rejection for that particular snubber and for others that may be generically susceptible; and

b. Functionally testing the affected snubber in the as-found condition and finding it operable per Specification 4.13.D.

However, when the fluid port of a hydraulic snubber is found to be uncovered, the snubber shall be considered inoperable for purposes of establishing the next visual inspection interval. All hydraulic snubbers connected to an inoperable common hydraulic fluid reservoir shall be considered as inoperable snubbers.

C. Except as specified below, functional testing of snubbers listed in Table TS.3.12-1 shall be conducted at least once per 18 months during cold shutdown. Ten percent of the total of each type snubber shall be functionally tested either in place or in a bench test. For each snubber that does not meet the functional test acceptance criteria in Specification 4.13.D below, an additional ten percent of that type of snubber shall be functionally tested until no more failures are found or all snubbers of that type have been tested.

The representative sample selected for functional testing shall include the various configurations, operating environments, and the range of size and capacity of the snubbers. Twenty five percent of the sample shall include snubbers from the following three categories.

a. The first snubber away from a reactor vessel nozzle

b. Snubbers within five feet of heavy equipment (valve, pump, turbine, motor, etc)

c. Snubbers within ten feet of the discharge of a safety/relief valve

Snubbers identified in Table TS.3.12-1 as "High Radiation Area" or "Difficult to Remove" are exempt from functional testing provided a justifiable basis for exemption is presented for Commission review; snubber life testing is performed to qualify snubber operability for all design conditions; or snubbers of the same type, configuration, and similar service have been tested for a ten year period and no failures have occurred. In such exempt cases, a qualitative test report shall be on file to substitute for the required functional testing.

In addition to the regular sample and specified re-sampling, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber, if it is repaired and installed in another position, and the spare snubber shall be retested.

If any snubber selected for functional testing either fails to lockup or fails to move (i.e. frozen in place) the cause shall be evaluated and all snubbers subject to the same defect shall be functionally tested. This testing is in addition to the regular sample and specified re-samples.

D. Hydraulic snubber functional tests shall verify that:

a. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

b. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required to not displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

E. An engineering evaluation shall be performed for all components supported by inoperable snubbers. The purpose of this engineering evaluation shall be to determine if the components were adversely affected by the inoperable snubber(s) to ensure that the components remain capable of meeting the designed service.

F. The installation and maintenance records for each snubber listed in Table TS.3.12-1 shall be reviewed at least once every 18 months to verify that the indicated service life will not be exceeded prior to the next scheduled snubber service life review. If the indicated service life will be exceeded, the snubber service life shall be reevaluated or the snubber shall be replaced or reconditioned to extend its service life beyond the date of the next scheduled service life review. This reevaluation, replacement, or reconditioning shall be indicated in the records.

Basis

The visual inspection frequency is based upon maintaining a constant level of snubber protection to systems. Therefore, the required inspection interval varies inversely with the observed snubber failures and is determined by the number of inoperable snubbers found during an inspection. Inspections performed before that interval has elapsed may be used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) may not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

When the cause of the rejection of a snubber is clearly established and remedied for that snubber and for any other snubbers that may be generically susceptible, and verified by inservice functional testing, that snubber may be exempted from being counted as inoperable. Generically susceptible snubbers are those which are of a specific make or model and have the same design features directly related to rejection of the snubber by visual inspection, or are similarly located or exposed to the same environmental conditions such as temperature, radiation, and vibration.

When a snubber is found inoperable, an engineering evaluation is performed, in addition to the determination of the snubber mode of failure, in order to determine if any safety-related component or system has been adversly affected by the inoperability of the snubber. The engineering evaluation shall determine whether or not the snubber mode of failure has imparted a significant effect or degradation on the supported component or system.

To provide assurance of snubber functional reliability, a representative sample of 10% of the installed snubbers will be functionally tested during plant shutdowns at 18 month intervals. Observed failures of these sample snubbers shall require functional testing of additional units.

The service life of a snubber is evaluated via manufacturer input and information through consideration of the snubber service conditions and associated installation and maintenance records (newly installed snubber, seal replaced, spring replaced, in high radiation area, in high temperature area, etc. . .). The requirement to monitor the snubber service life is included to ensure that the snubbers periodically undergo a performance evaluation in view of their age and operating conditions. These records will provide statistical bases for future consideration of snubber service life.

All safety-related snubbers installed or planned for use at Prairie Island are hydraulic snubbers. No mechanical snubbers are used.

TABLE TS.4.1-1
(Page 1 of 5)

MINIMUM FREQUENCIES FOR CHECKS, CALIBRATIONS AND TEST OF INSTRUMENT CHANNELS

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
1.	Nuclear Power Range	S(1) M(4)	D(2) Q(4)	M(3) M(5) M(6) P(7)	R	1) Once/shift when in service 2) Heat balance 3) Signal to ΔT; bistable action (permissive, rod stop, trips), with the exception of the items covered in Remark #7. 4) Upper and lower chambers for axial off-set using in-core detectors 5) Simulated signal for testing positive and negative rate bistable action 6) Quadrant Power Tilt Monitor 7) P8 and P10 permissives and the 25% High Flux Low Setpoint Trip.
2.	Nuclear Intermediate Range	*S(1)	NA	T(2)	R	1) Once/shift when in service 2) Log Level; bistable action (permissive, rod stop, trips)
3.	Nuclear Source Range	*S(1)	NA	T(2)	R	1) Once/shift when in service 2) Bistable action (alarm, trips)
4.	Reactor Coolant Temperature	S(1,2)	R(1,2,3)	M(1) M(2) T(3)	R(1) R(2)	1) Overtemperature ΔT 2) Overpower ΔT 3) Control Rod Bank Insertion Limit Monitor
5.	Reactor Coolant Flow	S	R	M	NA	
6.	Pressurizer Water Level	S	R	M	NA	
7.	Pressurizer Pressure	S	R	M	NA	

TABLE TS.4.1-1
(Page 5 of 5)

	Channel Description	Check	Calibrate	Functional Test	Response Test	Remarks
35.	Event Monitoring Instrumentation	M	R	NA	NA	Includes all those in FSAR Table 7.7-2 and Table TS.3.15-1 not included elsewhere in this Table
36.	Steam Exclusion Actuation System	W	R	M	NA	See FSAR Appendix I, Section I.14.6
37.	Pressurizer PORV Control	NA	R	R	NA	Instrument Channels for PORV Control Including Overpressure Mitigation System
38.	Degraded Voltage 4KV Safeguard Busses	NA	R	M	NA	
39.	Loss of Voltage 4KV Safeguard Busses	NA	R	M	NA	

S – Each Shift
D – Daily
W – Weekly
M – Monthly
Q – Quarterly
R – Each refueling shutdown
P – Prior to each startup if not done previous week
T – Prior to each startup following shutdown in excess of 2 days if not done in the previous 30 days
NA – Not Applicable
* – See Specification 4.1.D

6. Plant radiation and contamination surveys.

7. Changes made to the plant as it is described in the Final Safety Analysis Report, reflected in updated, corrected and as-built drawings.

8. Cycling beyond normal limits for those components that have been designed to operate safely for a limited number of cycles beyond such limits.

9. Reactor coolant system in-service inspections.

10. Minutes of meetings of the Safety Audit Committee.

11. Records of Environmental Qualification which are covered under the provisions of paragraph 6.8.

12. Records of the service lives of all safety-related snubbers, including the date at which the service life commences and associated installation and maintenance records.

~~Order dated October 24, 1980~~

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 50 TO FACILITY OPERATING LICENSE NO. DPR-42

AND AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. 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 letter dated August 14, 1981, Northern States Power Company (NSP) requested to amend Appendix A to Facility Operating Licenses No. DPR-42 and DPR-60 for Prairie Island Nuclear Generating Plant, Units No. 1 and 2. The amendment would revise the snubber surveillance requirements in the Technical Specifications of Appendix A as requested by our letter dated November 20, 1980.

Discussion and Evaluation

As a result of discoveries of numerous inoperative snubbers during the period from 1973 to 1975 we requested that all licensees include a snubber surveillance program in the Technical Specifications for operating reactor plants. However, several deficiencies in the requirements were identified after the original surveillance programs were in place for several years. These deficiencies are:

1. Mechanical snubbers were not included in these requirements.

2. The rated capacity of snubbers was used as a limit to the inservice test requirement.

3. NRC approval was necessary for the acceptance of seal materials.

4. Inservice test requirements were not clearly defined.

5. In-place inservice testing was not permitted.

Since mechanical snubbers were not subject to any surveillance requirements, some licensees and permit holders believed that mechanical snubbers were preferred by NRC. Many plants used mechanical snubbers as original equipment and many others requested to replace their hydraulic snubbers with mechanical ones to simplify or avoid an inservice surveillance program. This is directly contradictory to NRC's intention, where for an unsurveyed mechanical snubber, the most likely failure is permanent lock-up. This failure mode can be harmful to the system during normal plant operations.

In the proposed amendment, the licensee has not included surveillance requirements for mechanical snubbers since these snubbers are not installed nor planned for use at the facility. On that basis we agree with the licensee that surveillance requirements for mechanical snubbers need not be included in the Technical Specifications of Appendix A.

During the period of 1973-1975, when the first hydraulic snubber surveillance requirements in the Technical Specifications were drafted, a compromise was made to limit the testing of snubbers to those with rated capacity of not more than 50,000 lbs. This is because of the available capacity of the test equipment and the requirement to test some parameters at the snubber rated load. Since then, greater equipment capacity and better understanding of parametric correlation have been developed. To maintain this arbitrary 50,000 lb. limit could mean an unnecessary compromise on plant safety.

The original hydraulic snubber problem started from leaking seals. Most seal materials of the 1973 vintage could not withstand the temperature and irradiation environments. Ethylene propylene was the first material that could offer a reasonable service life for those seals. In order to discourage the use of unproven material for those seals, the words "NRC approved material" were used in Technical Specifications. Staff members were asked to approve different seal materials on many occasions. Consequently, since the basis for the approval was not defined, the development of better seal materials by the industry was actually discouraged.

The not-well-defined acceptance criteria in the earlier version of the testing requirements resulted in non-uniform interpretations and implementation. Acceptance Criteria were set individually at widely different ranges. Since the rationale of adopting a specific acceptance criterion was not clear, I&E inspectors found it impossible to make any necessary corrections. In some cases, snubbers were tested without reference to acceptance criteria.

Testing of snubbers was usually accomplished by removing snubbers from their installed positions, mounting them on a testing rig, conducting the test, removing them from the rig, and reinstalling them to the working position. Many snubbers were damaged in the removal and reinstallation process. This defeated the purpose for conducting tests. Since methods and equipment have been developed to conduct in-place tests on snubbers, taking advantage of these developments results in minimizing the damage to snubbers caused by removal and reinstallation plus time and cost savings to the plant.

Considering this accumulated experience described above we concluded that the existing Technical Specification be revised to reflect this experience gained in the past several years. Therefore, the revised surveillance requirements for snubbers include the following:

1. If mechanical snubbers are used on safety related systems, then they must be included in the surveillance program.

2. No arbitrary snubber capacity is to be used as a limit to the inservice test requirements.

3. Seal material no longer requires NRC approval. A monitoring program shall be implemented to assure that snubbers are functioning within their service life.

4. Clearly defined inservice test requirements for snubbers shall be implemented.

5. In-place inservice testing shall be permitted.

By our letter dated November 20, 1980 the above revised requirements were transmitted to all licensees of operating plants, including NSP, except for those licensees whose plants are being reviewed under the Systematic Evaluation Program (SEP). That letter requested that the licensees propose a Technical Specification change to amend Appendix "A" of the license, incorporating these requirements. By letter dated August 14, 1981, NSP responded to our request for Prairie Island Nuclear Generating Plant Units Nos. 1 and 2. We have reviewed the licensee's submittal and find that the proposed amendment does include all of our requirements for the surveillance of safety-related snubbers.

On this basis we find that the proposed amendment which revises the surveillance for safety-related snubbers 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: October 28, 1981

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 No. 50 to Facility Operating License No. DPR-42, and Amendment No. 44 to Facility Operating License No. DPR-60 issued to Northern States Power Company (the licensee), which revised Technical Specifications for operation of Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2 (the facilities) located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revise the Appendix A Technical Specifications by clarifying snubber surveillance requirements and also correct some administrative errors.

The application for the 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.

7590-01

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 application for amendments dated August 14, 1981, (2) Amendment Nos. 50 and 44 to License Nos. DPR-42 and DPR-60, 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, 300 Nicollet Mall, Minneapolis, Minnesota 55401. 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 28th day of October, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

R. A. Clark

R. A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing