Docket No. 50-282
50-306

JUL 18 1975

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Northern States Power Company
Attn: Mr. L. O. Mayer
Director of Nuclear Support Systems
414 Nicollet Avenue
Minneapolis, Minnesota 55401

Gentlemen:

Pursuant to a Decision issued by the Atomic Safety and Licensing Board on May 1, 1975, in connection with the Prairie Island Nuclear Generating Plant, Units 1 and 2, the Nuclear Regulatory Commission has issued Amendment No. 7 to Facility Operating License No. DPR-42, and Amendment No. 2 to Facility Operating License No. DPR-60 revising the inservice inspection program for steam generator tubes.

Copies of the Amendments and a related Federal Register notice are enclosed.

Sincerely,

Original signed by
K. Kniel
Karl Kniel, Chief
Light Water Reactors Branch 2-2
Division of Reactor Licensing

Enclosures:
1. Amendment No. 7 to DPR-42
2. Amendment No. 2 to DPR-60
3. Federal Register Notice

O/P
(1)

ccs: See next page

exc. Par. 3a.

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that pursuant to a Supplemental Initial Decision by the Atomic Safety and Licensing Board dated May 1, 1975, the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-42, and Amendment No. 2 to Facility Operating License No. DPR-60, issued to Northern States Power Company, which revised the licenses for operation of the Prairie Island Nuclear Generating Plant, Units 1 and 2 (the facility), located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revised Appendix A to the Technical Specifications revising the inservice inspection program for steam generator tubes.

For further details with respect to this action, see (1) Atomic Safety and Licensing Board Supplemental Initial Decision dated May 1, 1975, (2) Amendment No. 7 to License No. DPR-42, with attachments, and (3) Amendment No. 2 to License No. DPR-60, with attachments. These items are available

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-282

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 7
License No. DPR-42

Pursuant to a Supplemental Initial Decision by the Atomic Safety and Licensing Board dated May 1, 1975, the Nuclear Regulatory Commission has issued Amendment No. 7 to Facility Operating License DPR-42 by amending Paragraph 2.C.(2) to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 8.

This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl Kniel, Chief
Light Water Reactors Branch 2-2
Division of Reactor Licensing

Attachment:
Change No. 7 to Appendix A
Technical Specifications

Date of Issuance:

JUL 18 1975

AMERICAN REVOLUTION BICENTENNIAL 1776-1976

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

NORTHERN STATES POWER COMPANY

DOCKET NO. 50-306

PRAIRIE ISLAND NUCLEAR GENERATING PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 2
License No. DPR-60

Pursuant to a Supplemental Initial Decision by the Atomic Safety and Licensing Board dated May 1, 1975, the Nuclear Regulatory Commission has issued Amendment No. 2 to Facility Operating License DPR-60 by amending Paragraph 2.C.(2) to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 8.

This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl Kniel, Chief
Light Water Reactors Branch 2-2
Division of Reactor Licensing

Attachment:
Change No. 7 to Appendix A
Technical Specifications

Date of Issuance: JUL 18 1975

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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-282 AND 50-306

NORTHERN STATES POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

Notice is hereby given that pursuant to a Supplemental Initial Decision by the Atomic Safety and Licensing Board dated May 1, 1975, the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 7 to Facility Operating License No. DPR-42, and Amendment No. 2 to Facility Operating License No. DPR-60, issued to Northern States Power Company, which revised the licenses for operation of the Prairie Island Nuclear Generating Plant, Units 1 and 2 (the facility), located in Goodhue County, Minnesota. The amendments are effective as of the date of issuance.

The amendments revised Appendix A to the Technical Specifications revising the inservice inspection program for steam generator tubes.

For further details with respect to this action, see (1) Atomic Safety and Licensing Board Supplemental Initial Decision dated May 1, 1975, (2) Amendment No. 7 to License No. DPR-42, with attachments, and (3) Amendment No. 2 to License No. DPR-60, with attachments. These items are available

for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Environmental Library of Minnesota, 1222 S. E. 4th Street, Minneapolis, Minnesota 55414.

Single copies of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 18th day of July 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
K. Kniel

Karl Kniel, Chief
Light Water Reactors Branch 2-2
Division of Reactor Licensing

for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Environmental Library of Minnesota, 1222 S. E. 4th Street, Minneapolis, Minnesota 55414.

Single copies of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 18th day of July 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl Kniel, Chief
Light Water Reactors Branch 2-2
Division of Reactor Licensing

CHANGE NO. 7

to

TECHNICAL SPECIFICATIONS

for

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

UNITS 1 AND 2

DOCKET NOS. 50-282 AND 50-306

LICENSE NOS. DPR-42 AND DPR-60

Instructions for page changes, Appendix A

1. Add new pages provided in enclosure (TS3.1-1a and TS 4.12-1 through TS 4.12-7)

2. Replace old pages with revised pages provided in the enclosure (TS-i, TS-ii, TS 3.1-1, TS 3.1-3, TS 3.1-9 , TS 3.1-10, TS 4.2-3, TS 4.2-4, TS 4.2-5, and Table TS 4.2-1, pages 4, 5, and 9).

3. Delete page TS 4.2-3a

TECHNICAL SPECIFICATIONS

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3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the reactor coolant system.

Objective

To specify those limiting conditions for operation of the reactor coolant system which must be met to assure safe reactor operation.

Specification

A. Operational Components

Specification:

1. Coolant pumps

   a. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   b. When the reactor is made critical, except for low power physics tests, both reactor coolant pumps shall be in operation.

   c. If one reactor coolant pump is lost while operating below 10% rated power, the affected reactor shall be made subcritical except for low power physics tests.

   d. If both reactor coolant pumps are lost below 10% rated power, the affected reactor shall be made subcritical.

2. Steam Generators

   a. Both steam generators shall be operable when the average coolant temperature is raised above 350°F. Operability of steam generator tubes shall be determined in accordance with Technical Specification 4.12. 7

b. If one steam generator becomes inoperable with the reactor coolant temperature above 350°F, the affected reactor shall be brought to a cold shutdown condition.

The pressurizer is needed to maintain acceptable system pressure during normal plant operation, including surges that may result following anticipated transients. Each of the pressurizer safety valves is designed to relieve 325,000 lbs per hour of saturated steam at the valve set point. Below 350°F and 450 psig in the reactor coolant system, the residual heat removal system can remove decay heat and thereby control system temperature and pressure. If no residual heat were removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve therefore provides adequate defense against over-pressurization of the reactor coolant system for reactor coolant temperatures less than 350°F. The combined capacity of both safety valves is greater than the maximum surge rate resulting from complete loss of load.[1]

"Steam Generator Tube Surveillance", Technical Specification 4.12, identifies steam generator tube imperfections having a depth ≥50% of the 0.050-inch tube wall thickness as being unacceptable for power operation. The results of steam generator burst and tube collapse tests submitted to the staff have demonstrated that tubes having a wall thickness greater than 0.025-inch have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents.[2] 7

References

[1]FSAR, Section 14.1.9

[2]Testimony by J. Knight in the Prairie Island Public Hearing on January 28, 1975.

C. LEAKAGE

Specification:

The following specification applies to each unit of the plant.

1. Detected or suspected leakage from the reactor coolant system shall be investigated and evaluated. At least two means of detection shall be available, one of which must depend on the detection of radionuclides in the containment.

2. If the leakage rate, from other than controlled leakage sources, such as the reactor coolant pump controlled leakage seals, exceeds 1 gpm and the source of the leakage is not identified within 4 hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.

3. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in C.4 below.

4. If it is determined that leakage exists through a fault which has developed in a reactor coolant system component body, pipe well, vessel wall, or pipe weld, and that the fault cannot be isolated, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.

5. If the total leakage, other than leakage from controlled sources, exceeds 10 gpm reactor hot shutdown shall be initiated immediately. If the condition is not corrected within 8 hours the reactor shall be taken to cold shutdown and remain in cold shutdown until the condition is corrected.

6. If steam generator leakage between the primary coolant system and the secondary coolant system of a unit exceeds 1.0 gallon per minute (gpm), that unit shall be brought to a cold shutdown and an inservice steam generator tube inspection shall be performed in accordance with Technical Specification 4.12. 7

Basis

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, the gaseous and liquid waste disposal systems, the component cooling system, and the chemical and volume control system.

Detection of leaks from the reactor coolant system is by one or more of the following:[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.[2]

A leak rate of 1 gpm corresponds 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.[3] 7

References

[1]FSAR Section 6.5

[2]FSAR Section 11.2.3

[3]Testimony by J. Knight in the Prairie Island public hearing on January 28, 1975, pp 13-17.

The following corrective measures shall be applied:

(a) An evaluation of the effect of any corroded area upon the structural integrity of the component shall be performed in accordance with the provisions of Article IS-311 of Section XI Code.

(b) Repairs of corroded areas, if necessary, shall be performed in accordance with the procedures of Article IS-400 of Section XI Code.

3. The structural integrity of the primary system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence as a result of the inspections listed in Table TS.4.2-1 that defects have initiated or grown, shall be investigated, including evaluation of comparable areas of the primary system. In the event further unacceptable structural defects are revealed, all remaining system components or piping in this category shall be examined to the extent specified in that examination category. 7

C. Records

Detailed records of each inspection including the preoperational base line inspection, shall be maintained to allow comparison and evaluation of future inspections in accordance with Specification 6.6.B.10.

Basis

The inspection program, where practical, is in compliance with Section XI of the ASME Code 1971 Edition and Summer and Winter 1971 addenda for inservice inspection of nuclear reactor coolant system. Though examinations in certain areas are desirable, it should be recognized that equipment and techniques to perform the inspection are still in development. In all areas scheduled for volumetric examination, a detailed preservice mapping will be conducted using techniques anticipated to be used for post-operation examinations. The areas indicated for inspection represent those of representative stress levels and therefore, will serve to indicate potential problems before significant flows develop there or at other areas. As more experience is gained in operation of pressurized water reactors, the time schedule and location of inspection may be altered or, should new techniques be developed, consideration may be given to incorporate these new techniques into this inspection program.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of most primary loop components except the reactor vessel. The reactor vessel presents special problems because of the radiation levels and the requirement for remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps[1] have been incorporated into the design and manufacturing procedures in preparation for nondestructive test techniques which may be available in the future.

The techniques anticipated for inservice inspection include visual inspections, ultrasonic, radiographic, magnetic particle, and dye penetrant testing of selected parts during refueling periods or other appropriate plant outages. 7

The program outlined in Table TS.4.2-1 is discussed in detail in FSAR Section 4. The extent to which systems will be subject to the examination requirements of ASME Table IS-251 have been determined in accordance with the criteria of ASME Category IS-120. The exclusion criteria of ASME Category IS-121 have been applied to determine which parts of systems or components are subject to a visual examination for evidence of leakage during the system hydrostatic test. A description of the system boundaries, delineating those parts subject to surface examination and those parts requiring visual inspection during hydro, are given in the attached Tables TS.4.2-2, 3, and 4.

It is anticipated that the data collected during the first 5 years of operation along with the overall operating experience will provide a suitable basis for determining the inspection program to be implemented for the lifetime of the facility. The results of this study together with the proposed lifetime program will be submitted to the AEC in accordance with Table TS.6.7-1, Item 9. 7

IS-260 Item	Component	Method of Examination	Extent and Frequency
3.0	STEAM GENERATORS AND CLASS A HEAT EXCHANGERS		
	STEAM GENERATOR		
3.1	Channel Head to Tubesheet Weld	V U.T.	Examinations Performed During 10 Year Period Shall Cumulatively Cover 5 Percent of Total Length of Weld
3.3	Nozzle-to-Safe-End Welds	V U.T. P.T.	Examine 100 Percent of One Safe-End in First 3 1/3 Year Period, 100 Percent of Second Safe-End in Next 3 1/3 Year Period and Remaining Two by the End of the 10 Year Period
3.5	Manway Bolting	V	The Examinations Performed Over the 10 Year Interval Shall Cumulatively Cover All the Bolts
3.7	Cladding	V	One Patch (36 square inches) Inside Each Manway Will be Examined at the first Refueling and at the End of the 10 Year Interval

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7

3.1	REGENERATIVE HEAT EXCHANGER		
	Tubesheet-Shell Welds	V U.T.	5 Percent of Each Weld is to be Cumulatively Examined over the 10 Year Interval
3.1	EXCESS LETDOWN HEAT EXCHANGER		
	Shell Welds	V During Hydro	Vessel Examined for Evidence of Leakage During Hydro at the End of 10 Year Interval

NOTES (contd)

4. With the exception of those components or areas for which the examination may be deferred to the end of the inspection interval, at least 25% of the required examinations shall have been completed by the expiration of one-third of the inspection interval (with credit for no more than 33-1/3% if additional examinations are completed) and at least 50% shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66-2/3%). The remaining required examinations shall be completed by the end of the inspection interval. Successive inspections shall meet the requirements of paragraph ISI-243 of the ASME rules for in-service inspection of nuclear reactor coolant systems.

7

4.12 STEAM GENERATOR TUBE SURVEILLANCE

Applicability

Applies to inservice surveillance of the steam generator tubes.

Objective

To assure the continued integrity of the steam generator tubes that are a part of the primary coolant pressure boundary.

Specification

Steam generator tubes in each unit shall be determined operable by the following inspections, corrective measures, and reports.

A. Inspection Requirements

1. Baseline Inspection

(a) All tubes in the steam generators in the unit shall be inspected[1] by eddy current and/or alternate techniques prior to service to establish a baseline condition of the tubing.

(b) For operating units without an initial baseline inspection the first inservice inspection performed according to Specifications 4.12.A.2 and 4.12.A.3 will define the baseline condition for subsequent inspections.

(c) For operating units having a major change in secondary water chemistry (e.g., phosphate to volatile treatment) an inservice inspection shall be performed in each steam generator of that unit in accordance with Specifications 4.12.A.2 and 4.12.A.3 prior to resumption of power operation to define baseline conditions for subsequent inspections.

2. Sample Selection and Testing

Selection and testing of steam generator tubes shall be made on the basis of the following:

(a) The baseline inspection required by Specification 4.12.A.1(a) shall include all the tubes in the steam generators.

[1]Entry from the hot-leg side with examination from the point of entry completely around the U-bend to the top support of the cold-leg is considered sufficient to constitute a tube inspection.

(b) Tubes for the inspection of operating units required by Specification 4.12.A.1(b) shall be selected on a random basis except that one-third of the selected tubes shall be inspected from the cold-leg side of the steam generators.

(c) At least 3% of the total number of tubes in each steam generator shall be tested during each inspection (see Specification 4.12.A.1 and 4.12.A.4).

(d) Following the first two steam generator tube inservice inspection, inspections shall routinely concentrate on the hot-leg side of the steam generators.

(e) Every inspection subsequent to the baseline inspection shall include all nonplugged tubes that previously had defect indications and shall also include tubes in those areas (including the cold leg side) where design and experience have indicated potential problems.

3. Supplementary Sampling Requirements

(a) If the eddy current inspection pursuant to Specification 4.12.A.2 indicates that more than 10%[2] of the inspected tubes have detectable wall penetration (>20%) or that one or more of the inspected tubes have an indication in excess of 50% wall penetration, an additional 3% of the tubes shall be inspected. At least 50% of these additional tubes selected for eddy current inspection shall be from those areas of the tube sheet array where tubes with imperfections were found.

(b) If the additional inspection pursuant to Specification 4.12.A.3(a) indicates that more than 10% of these additional inspected tubes have detectable wall penetration (>20%) or one or more of these additionally inspected tubes has an indication in excess of 50% wall penetration,

[2] In all inspections, tubes with previously detected imperfections that exhibit significant (>5%) further wall penetration must be included in the 10%.

additional tubes (no less than 3% of the total tubes in that steam generator) in the area of the tube sheet array where tubes with imperfections were found shall be inspected.

4. Inspection Intervals

(a) The first inservice inspection of steam generators shall be performed after 6 months of operation but before 20 calendar months.

(b) Subsequent inservice inspections shall be not less than 12 nor more than 20 calendar months after the previous inspection unless a shorter inspection interval is required to assure that tube defects having a remaining wall thickness less than the minimum acceptable wall thickness as defined in Specification 4.12.A.5(a) will not develop prior to the next inspection.

(c) If two consecutive inspections, not including the baseline inspection, result in no additional tubes with detectable wall penetration (>20%), and no significant (>5%) further penetration of tubes with previous indications, the inspection frequency may revert to 1 inspection each 40 months. If 2 consecutive inservice inspections of the cold-leg side of a steam generator find no additional tubes with detectable wall penetration (>20%) and no signicant further penetration of tubes with previous indications the inspection frequency for cold-leg tubes may revert to 1 inspection each 10 years.

(d) Unscheduled inspections shall be conducted in each steam generator in accordance with Specifications 4.12.A.2 and 4.12.A.3 in the event of primary-to-secondary tube leakage exceeding Specification 3.1.C.6, a seismic occurrence greater than an operating basis earthquake,[3] a loss-of-coolant accident requiring actuation of engineered safeguards, or a major steam line or feedwater line break.

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[3]As defined in Appendix A, "Seismic and Geologic Siting Criteria for Nuclear Power Plants" to 10 CFR Part 100 "Reactor Site Criteria."

5. Acceptance Limits

(a) Definitions

Imperfection is an exception to the dimensions, finish, or contour required by drawing or specification. A detectable imperfection has a wall penetration >20% of the nominal tube wall thickness.

Defect is an imperfection that extends partly through the tube wall. The minimum acceptable remaining wall thickness at a defect is 50% of the nominal tube wall thickness.

Plugging limit is the imperfection depth at or beyond which plugging of the tube must be performed to make the steam generator operable. The plugging limit is less than the defect that results in the minimum acceptable wall thickness by an amount equal to the allowance for localized corrosion.

Plugging criteria are those calculational and analytical procedures used to arrive at the plugging limit. These criteria shall be submitted for approval by the Commission prior to use.

(b) If in the inspection performed under Specification 4.12.A.2 less than 10% of the tubes inspected have detectable wall penetration (>20%) and no tube exceeds the plugging limit, plant operation may resume.

(c) If in the inspections performed under Specification 4.12.A.3 less than 10% of the total tubes inspected have detectable wall penetration (>20%) and no more than three tubes exceed the plugging limit, plant operation may resume after required corrective measures have been taken.

(d) If in the inspections performed under Specification 4.12.A.3 more than 10% of the total tubes inspected have detectable wall penetration (>20%) or more than three of the tubes inspected exceed the plugging limit, the situation shall be reported in accordance with Specifications 4.12.C.2 and 6.3.

(e) If in the inspections performed under Specification 4.12.A.3 one or more of the tubes inspected has a defect for which the remaining wall thickness is less than the minimum acceptable wall thickness, the situation shall be reported in accordance with Specifications 4.12.C.3 and 6.3 for resolution and approval of the proposed remedial action.

B. Corrective Measures

Tubes shall be plugged prior to resumption of unit operation if they contain (a) imperfections exceeding the plugging limit, or (b) through-the-wall cracks detected during plant operation or during the water leakage test subsequent to tube plugging.

C. Reports

1. The results of these steam generator tube inservice inspections shall be included in the Operating Report for the period in which this inspection was completed.

2. Reports required by Specification 4.12.A.5(d) shall include:

   (a) The number and extent of tubes inspected

   (b) The location and percent of wall thickness penetration for each indication of an imperfection.

   (c) Identification of all tubes that are plugged.

   (d) Determination of and bases for the date of the next inservice inspection, including an allowance for localized corrosion based on that unit's previous steam generator tube inspections and secondary water treatment.

3. Reports required by Specification 4.12.A.5(e) shall include information in Specification 4.12.C.2 and planned investigations to determine the cause of the event and corrective measures to prevent recurrence.

Basis

The program for inservice inspection of steam generator tubes is based on Regulatory Guide 1.83 dated June 1974 pursuant to the May 1, 1975 Supplemental Initial Decision of the Atomic Safety Licensing Board in the matter of Northern States Power Company, Prairie Island Nuclear Generating Plant, Units 1 and 2. Inservice inspection of steam generators is essential in order to monitor the integrity of the tubing and to maintain surveillance in the event that there is evidence of mechanical damage or progressive deterioration due to design, manufacturing errors, or chemical imbalance. Inservice 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.

The requirements in Specification 4.12.A.1 for baseline inspections have been included to make the specification complete, although the baseline inspections have already been performed for Units 1 and 2. The Unit 1 baseline inspection was performed after approximately 3 effective full power months of operation when a change was made from phosphate treatment to all volatile treatment of secondary coolant. Unit 2 baseline inspection was made prior to initial operation and has been operated with all volatile treatment of secondary coolant. Unit 2 baseline inspection was made prior to initial operation and has been operated with all volatile treatment of secondary coolant.

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 = 1.0 gpm). Cracks having a primary-to-secondary leakage less than 1.0 gpm during operation will have an adequate margin of safety aginst failure due to loads imposed by design basis accidents.[1] Operating plants have demonstrated that primary-to-secondary leakage as low as 0.1 gpm will be detected by radiation monitors of steam generator blowdown. Leakage in excess of 1.0 gpm will require plant shutdown and an unscheduled eddy current inspection, during which the leaking tubes will be located and plugged.

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

Wastage-type defects are unlikely with the planned all volatile treatment (AVT) of secondary coolant. However, even if this type of defect occurs it will be found during scheduled inservice steam generator tube inspections. 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.A.5(a), is 50% of the tube nominal wall thickness (0.050 inches). Wastage type defects having a wall thickness greater than 0.025 inches will have adequate margins of safety against failure due to loads imposed by normal plant operation and design basis accidents. [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.[2]

A defect having less than the minimum acceptable wall thickness (0.025 inches) as defined by Specification 4.12.A.5(a) is a defect that could have resulted in tube failure if a design basis accident had occurred. If such a tube is found, the specifications require that a report be made to the Commission for approval of the licensee's planned investigation to determine the cause of such severe defects and proposed corrective measures to prevent recurrence. Such cases will be considered by the Commission on a case-by-case basis and may result in a requirement for extensive analysis, laboratory examinations, tests, additional eddy current inspection, and additional Technical Specifications.

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[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