DOCKET FILE
50-282/306

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 18, 1997

Mr. Roger O. Anderson, Director
Nuclear Energy Engineering
Northern States Power Company
414 Nicollet Mall
Minneapolis, Minnesota 55401

SUBJECT: PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NOS. 1 AND 2 - ISSUANCE OF AMENDMENTS RE: INCORPORATION OF VOLTAGE-BASED STEAM GENERATOR TUBE REPAIR CRITERIA (TAC NOS. M98944 AND M98945)

Dear Mr. Anderson:

The Commission has issued the enclosed Amendment No.133to Facility Operating License No. DPR-42 and Amendment No125 to Facility Operating License No. DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TS) in response to your application dated May 15, 1997, as supplemented August 29, October 20, October 24, and October 28, 1997.

The amendments revise certain TS limitations on reactor coolant system leakage and steam generator tube surveillance, and implement voltage-based repair criteria per requirements of NRC Generic Letter 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking." In addition, the amendments correct a typographical error in TS Section 4.12.c. The amendments also add a license condition to Appendix B of the licenses. This license condition was proposed by Northern States Power Company (NSP) in its letter dated October 24, 1997.

During the staff's review of the amendments, errors in NSP's dose calculation made in support of the amendments were identified by both NSP and the NRC staff. These errors necessitated numerous teleconferences between NSP and NRC staff to clarify the application and resulted in additional supplements to the original application to correct the calculational errors.

DF01

A copy of our related Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY

Beth A. Wetzel, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-282 and 50-306

Enclosures: 1. Amendment No.133 to DPR-42
2. Amendment No.125 to DPR-60
3. Safety Evaluation

cc w/encl: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 133
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 May 15, 1997, as supplemented August 29, October 20, October 24, and October 28, 1997, 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 paragraphs 2.C.(2) and 2.C.(5) of Facility Operating License No. DPR-42 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 133 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 133 , are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance, with full implementation of the Technical Specifications within 30 days. License Condition 5 of Appendix B shall be implemented immediately upon issuance of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

Beth A. Wetzel, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment: 1. Changes to the Technical Specifications
2. Appendix B - Additional Conditions

Date of Issuance: November 18, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 133

FACILITY OPERATING LICENSE NO. DPR-42

DOCKET NO. 50-282

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS-vi	TS-vi
TS.3.1-9	TS.3.1-9
TS.4.12-3	TS.4-12-3
TS.4.12-4	TS.4.12-4
TS.4.12-5	TS.4.12-5
TS.4.12-6	TS.4.12-6
TS.4.12-7	TS.4.12-7
B.3.1-7	B.3.1-7
B.4.12-1	B.4.12-1
B.4.12-2	B.4.12-2
B.4.12-3	B.4.12-3
B.4.12-4	B.4.12-4

TABLE OF CONTENTS (Continued)

3.1.C.2 e. If the total reactor coolant system to secondary coolant system leakage through any one steam generator of a unit exceeds 150 gallons per day (GPD), within one hour initiate action to place the unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours and perform an inservice steam generator tube inspection in accordance with Technical Specification 4.12.

3. Pressure Isolation Valve Leakage

Leakage through the pressure isolation valves shall not exceed the maximum allowable leakage specified in Specification 4.3 when reactor coolant system average temperature exceeds 200°F. If the maximum allowable leakage is exceeded, within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

C. Inspection Frequencies-The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table TS.4.12-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.12.C.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table TS.4.12-1 during the shutdown subsequent to any of the following conditions.

   (a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.C.6.

   (b) A seismic occurrence greater than the Operating Basis Earthquake.

   (c) A loss-of-coolant accident requiring actuation of the engineered safeguards.

   (d) A main steam line or feedwater line break.

D. Acceptance Criteria

1. As used in this Specification:

(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

(c) Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is equal to 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criteria will be reduced to 40% wall penetration. This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* distance for F* tubes. The repair limit for the pressure boundary region of any sleeve is 31% of the nominal sleeve wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 4.12.D.4 for the repair limit applicable to these intersections.

(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

(h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.

(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty).

(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* tubes) required by Tables TS.4.12-1 and TS.4.12-2.

3. Tube repair, after October 1, 1997, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:

   CEN-629-P, Revision 2, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves";

   CEN-629-P, Addendum 1, Revision 1, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves"

4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:

   a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

   b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts, will be repaired or plugged, except as noted in Specification 4.12.D.4.c below.

   c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 4.12.D.4.a, b and c. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0 + NDE + Gr\left(\frac{CL - \Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL - \Delta t}{CL}\right)$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 4.12.D.4.a, b and c.

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

E. Reports

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube inservice inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube inservice inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection and (3) identification of tubes plugged or sleeved.

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 4.12.B for all tubes that have defects below the F* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

   e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

3.1 REACTOR COOLANT SYSTEM

Bases continued

C. Reactor Coolant System Leakage

Leakage from the reactor coolant system is collected in the containment or by other systems. These systems are the main steam system, condensate and feedwater system and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following (Reference 1):

1. An increased amount of makeup water required to maintain normal level in the pressurizer.
2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.
3. Containment sump water level indication.
4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the area radiation monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor (Reference 2).

The historical leak rate limit of 1 gpm corresponded to a through wall crack less than 0.6 inches long based on test data. Steam generator tubes having a 0.6-inch long through-wall crack have been shown to resist failure at pressures resulting from normal operation, LOCA, or steam line break accidents (Reference 3).

The leakage limits incorporated into Specification 3.1.C for implementation of the Steam Generator Voltage Based Alternate Repair Criteria are more restrictive than the standard operating leakage limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

Specification 3.1.C.3 specifies actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.1
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

At the request of the NRC, a requirement for in-service inspection of at least 20% of the total number of sleeves in service in both steam generators was added to TS 4.12.B. In addition, Table TS 4.12-2 was added to provide sample expansion requirements based on the results of the initial sample inspection similar to Table 4.12-1 This type of sample size and expansion requirement is consistent with the EPRI PWR Steam Generator Examination Guidelines. The sample selection is applied to each type of sleeve. Types of sleeves are categorized by such characteristics as the installation vendor, the sleeve material, the type of joint such as lower edge weld or lower hard roll joints, the sleeve location such as tube support plate or tubesheet and whether or not the welded joints have received post weld heat treatment.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameters found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameters, localized corrosion would most likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day from one steam generator). Historically, cracks not addressed by voltage-based alternate repair criteria and having a primary-to-secondary leakage less than 1.0 gpm (1440 gallons per day) during operation would have an adequate margin of safety against failure due to loads imposed by design basis accidents (Reference 1). Operational experience has demonstrated that primary-to-secondary leakage as low as 5 gallons per day will be detected by secondary system radiation monitors. To provide defense in depth for the voltage based repair criteria, leakage in excess of 150 gallons per day from one steam generator will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged or sleeved.

Wastage-type defects are unlikely with proper chemistry treatment of secondary coolant. However, even if this type of defect occurs it will be found during scheduled in-service steam generator tube inspections. Repair or plugging will be required of all tubes with imperfections that could develop defects having less than the minimum acceptable wall thickness prior to the next inservice inspection which, by the definition of Specification 4.12.D.1.(f), is 50% of the tube or sleeve nominal wall thickness. Wastage type defects having a wall thickness greater than

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

0.025 inches will have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 1). Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050-inch wall thickness (Reference 2).

Plugging or sleeving is not required for tubes meeting the F* criteria.

The F* distance will be controlled by a combination of eddy current inspection and/or process control. For a new additional roll expansion, the requirement will be at least 1.2 inches of new hard roll. This is controlled by the length of the rollers (1.25 inch effective length). The distance from the original roll transition zone is also controlled by the process in that the lower end of the new roll expansion is located one inch above the original roll expansion. In the case of the new roll, eddy current examination will confirm there are no indications in the new roll region and that there is a new roll region with well defined upper and lower expansion transitions.

When eddy current examination, alone, must determine the F* distance, such as in the existing hard roll region, or when multiple lengths of additional hard rolls have been added, the eddy current uncertainty is qualified by testing against known standards. That value is expected to be 0.18 inches. Therefore, the F* distance measured by eddy current (sum of 1.07 and 0.18) will be conservatively set at 1.3 inches.

When more than one Alternate Repair Criteria are used, the summation of leakage from all tubes left in service by all repair criteria must be less than the allowable leakage for the most limiting of those Alternate Repair Criteria.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

The following sleeve designs have been found acceptable by the NRC Staff:
a. Westinghouse Mechanical Sleeves (WCAP 10757)
b. Westinghouse Brazed Sleeves (WCAP-10820)
c. Combustion Engineering Leak Tight Sleeves (CEN-294-P, for sleeves installed prior to October 1, 1997)
d. Combustion Engineering Leak Tight Sleeves (CEN-629-P, for sleeves installed after October 1, 1997)

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve designs shall be made at least 90 days prior to use.

Tube Support Plate Repair Limit

The voltage-based repair limits of Specification 4.12.D.4 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of steam generator tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generator. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to Generic Letter 95-05 for additional description of the degradation morphology.

Implementation of Specification 4.12.D.4 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in Generic Letter 95-05.

The mid-cycle equation in Specification 4.12.D.4 should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

Specification 4.12.E.5 implements several reporting requirements recommended by Generic Letter 95-05 for situations which the NRC wants to be notified prior to returning the steam generators to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to Generic Letter 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the steam generators to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing Generic Letter 95-05 Section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per Generic Letter 95-05 Section 6.b (c) criteria.

References

1. Testimony of J Knight in the Prairie Island Public Hearing on 1/28/75

2. Testimony of L Frank in the Prairie Island Public Hearing on 1/28/75

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-42

Northern States Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
128	1. NSP will provide a licensed operator in the control room on an interim basis for the dedicated purpose of identifying an earthquake which results in a decreasing safeguards cooling water bay level. This operator will be in addition to the normal NSP administrative control room staffing requirements and will be provided until License Condition 2 is satisfied.	Prior to Unit 2 entering Mode 2.
128	2. NSP will submit dynamic finite element analyses of the intake canal banks by July 1, 1997 for NRC review. By December 31, 1998, NSP will complete, as required, additional analyses or physical modifications which provide the basis for extending the time for operator post-seismic cooling water load management and eliminating the dedicated operator specified in License Condition 1.	July 1, 1997, and December 31, 1998, as stated in Condition 2.
128	3. Based on the results of License Condition 2, NSP will revise the Updated Safety Analysis Report to incorporate the changes into the plant design bases. These changes will be included in the next scheduled revision of the Updated Safety Analysis Report following completion of License Condition 2 activities.	At the next USAR update following completion of Condition 2, but no later than June 1, 1999.
130	4. Prairie Island will assure that heavy loads do not present a potential for damaging irradiated fuel through use of 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.	This is effective immediately upon issuance of the amendment.
133	5. NSP will assure that during the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steamline break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).	This is effective immediately upon issuance of the amendment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 125
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 May 15, 1997, as supplemented August 29, October 20, October 24, and October 28, 1997, 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 paragraphs 2.C.(2) and 2.C.(5) of Facility Operating License No. DPR-60 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 125 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(5) Additional Conditions

The Additional Conditions contained in Appendix B, as revised through Amendment No. 125, are hereby incorporated into this license. The licensee shall operate the facility in accordance with the Additional Conditions.

3. This license amendment is effective as of the date of issuance, with full implementation of the Technical Specifications within 30 days. License Condition 5 of Appendix B shall be implemented immediately upon issuance of this amendment.

FOR THE NUCLEAR REGULATORY COMMISSION

Beth A. Wetzel

Beth A. Wetzel, Senior Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachment 1. Changes to the Technical Specifications
2. Appendix B - Additional Conditions

Date of Issuance: November 18, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 125

FACILITY OPERATING LICENSE NO. DPR-60

DOCKET NO. 50-306

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the area of change.

REMOVE	INSERT
TS-vi	TS-vi
TS.3.1-9	TS.3.1-9
TS.4.12-3	TS.4-12-3
TS.4.12-4	TS.4.12-4
TS.4.12-5	TS.4.12-5
TS.4.12-6	TS.4.12-6
TS.4.12-7	TS.4.12-7
B.3.1-7	B.3.1-7
B.4.12-1	B.4.12-1
B.4.12-2	B.4.12-2
B.4.12-3	B.4.12-3
B.4.12-4	B.4.12-4

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3.1.C.2 e. If the total reactor coolant system to secondary coolant system leakage through any one steam generator of a unit exceeds 150 gallons per day (GPD), within one hour initiate action to place the unit in HOT SHUTDOWN and be in at least HOT SHUTDOWN within the next 6 hours and be in COLD SHUTDOWN within the following 30 hours and perform an inservice steam generator tube inspection in accordance with Technical Specification 4.12.

3. Pressure Isolation Valve Leakage

Leakage through the pressure isolation valves shall not exceed the maximum allowable leakage specified in Specification 4.3 when reactor coolant system average temperature exceeds 200°F. If the maximum allowable leakage is exceeded, within one hour initiate the action necessary to place the unit in HOT SHUTDOWN, and be in at least HOT SHUTDOWN within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

5. Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages.

6. Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length.

C. Inspection Frequencies-The above required in-service inspections of steam generator tubes shall be performed at the following frequencies:

1. In-service inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the preservice inspection, result in all inspection results falling into the C-1 category or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

2. If the results of the inservice inspection of a steam generator conducted in accordance with Table TS.4.12-1 at 40 month intervals fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until the subsequent inspections satisfy the criteria of Specification 4.12.C.1; the interval may then be extended to a maximum of once per 40 months.

3. Additional, unscheduled inservice inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table TS.4.12-1 during the shutdown subsequent to any of the following conditions.

   (a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.1.C.6.

   (b) A seismic occurrence greater than the Operating Basis Earthquake.

   (c) A loss-of-coolant accident requiring actuation of the engineered safeguards.

   (d) A main steam line or feedwater line break.

D. Acceptance Criteria

1. As used in this Specification:

(a) Imperfection means an exception to the dimensions, finish or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections.

(b) Degradation means a service-induced cracking, wastage, wear or general corrosion occurring on either inside or outside of a tube.

(c) Degraded Tube means a tube containing imperfections ≥20% of the nominal wall thickness caused by degradation.

(d) % Degradation means the percentage of the tube wall thickness affected or removed by degradation.

(e) Defect means an imperfection of such severity that it exceeds the repair limit. A tube containing a defect is defective.

(f) Repair Limit means the imperfection depth at or beyond which the tube shall be removed from service by plugging or repaired by sleeving because it may become unserviceable prior to the next inspection and is equal to 50% of the nominal tube wall thickness. If significant general tube thinning occurs, this criteria will be reduced to 40% wall penetration. This definition does not apply to the portion of the tube in the tubesheet below the F* distance provided the tube is not degraded (i.e., no indications of cracks) within the F* distance for F* tubes. The repair limit for the pressure boundary region of any sleeve is 31% of the nominal sleeve wall thickness. This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 4.12.D.4 for the repair limit applicable to these intersections.

(g) Unserviceable describes the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break.

(h) Tube Inspection means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

(i) Sleeving is the repair of degraded tube regions using a new Alloy 690 tubing sleeve inserted inside the parent tube and sealed at each end by welding or by replacing the lower weld in a full depth tubesheet sleeve with a hard rolled joint. The new sleeve becomes the pressure boundary spanning the original degraded tube region.

(j) F* Distance is the distance from the bottom of the hardroll transition toward the bottom of the tubesheet that has been conservatively determined to be 1.07 inches (not including eddy current uncertainty).

(k) F* Tube is a tube with degradation, below the F* distance, equal to or greater than 40%, and not degraded (i.e., no indications of cracking) within the F* distance.

2. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug or repair by sleeving all tubes exceeding the repair limit and all tubes containing through-wall cracks or classify as F* tubes) required by Tables TS.4.12-1 and TS.4.12-2.

3. Tube repair, after October 1, 1997, using Combustion Engineering welded sleeves shall be in accordance with the methods described in the following:

CEN-629-P, Revision 2, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves";

CEN-629-P, Addendum 1, Revision 1, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves"

4. Tube Support Plate Repair Limit is used for the disposition of a steam generator tube for continued service that is experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates. At tube support plate intersections, the repair limit is based on maintaining steam generator serviceability as described below:

a. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to 2.0 volts will be allowed to remain in service.

b. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts, will be repaired or plugged, except as noted in Specification 4.12.D.4.c below.

c. Steam generator tubes, with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than 2.0 volts but less than or equal to the upper voltage repair limit, may remain in service if a rotating pancake coil (or comparable examination technique) inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits in Specifications 4.12.D.4.a, b and c. The mid-cycle repair limits are determined from the following equations:

$$V_{MURL} = \frac{V_{SL}}{1.0+NDE+Gr\left(\frac{CL-\Delta t}{CL}\right)}$$

$$V_{MLRL} = V_{MURL} - (V_{URL} - 2.0)\left(\frac{CL-\Delta t}{CL}\right)$$

where:

V_{URL} = upper voltage repair limit

V_{LRL} = lower voltage repair limit

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle

V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle

Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented

CL = cycle length (time between two scheduled steam generator inspections)

V_{SL} = structural limit voltage

Gr = average growth rate per cycle length

NDE = 95 percent cumulative probability allowance for nondestructive examination uncertainty (i.e., a value of 20 percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach as described in Specifications 4.12.D.4.a, b and c.

Note: The upper voltage repair limit is calculated according to the methodology in Generic Letter 95-05 as supplemented.

E. Reports

1. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or sleeving, the number of tubes plugged or sleeved in each steam generator shall be reported to the Commission within 15 days.

2. The results of steam generator tube inservice inspections shall be included with the summary reports of ASME Code Section XI inspections submitted within 90 days of the end of each refueling outage. Results of steam generator tube inservice inspections not associated with a refueling outage shall be submitted within 90 days of the completion of the inspection. These reports shall include: (1) number and extent of tubes inspected, (2) location and percent of wall-thickness penetration for each indication of an imperfection and (3) identification of tubes plugged or sleeved.

3. Results of steam generator tube inspections which fall into Category C-3 require notification to the Commission prior to resumption of plant operation, and reporting as a special report to the Commission within 30 days. This special report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

4. The results of inspections performed under Specification 4.12.B for all tubes that have defects below the F* distance, and were not plugged, shall be reported to the Commission within 15 days following the inspection. The report shall include:

   a. Identification of F* tubes, and

   b. Location and extent of degradation.

5. For implementation of the voltage-based repair criteria to tube support plate intersections, notify the NRC staff prior to returning the steam generators to service should any of the following conditions arise:

   a. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leak limit (determined from the licensing basis dose calculation for the postulated main steamline break) for the next operating cycle.

   b. If circumferential crack-like indications are detected at the tube support plate intersections.

   c. If indications are identified that extend beyond the confines of the tube support plate.

   d. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.

   e. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

3.1 REACTOR COOLANT SYSTEM

Bases continued

C. Reactor Coolant System Leakage

Leakage from the reactor coolant system is collected in the containment or by other systems. These systems are the main steam system, condensate and feedwater system and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following (Reference 1):

1. An increased amount of makeup water required to maintain normal level in the pressurizer.

2. A high temperature alarm in the leakoff piping provided to collect reactor head flange leakage.

3. Containment sump water level indication.

4. Containment pressure, temperature, and humidity indication.

If there is significant radioactive contamination of the reactor coolant, the radiation monitoring system provides a sensitive indication of primary system leakage. Radiation monitors which indicate primary system leakage include the containment air particulate and gas monitors, the area radiation monitors, the condenser air ejector monitor, the component cooling water monitor, and the steam generator blowdown monitor (Reference 2).

The historical leak rate limit of 1 gpm corresponded to a through wall crack less than 0.6 inches long based on test data. Steam generator tubes having a 0.6-inch long through-wall crack have been shown to resist failure at pressures resulting from normal operation, LOCA, or steam line break accidents (Reference 3).

The leakage limits incorporated into Specification 3.1.C for implementation of the Steam Generator Voltage Based Alternate Repair Criteria are more restrictive than the standard operating leakage limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

Specification 3.1.C.3 specifies actions to be taken in the event of failure or excessive leakage of a check valve which isolates the high pressure reactor coolant system from the low pressure RHR system piping.

References

1. USAR, Section 6.5
2. USAR, Section 7.5.1
3. Testimony by J Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases

The Surveillance Requirements for inspection of the steam generator tubes ensure that the structural integrity of this portion of the RCS will be maintained. The program for inservice inspection of steam generator tubes is based on a modification of Regulatory Guide 1.83, Revision 1. In-service inspection of steam generator tubing is essential in order to maintain surveillance of the conditions of the tubes in the event that there is evidence of mechanical damage or progressive degradation due to design, manufacturing errors, or in-service conditions that lead to corrosion. In-service inspection of steam generator tubing also provides a means of characterizing the nature and cause of any tube degradation so that corrective measures can be taken.

At the request of the NRC, a requirement for in-service inspection of at least 20% of the total number of sleeves in service in both steam generators was added to TS 4.12.B. In addition, Table TS 4.12-2 was added to provide sample expansion requirements based on the results of the initial sample inspection similar to Table 4.12-1 This type of sample size and expansion requirement is consistent with the EPRI PWR Steam Generator Examination Guidelines. The sample selection is applied to each type of sleeve. Types of sleeves are categorized by such characteristics as the installation vendor, the sleeve material, the type of joint such as lower edge weld or lower hard roll joints, the sleeve location such as tube support plate or tubesheet and whether or not the welded joints have received post weld heat treatment.

The plant is expected to be operated in a manner such that the secondary coolant will be maintained within those parameters found to result in negligible corrosion of the steam generator tubes. If the secondary coolant chemistry is not maintained within these parameters, localized corrosion would most likely result in stress corrosion cracking. The extent of cracking during plant operation would be limited by the limitation of steam generator leakage between the primary coolant system and the secondary coolant system (primary-to-secondary leakage = 150 gallons per day from one steam generator). Historically, cracks not addressed by voltage-based alternate repair criteria and having a primary-to-secondary leakage less than 1.0 gpm (1440 gallons per day) during operation would have an adequate margin of safety against failure due to loads imposed by design basis accidents (Reference 1). Operational experience has demonstrated that primary-to-secondary leakage as low as 5 gallons per day will be detected by secondary system radiation monitors. To provide defense in depth for the voltage based repair criteria, leakage in excess of 150 gallons per day from one steam generator will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged or sleeved.

Wastage-type defects are unlikely with proper chemistry treatment of secondary coolant. However, even if this type of defect occurs it will be found during scheduled in-service steam generator tube inspections. Repair or plugging will be required of all tubes with imperfections that could develop defects having less than the minimum acceptable wall thickness prior to the next inservice inspection which, by the definition of Specification 4.12.D.1.(f), is 50% of the tube or sleeve nominal wall thickness. Wastage type defects having a wall thickness greater than

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

0.025 inches will have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents (Reference 1). Steam generator tube inspections of operating plants have demonstrated the capability to reliably detect wastage type defects that have penetrated 20% of the original 0.050-inch wall thickness (Reference 2).

Plugging or sleeving is not required for tubes meeting the F* criteria.

The F* distance will be controlled by a combination of eddy current inspection and/or process control. For a new additional roll expansion, the requirement will be at least 1.2 inches of new hard roll. This is controlled by the length of the rollers (1.25 inch effective length). The distance from the original roll transition zone is also controlled by the process in that the lower end of the new roll expansion is located one inch above the original roll expansion. In the case of the new roll, eddy current examination will confirm there are no indications in the new roll region and that there is a new roll region with well defined upper and lower expansion transitions.

When eddy current examination, alone, must determine the F* distance, such as in the existing hard roll region, or when multiple lengths of additional hard rolls have been added, the eddy current uncertainty is qualified by testing against known standards. That value is expected to be 0.18 inches. Therefore, the F* distance measured by eddy current (sum of 1.07 and 0.18) will be conservatively set at 1.3 inches.

When more than one Alternate Repair Criteria are used, the summation of leakage from all tubes left in service by all repair criteria must be less than the allowable leakage for the most limiting of those Alternate Repair Criteria.

Whenever the results of any steam generator tubing in-service inspection fall into Category C-3, these results will be promptly reported to the Commission prior to resumption of plant operation. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for analysis, laboratory examinations, tests, additional eddy-current inspection, and revision of the Technical Specifications, if necessary.

Degraded steam generator tubes may be repaired by the installation of sleeves which span the section of degraded steam generator tubing. A steam generator tube with a sleeve installed meets the structural requirements of tubes which are not degraded.

The following sleeve designs have been found acceptable by the NRC Staff:

a. Westinghouse Mechanical Sleeves (WCAP 10757)
b. Westinghouse Brazed Sleeves (WCAP-10820)
c. Combustion Engineering Leak Tight Sleeves (CEN-294-P, for sleeves installed prior to October 1, 1997)
d. Combustion Engineering Leak Tight Sleeves (CEN-629-P, for sleeves installed after October 1, 1997)

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

Descriptions of other future sleeve designs shall be submitted to the NRC for review and approval prior to their use in the repair of degraded steam generator tubes. The submittals related to other sleeve designs shall be made at least 90 days prior to use.

Tube Support Plate Repair Limit

The voltage-based repair limits of Specification 4.12.D.4 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of steam generator tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generator. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to Generic Letter 95-05 for additional description of the degradation morphology.

Implementation of Specification 4.12.D.4 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit; V_{URL}, is determined from the structural voltage limit by applying the following equation:

$$V_{URL} = V_{SL} - V_{Gr} - V_{NDE}$$

where V_{Gr} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in Generic Letter 95-05.

The mid-cycle equation in Specification 4.12.D.4 should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Bases continued

Specification 4.12.E.5 implements several reporting requirements recommended by Generic Letter 95-05 for situations which the NRC wants to be notified prior to returning the steam generators to service. For the purposes of this reporting requirement, leakage and conditional burst probability can be calculated based on the as-found voltage distribution rather than the projected end-of-cycle voltage distribution (refer to Generic Letter 95-05 for more information) when it is not practical to complete these calculations using the projected EOC voltage distributions prior to returning the steam generators to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing Generic Letter 95-05 Section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per Generic Letter 95-05 Section 6.b (c) criteria.

References

1. Testimony of J Knight in the Prairie Island Public Hearing on 1/28/75

2. Testimony of L Frank in the Prairie Island Public Hearing on 1/28/75

APPENDIX B

ADDITIONAL CONDITIONS

FACILITY OPERATING LICENSE NO. DPR-60

Northern States Power Company shall comply with the following conditions on the schedules noted below:

Amendment Number	Additional Condition	Implementation Date
120	1. NSP will provide a licensed operator in the control room on an interim basis for the dedicated purpose of identifying an earthquake which results in a decreasing safeguards cooling water bay level. This operator will be in addition to the normal NSP administrative control room staffing requirements and will be provided until License Condition 2 is satisfied.	Prior to Unit 2 entering Mode 2.
120	2. NSP will submit dynamic finite element analyses of the intake canal banks by July 1, 1997 for NRC review. By December 31, 1998, NSP will complete, as required, additional analyses or physical modifications which provide the basis for extending the time for operator post-seismic cooling water load management and eliminating the dedicated operator specified in License Condition 1.	July 1, 1997, and December 31, 1998, as stated in Condition 2.
120	3. Based on the results of License Condition 2, NSP will revise the Updated Safety Analysis Report to incorporate the changes into the plant design bases. These changes will be included in the next scheduled revision of the Updated Safety Analysis Report following completion of License Condition 2 activities.	At the next USAR update following completion of Condition 2, but no later than June 1, 1999.
122	4. Prairie Island will assure that heavy loads do not present a potential for damaging irradiated fuel through use of 1) a single-failure-proof crane with rigging and procedures which implement Prairie Island commitments to NUREG-0612; or 2) spent fuel pool covers with their implementing plant procedures for installation and use.	This is effective immediately upon issuance of the amendment.
125	5. NSP will assure that during the implementation of steam generator repairs utilizing the voltage-based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steamline break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578 °F).	This is effective immediately upon issuance of the amendment.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 133 AND 125 TO

FACILITY OPERATING 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

1.0 INTRODUCTION

By letter dated May 15, 1997, as supplemented August 29, October 20, October 24, and October 28, 1997, the Northern States Power Company (NSP or the licensee) requested amendments to the Technical Specifications (TS) appended to Facility Operating License Nos. DPR-42 and DPR-60 for the Prairie Island Nuclear Generating Plant, Unit Nos. 1 and 2. The proposed amendments would implement voltage-based alternate repair criteria for steam generator tubes in the TS. The proposed alternate repair criteria would allow steam generator tubes having outside diameter stress-corrosion cracking (ODSCC) that is predominately axially oriented and confined within the tube support plates to remain in service on the basis of bobbin coil voltage response. The NRC guidance on the alternate repair criteria is specified in Generic Letter (GL) 95-05, "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking."

The August 29, October 20, October 24, and October 28, 1997, supplements provided clarifying information, proposed license conditions, and updated TS pages. This information was within the scope of the original application and did not change the staff's initial proposed no significant hazards consideration determination (62 FR 43371).

2.0 BACKGROUND

The acceptance criteria (i.e., plugging limits) for degraded steam generator tubes are specified in the plant TS. The traditional strategy for achieving adequate structural and leakage integrity of the degraded tubes has been to establish a minimum wall thickness requirement in accordance with NRC Regulatory Guide (RG) 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes." The minimum wall thickness requirement was developed with the assumption of a uniform thinning of the tube wall. This assumed degradation mechanism is inherently conservative for certain forms of tube degradation. Conservative repair limits may lead to removing degraded tubes from service that may otherwise have adequate structural and leakage integrity for further service.

To reduce unnecessary conservatism in the minimum wall thickness requirement for certain degradation, the industry proposed voltage-based repair criteria for ODSCC confined within the thickness of the tube support plates. The staff published several conclusions regarding voltage-based repair criteria in draft NUREG-1477, "Voltage-Based Interim Plugging Criteria for Steam Generator Tubes," and in a draft GL titled "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes." The latter document was published for public comment in the *Federal Register* on August 12, 1994 (59 FR 41520). On August 3, 1995, the staff issued GL 95-05 that took into consideration public comments on the draft GL cited above, domestic operating experience under the voltage-based repair criteria, and additional data made available from European nuclear power plants.

The guidance of GL 95-05 does not set depth-based limits on predominantly axially oriented ODSCC at tube support plate locations; rather it relies on empirically derived correlations between a nondestructive inspection parameter, the bobbin coil voltage, and tube burst pressure and leak rate. The staff recognizes that although the total tube integrity margins may be reduced following application of a voltage-based repair criteria, the guidance in GL 95-05 ensures structural and leakage integrity continue to be maintained at acceptable levels consistent with the requirements of 10 CFR Part 50 and 10 CFR Part 100. Since the voltage-based repair criteria do not require minimum tube wall thickness, there is the possibility for tubes with through-wall cracks to remain in service. Because of the increased likelihood of such flaws, the staff included provisions for augmented steam generator tube inspections and restrictive operational leakage limits.

GL 95-05 specifies, in part, that (1) the repair criteria are only applicable to predominantly axially oriented ODSCC located within the bounds of the tube support plates, (2) licensees perform an evaluation to confirm that the degraded steam generator tubes will retain adequate structural and leakage integrity from cycle to cycle, (3) licensees adhere to specific inspection criteria to ensure consistency in methods between inspections, (4) tubes must be periodically removed from the steam generators, examined, and destructively tested to verify the morphology of the degradation and provide burst and leakage data for structural and leakage integrity evaluations, (5) the operational leakage limit in the plant TS be reduced, (6) licensees implement an operational leakage monitoring program, and (7) specific reporting requirements be incorporated into the plant TS.

Each Prairie Island unit has two Westinghouse Model 51 steam generators, which use mill-annealed alloy 600 tubing. These steam generators use carbon steel drilled-hole tube support plates and do not have flow distribution baffle plates. The outside diameter and nominal wall thickness of each tube are 7/8 inch and 0.050 inch, respectively.

3.0 EVALUATION

The licensee stated that it will comply with the guidance in GL 95-05 when implementing its voltage-based alternate repair criteria. In addition, the licensee proposed to incorporate verbatim the model TS in GL 95-05 into the TS for Prairie Island Units 1 and 2. The major issues related to the licensee's implementation of the alternate repair criteria are discussed below.

3.1 Tube Repair Limits

The proposed repair criteria will (1) permit degraded tubes having indications confined to within the thickness of the tube support plates with bobbin voltages less than or equal to 2.0 volts to remain in service, (2) permit degraded tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than 2.0 volts but less than or equal to the upper voltage limit to remain in service if a motorized rotating pancake coil probe or acceptable alternative inspection does not detect degradation, and (3) require degraded tubes having indications confined to within the thickness of the tube support plates with bobbin voltages greater than the upper voltage limit be plugged or repaired.

The proposed lower voltage limit of 2.0 volts is derived based on the use of a correlation between the burst pressure and the bobbin coil voltage of pulled tube and model boiler data and is consistent with the recommended value specified in GL 95-05 for 7/8-inch steam generator tubing. The upper voltage limit is derived based on the lower 95-percent prediction interval of the burst pressure versus bobbin voltage correlation, adjusted for lower bound material properties evaluated at the 95-percent confidence level. The upper voltage limit is further reduced to account for uncertainty in the nondestructive examination technique and flaw growth over the next operating cycle. The industry periodically updates the database for burst pressure and bobbin voltage when the destructive test data from pulled tubes are available; therefore, the upper voltage limit may vary as additional data are incorporated into the database.

3.2 Inspection Issues

Section 3.c.3 of Attachment 1 to GL 95-05 specifies guidance for probe wear. The licensee proposed to use an alternative to section 3.c.3. The alternative approach, developed through the Nuclear Energy Institute, specifies that if the probe does not satisfy the voltage variability criterion for wear of ± 15-percent limit before its replacement, all tubes that exhibited flaw signals with voltage responses measured at 75 percent or greater of the lower repair limit must be reinspected with a bobbin probe satisfying the ± 15-percent wear standard criterion. The voltages from the reinspection should be used as the basis for tube repair. The staff completed a review of the Nuclear Energy Institute proposed alternative method and concluded that the approach is acceptable as discussed in a letter from Brian Sheron of the NRC to Alex Marion of the Nuclear Energy Institute dated March 18, 1996. The licensee's proposal to follow the industry approach to address probe wear is acceptable.

In the laboratory and field studies supporting the alternative probe wear criteria, the correlation of voltages measured by worn probes and new probes shows that for all significant voltage levels, the worn probe voltages are never less than 75 percent of the new probe voltage as discussed in the letter from Alex Marion of the Nuclear Energy Institute to Brian Sheron of the NRC dated January 23, 1996. However, in the 90-day inspection report for Byron Unit 1 dated September 9, 1996, Commonwealth Edison, the licensee for Byron, compared the worn probe voltage to the new probe voltage and found that the worn probe voltage was substantially less than 75 percent of the new probe voltage for a few indications. Commonwealth Edison evaluated these indications and concluded that the criteria to retest tubes with worn probe voltages above 75 percent of the repair limit is adequate and generally conservative due to the trend for worn probe voltages to exceed new probe voltages. Comparison of the actual and projected end-of-cycle voltages did not show anything unusual attributable to the alternate probe wear criteria. The staff concludes that the aforementioned probe wear results do not indicate an immediate need to modify the probe wear criteria developed by the industry. However, the staff will continue to monitor probe wear in the licensees' 90-day inspection reports.

With respect to probe variability, the licensee proposed to follow an alternative approach developed through the Nuclear Energy Institute. The proposed procedures and methodology are described in an October 15, 1996, letter from A. Marion, Nuclear Energy Institute, to B. Sheron, NRC. The approach specifies that the voltage responses from the primary frequency and mix frequency channels of new probes be within ±10 percent of the nominal voltage responses when voltages are normalized to the 20-percent flaw values. The nominal voltage responses were established as the average voltages obtained from the American Society of Mechanical Engineers (ASME) standard drilled-hole flaws for at least 10 production probes. The licensee's proposal to follow the industry approach to address probe variability is acceptable.

3.3 Structural and Leakage Integrity Assessments

GL 95-05 guidance for the voltage-based repair criteria focuses on maintaining tube structural integrity during the full range of normal, transient, and postulated accident conditions with adequate allowance for eddy-current test uncertainty and flaw growth projected to occur during the next operating cycle. RG 1.121 recommends that a margin of safety of 1.43 against tube failure under postulated accident conditions and a margin of safety of 3 against burst during normal operation be maintained for steam generator tubes. Because GL 95-05 addresses tubes affected with ODSCC confined to within the thickness of the tube support plate during normal operation, the staff concluded that the structural constraint provided by the tube support plate ensures all tubes to which the voltage-based criteria apply will retain a margin of 3 with respect to burst under normal operating conditions. For a postulated main steamline break accident, however, the tube support plate may displace axially during steam generator blowdown such that the ODSCC-affected portion of the tubing may no longer be fully constrained by the tube support plate. Accordingly, it is appropriate to consider the ODSCC affected regions of the tubes as free standing tubes for the purpose of assessing burst integrity under postulated main steamline break conditions.

In order to confirm the structural and leakage integrity of the tube until the next scheduled inspection, GL 95-05 specifies a methodology to determine the conditional burst probability and the total primary-to-secondary leak rate from an affected steam generator during a postulated main steamline break event. To complete GL 95-05 prescribed assessments, the licensee proposes to follow the methodology described in WCAP-14277, Revision 1, "SLB Leak Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," dated December 1996 (See E. Fitzpatrick, Indiana Michigan Power Co., letter to NRC dated December 20, 1996). The staff finds the methodology in WCAP-14277, Revision 1, acceptable for Prairie Island.

GL 95-05 specifies that the structural and leakage integrity assessments should use the latest available database from destructive examinations of tubes removed from Westinghouse-designed steam generators. A protocol is being established between the industry and the NRC to formalize the requirements for updating the industry database. The licensee indicated that it will follow the protocol as documented in a letter to G. Lainas, NRC, from D. Modeen of the Nuclear Energy Institute, dated January 15, 1997, until the final version of the protocol is completed. In addition, the licensee will describe, in the GL 95-05 90-day reports, the database that was used for GL 95-05 specified calculations. The staff finds that the licensee's commitment to follow the protocol and to use NRC approved database to perform structural and leakage assessments are acceptable.

3.3.1 Conditional Probability of Burst

The licensee will use the methodology described in Revision 1 of WCAP-14277 for performing a probabilistic analysis to quantify the potential for steam generator tube ruptures given a main steamline break event. The results of the probabilistic analysis will be compared to a threshold value of $1x10^{-2}$ per cycle in accordance with GL 95-05. This threshold value provides assurance that the probability of burst is acceptable considering the assumptions of the calculation and the results of the staff's generic risk assessment for steam generators contained in NUREG-0844, "NRC Integrated Program for the Resolution of Unresolved Safety Issues A-3, A-4, and A-5 Regarding Steam Generator Tube Integrity." Failure to meet the threshold value indicates ODSCC confined to within the thickness of the tube support plate could contribute a significant fraction to the overall conditional probability of tube rupture from all forms of degradation assumed and evaluated as acceptable in NUREG-0844. The NRC staff concludes the licensee's proposed methodology for calculating the conditional burst probability is consistent with the guidance in GL 95-05 and is acceptable.

3.3.2 Accident Leakage

The licensee will use the methodology described in Revision 1 of WCAP-14277 for calculating the steam generator tube leakage from the faulted steam generator during a postulated main steamline break event. The model consists of two major components: (1) a model predicting the probability that a given indication will leak as a function of voltage (i.e., the probability of leakage model); and (2) a model predicting leak rate as a function of voltage, given that leakage occurs (i.e., the conditional leak rate model). The staff concludes that the licensee's proposed methodology for calculating the tube leakage is consistent with the guidance in GL 95-05 and is acceptable.

3.3.3 Primary-to-Secondary Leakage

Because the voltage-based repair criteria would allow degraded tubes to remain in service, the degraded tubes may develop through-wall cracks which may leak during normal operation, transients, or postulated accidents. Therefore, as a defense-in-depth measure, GL 95-05 specifies that the operational leakage limits of the plant TS be limited to 150 gallons per day (gpd) from any one steam generator. The licensee proposed to change the leakage limits in the plant TS to 150 gpd through any one steam generator. In addition, the licensee has incorporated the guidelines in Electric Power Research Institute (EPRI) Report TR-104788, "PWR Primary-to-Secondary Leak Guidelines," into the Prairie Island plant operating procedures. The staff concludes that the proposed operational leakage limit of 150 gpd for Prairie Island TS is consistent with GL 95-05 and, therefore, is acceptable.

3.3.4. Potential for Tube Collapse

There is a potential for tube collapse in the steam generator at some plants during a loss-of-coolant accident (LOCA) in combination with a safe shutdown earthquake (SSE). This is the case as the tube support plates may become deformed as a result of lateral loads at the wedge supports at the periphery of the plate due to the combined effects of the LOCA rarefaction wave and SSE loadings. Then, the resulting pressure differential on the deformed tubes may cause some of the tubes to collapse. There are two issues associated with the steam generator tube collapse. First, the collapse of steam generator tubing reduces the RCS [reactor coolant system] flow area through the tubes. The reduction in flow area increases the resistance to flow of steam from the core during a LOCA which, in turn, may potentially increase the peak clad temperature. Second, there is a potential that partial through-wall cracks in tubes could progress to complete through-wall cracks during tube deformation or collapse.

Tubes that are susceptible to collapse during accident conditions will be excluded from application of the voltage-based repair criteria. Since the leak-before-break methodology is applicable to the reactor coolant loop piping at Prairie Island, the probability of breaks in the primary loop piping is sufficiently low that they need not be considered in the structural design of the plant. However, review of Westinghouse seismic umbrella spectra for Model 51 steam generators shows that Prairie Island is bounded by these spectra, such that no tubes will undergo deformation due to the combined effects of LOCA plus SSE, and, therefore, no tubes will be excluded from application of the criteria due to loading from LOCA plus SSE.

Based on the staff's review, as discussed above, the staff concurs with the licensee's assessment that no tubes need be excluded from application of the voltage-based repair criteria due to combined LOCA plus SSE loads.

3.4 Degradation Monitoring

To confirm the nature of the degradation at the tube support plate elevations, tubes are periodically removed from the steam generators for destructive tests. The test data from removed tubes can confirm that the nature of the degradation observed at these locations is predominantly axially oriented ODSCC, provide data for assessing the reliability of the inspection methods, and supplement the existing databases (e.g., burst pressure, probability of leakage, and leak rate). GL 95-05 specifies that at least two tubes be removed from steam generators with the objective of retrieving as many intersections as practical (minimum of four intersections) during the plant steam generator inspection outage preceding initial application of the voltage-based repair criteria. On an ongoing basis, additional tube specimens (minimum of two intersections) should be removed at the first refueling outage following 34 effective full power months of operation or at the maximum interval of three refueling outages after the previous tube pull. Alternatively, the licensee may participate in an industry-sponsored tube pull program endorsed by the staff as described in GL 95-05. The licensee pulled two tubes during the current Unit 1 outage and confirmed that tube degradation was consistent with that attributable to ODSCC. The licensee plans to pull future tubes consistent with the tube removal guidelines in GL 95-05 until an NRC-endorsed industry program is available as described in Section 4.a of GL 95-05.

3.5 Assessment of Radiological Consequences

In its license amendment submittal dated May 15, 1997, the licensee requested that the specific activity limits of dose-equivalent iodine-131 (^{131}I) in the primary coolant be established at 1.0 μCi/g for the 48-hour limit and at 60 μCi/g for the maximum instantaneous limit (in accordance with GL 95-05). The allowable activity level of dose-equivalent ^{131}I in the secondary coolant was assumed to be equivalent to the TS limit of 0.1 μCi/g. This license submittal also requested that Prairie Island be approved to operate based upon a 6.4 gallons per minute (gpm) primary-to-secondary leak initiated by an accident in the faulted steam generator and the TS allowable value for primary-to-secondary leakage from the intact steam generator of 150 gpd. As part of the request for license amendment, the licensee performed an assessment of the radiological dose consequences of a main steamline break accident. The licensee's calculations assumed that the duration of the accident initiated iodine spike is 1.6 hours.

In a telephone conversation with the licensee on October 1, 1997, the licensee stated that it had found an error in its calculations and based on the revised calculations, NSP would like to revise the requested allowable leak rate for the faulted steam generator by reducing the leakage from 6.4 gpm to 4.64 gpm. Using this revised leakage figure, the licensee stated that it had calculated doses that met the NRC acceptance criteria for doses at the Exclusion Area Boundary (EAB), the Low-Population Zone (LPZ), and the control room.

The staff reviewed the licensee's calculations and performed confirmatory calculations to check the acceptability of the licensee's methodology and resulting doses. As part of their review, the staff calculated the doses resulting from a main steamline break accident using the methodology associated with Standard Review Plan (SRP) 15.1.5, Appendix A. The staff performed two separate assessments. One was based upon a pre-existing iodine spike activity level of 60 μCi/g of dose equivalent ^{131}I in the primary coolant and the other was based upon an accident-initiated iodine spike. For the accident-initiated spike case, the staff assumed that the primary coolant activity level was 1.0 μCi/g of dose-equivalent ^{131}I. The accident initiated an increase in the release rate of iodine from the fuel by a factor of 500 over the normal release rate to maintain an activity level of 1.0 μCi/g of dose-equivalent ^{131}I in the primary coolant. For these two cases, the staff calculated the thyroid doses for individuals located at the EAB and the LPZ and thyroid doses to the control room operator. The parameters that were utilized in the staff's assessment are presented in Table 1.

Using the licensee's estimated allowable leak rate for the faulted steam generator of 4.64 gpm, the staff calculated that the thyroid doses to the control room operator would exceed the guidelines of SRP 6.4 of NUREG-0800 (acceptance criteria of 30 rem thyroid to the control room operator). In performing its calculations, the staff assumed that the accident-initiated iodine spike continued for the duration of the accident (8 hours). In its October 20, 1997, supplemental submittal the licensee revised its calculation so that the duration of the iodine spike would last for 8 hours which resulted in a 1.66 gpm allowable primary-to-secondary leakage. The values the staff used for the efficiencies of the control room ventilation filters were reduced by 1 percent (below the licensee's numbers) to account for bypass flow. For the thyroid dose to the control room operator to be within the 30-rem limit, the staff calculated that the maximum allowable leakage from the faulted steam generator would have to be less than or equal to 3.4 gpm for the pre-existing case (giving a control room dose of 29.7 rem thyroid) or 1.42 gpm for the accident-initiated case (giving a dose of 29.6 rem thyroid)(both of these leakage values are for an RCS temperature of 578 degrees F). Since the lower of these two calculated leakages is 1.42 gpm, the accident-initiated case is the limiting case.

Using a leak rate of 1.42 gpm for both the pre-existing and accident-initiated cases (and maintaining the specific activity limits of dose-equivalent ^{131}I in the primary coolant at 1.0 μCi/g for the 48-hour limit and at 60 μCi/g for the maximum instantaneous limit), the staff's calculations showed that the thyroid doses at the EAB and LPZ would be less than the guidelines established by SRP 15.1.5, Appendix A of NUREG-0800 (acceptance criterion of 300 rem thyroid dose at the EAB and LPZ for the pre-existing spike case and 30 rem thyroid dose at the EAB and LPZ for the accident-initiated spike case). The control room operator thyroid dose would be less than the guidelines of SRP 6.4 of NUREG-0800 (acceptance criterion of 30 rem thyroid to the control room operator)(see Table 2).

In order for the licensee not to exceed the 30 rem thyroid dose limit to the control room operator (while maintaining the specific activity limits of dose-equivalent ^{131}I in the primary coolant at 1.0 μCi/g for the 48-hour limit and at 60 μCi/g for the maximum instantaneous limit), the staff has determined that the licensee must limit the maximum allowable leakage in the faulted steam generator to 1.42 gpm (at an RCS temperature of 578 degrees F). In order to comply with this limit, by letter dated October 24, 1997, the licensee proposed the following license condition.

> NSP will assure that during the implementation of steam generator repairs utilizing the voltage based repair criteria, the total calculated primary to secondary side leakage from the faulted steam generator, under main steam line break conditions (outside containment and upstream of the main steam isolation valves), will not exceed 1.42 gallons per minute (based on a reactor coolant system temperature of 578°F).

This license condition will ensure that doses to the control room operator will be maintained within the 30 rem thyroid limit under main steamline break conditions by ensuring that the leakage from the faulted steam generator will not exceed 1.42 gpm (based on an RCS temperature of 578 degrees F). Therefore, the staff finds this license condition to be acceptable and is including it in Appendix B to the licenses.

3.6 Proposed TS Changes

In order to incorporate a voltage-based steam generator repair criteria, the licensee has proposed the following changes to the TS.

1. Proposed Changes to TS 3.1.C.2.e.

   The limit for total reactor coolant system to secondary coolant system leakage through both steam generators of 1.0 gallon per minute will be changed to a limit of 150 gallons per day of primary-to-secondary leakage through any one steam generator.

2. Proposed new TS 4.12.B.5.

   "Indications left in service as a result of application of tube support plate voltage-based repair criteria shall be inspected by bobbin coil probe during all future refueling outages."

3. Proposed new TS 4.12.B.6.

   "Implementation of the steam generator tube/tube support plate repair criteria requires a 100 percent bobbin coil inspection for hot leg and cold leg tube support plate intersections down to the lowest cold leg tube support plate with known outside diameter stress corrosion cracking (ODSCC) indications. The determination of the lowest cold leg tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20 percent random sampling of tubes inspected over their full length."

4. Proposed Changes to TS 4.12.D.1.(f).

   An exemption is added for the voltage-based repair criteria. "This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 4.12.D.4 for the repair limit applicable to these intersections."

5. Proposed new TS 4.12.D.4.

   This new section provides the detailed requirements for the voltage-based repair criteria.

6. Proposed new TS 4.12.E.5.

   This new section provides the detailed reporting requirements for the voltage-based repair criteria.

7. Correction to TS 4.12.C.1.

   A typographical error in the word "category" has been corrected in Section 4.12.C.1.

8. Proposed change to Bases

   The Bases for TS 3.1.C and 4.12 are revised to incorporate the voltage-based repair criteria.

   A typographical omission of the word "or" was corrected in the last sentence of the second paragraph of Bases page B.4.12-1.

9. Proposed change to Index

   The Index has been revised to reflect pagination changes.

The staff has reviewed the TS changes discussed above and finds that they consistently incorporate the voltage-based repair criteria per the requirements of NRC Generic Letter 95-05 as previously discussed in this safety evaluation. The staff also concludes that adequate structural and leakage integrity can be assured, consistent with applicable regulatory requirements, for indications to which the voltage-based repair criteria will be applied. Therefore, the proposed changes are acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Minnesota State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (62 FR 43371). In addition, the amendments change reporting requirements. Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: November 18, 1997

Attachments: 1. Table 1
2. Table 2

TABLE 1
INPUT PARAMETERS FOR PRAIRIE ISLAND UNITS 1 AND 2 EVALUATION OF MAIN STEAMLINE BREAK ACCIDENT

1. Primary Coolant Concentration of 60 μCi/g of Dose Equivalent ^{131}I

<u>Pre-existing Spike Value (μCi/g)</u>

^{131}I = 49.8
^{132}I = 14.6
^{133}I = 60.2
^{134}I = 9.2
^{135}I = 33.1

2. Primary and Secondary Coolant Specifications

Primary Coolant Volume (ft^3)	5,227.39
Primary Coolant Temperature (°F)	578
Secondary Coolant Steam Mass (lbm)	107,000
Secondary Coolant Liquid Mass (lbm)	5,700
Secondary Coolant Steam Temperature (°F)	510
Secondary Coolant Feedwater Temperature (°F)	427.3

3. TS Limits for DE ^{131}I in the Primary and Secondary Coolant

Maximum Instantaneous DE ^{131}I Concentration (μCi/g)	60.0
Primary Coolant DE ^{131}I Concentration (μCi/g)	1.0
Secondary Coolant DE ^{131}I Concentration (μCi/g)	0.1

4. TS Value for the Primary to Secondary Leak Rate

Primary to secondary leak rate, maximum any SG (gpd)	150
Primary to secondary leak rate, both SGs (gpd)	300

5. Maximum Primary to Secondary Leak Rate to the Faulted and Intact SGs

Faulted SG (gpm)	1.42
Intact SG (gpm/SG)	0.1

6. Iodine Partition Factor

Faulted SG	1.0
Intact SG	0.01

7. Steam Released to the Environment

Faulted SG (0 - 2 hours)	109,155 lbs
Intact SGs (0 - 2 hours)	254,400 lbs
Intact SGs (2 - 8 hours)	486,000 lbs

8. Letdown Flow Rate (gpm) 40

9. Release Rate for 1.0 μCi/g of Dose Equivalent ^{131}I

	Release Rate (Ci/hr)	500X Release Rate (Ci/hr)
^{131}I =	5.62	2810
^{132}I =	9.06	4530
^{133}I =	9.9	4950
^{134}I =	13.3	6640
^{135}I =	9.55	4770

10. Atmospheric Dispersion Factors

	sec/m^3
EAB (0-2 hours)	* 9.8×10^{-4}
LPZ (0-8 hours)	1.77×10^{-4}
Control Room (0-8 hours)	5.58×10^{-3}

11. Control Room Parameters

Filter Efficiency (%)	
Air intake filter	0
Air recirculation filter	
Elemental	90
Organic	90
Particulate	95
Volume (ft^3)	165,000
Makeup flow (cfm)	
Mode 1 (0-2 minutes)	1,835
Mode 2 (after 2 minutes)	0
Recirculation Flow (cfm)(filtered)	
Mode 1 (0-2 minutes)	0
Mode 2 (after 2 minutes)	3,600
Unfiltered Inleakage (cfm)	
Mode 1 (0-2 minutes)	165
Mode 2 (after 2 minutes)	165
Occupancy Factors	
0-1 day	1.0

* NRC staff calculated value

TABLE 2 - THYROID DOSES FROM PRAIRIE ISLAND UNITS 1 AND 2 MAIN STEAM LINE BREAK ACCIDENT (REM) (VALUES CALCULATED BY NRC STAFF)

	DOSE	
LOCATION	**Pre-Existing Spike**	**Accident-Initiated Spike****
EAB	10.0*	7.34
LPZ	7.0*	17.1
Control Room **	12.4	29.6

* Acceptance Criterion = 300 rem thyroid
** Acceptance Criterion = 30 rem thyroid