May 3, 1991

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Docket No. 50-305

Mr. Ken H. Evers
Manager - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Evers:

SUBJECT: AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-43 (TAC NO. 80005)

The Commission has issued the enclosed Amendment No. 93 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications in response to your application dated March 19, 1991, as supplemented by letter dated April 26, 1991.

The amendment provides clarification on how motorized rotating ancake coil (MRPC) eddy current indications in the steam generator (SG) hot leg tubesheet crevice area will be dispositioned during your spring 1991 refueling outage. The method of dispositioning crevice area indications described in the enclosed Technical Specification (TS) change is an interim measure to be utilized for the upcoming 1991-1992 operating cycle only.

If you wish to repair tubes with eddy current indications located within the partial-depth roll area, your repair method must be submitted in accordance with 10 CFR Part 50 and Technical Specification 4.2.b.4.a. As described in the basis for Technical Specification 4.2.b.4, please provide the staff a minimum of 90 days for the review of your proposed repair method.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,
Original Signed By:
Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 93 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures: See next page

*See Previous Concurrence

LA/PDIII-3/DRPW	PE/PDIII-3/DRPW	PM/PDIII-3/DRPW	D/PDIII-3/DRPW
PKreutzer	AHansen	AGodyJr./bj	JHannon
5/3/91	5/3/91	5/3/91	5/3/91
*D/EMCB	*D/PRPB	*D/SRXB	*OGC-OWFN
CYChen	LCunningham	RJones	JTHull
4/26/91	4/26/91	4/26/91	4/30/91

DOCUMENT NAME: 80005.AMD

CP-1

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

May 3, 1991

Docket No. 50-305

Mr. Ken H. Evers
Manager - Nuclear Power
Wisconsin Public Service Corp.
P. O. Box 19002
Green Bay, Wisconsin 54037-9002

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1. Amendment No. 93 to License No. DPR-43
2. Safety Evaluation

cc w/enclosures: See next page

Mr. Ken H. Evers
Wisconsin Public Service Corporation

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

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 93
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensees) dated March 19, 1991, as supplemented by letter dated April 26, 1991, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 93, are hereby incorporated in the license. The licensees shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance, and is to be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Anthony T. Gody, Jr.

Anthony T. Gody, Jr., Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV/V
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of issuance: May 3, 1991

ATTACHMENT TO LICENSE AMENDMENT NO. 93

FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

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

REMOVE	INSERT
TS ii	TS ii
TS iii	TS iii
4.2-1	4.2-1
4.2-2	4.2-2
4.2-2.a	-
4.2-3 through 4.2-11	4.2-3 through 4.2-11

Section	Title	Page TS
3.3	Engineered Safety Features and Auxiliary Systems	3.3-1
	3.3.a Accumulators	3.3-1
	3.3.b Safety Injection and Residual Heat Removal Systems	3.3-2
	3.3.c Containment Cooling Systems	3.3-4
	3.3.d Component Cooling System	3.3-6
	3.3.e Service Water System	3.3-6
3.4	Steam and Power Conversion System	3.4-1
3.5	Instrumentation System	3.5-1
3.6	Containment System	3.6-1
3.7	Auxiliary Electrical Systems	3.7-1
3.8	Refueling	3.8-1
3.9	Deleted	
3.10	Control Rod and Power Distribution Limits	3.10-1
	3.10.a Shutdown Reactivity	3.10-1
	3.10.b Power Distribution Limits	3.10-1
	3.10.c Quadrant Power Tilt Limits	3.10-5
	3.10.d Rod Insertion Limits	3.10-5
	3.10.e Rod Misalignment Limitations	3.10-6
	3.10.f Inoperable Rod Position Indicator Channels	3.10-6a
	3.10.g Inoperable Rod Limitations	3.10-6a
	3.10.h Rod Drop Time	3.10-7
	3.10.i Rod Position Deviation Monitor	3.10-7
	3.10.j Quadrant Power Tilt Monitor	3.10-7
	3.10.k Inlet Temperature	3.10-7a
	3.10.l Operating Pressure	3.10-7a
	3.10.m Coolant Flow Rate	3.10-7a
3.11	Core Surveillance Instrumentation	3.11-1
3.12	Control Room Postaccident Recirculation System	3.12-1
3.14	Shock Suppressors (Snubbers)	3.14-1
3.15	Deleted	3.15-1
4.0	Surveillance Requirements	4.1-1
4.1	Operational Safety Review	4.1-1
4.2	ASME Code Class In-service Inspection and Testing	4.2-1
	4.2.a ASME Code Class 1, 2, and 3 Components and Supports	4.2-1
	4.2.b Steam Generator Tubes	4.2-2
	4.2.b.1 Steam Generator Sample Selection and Inspection	4.2-3
	4.2.b.2 Steam Generator Tube Sample Selection and Inspection	4.2-3

Amendment No. ~~64~~/~~91~~ 93

Section	Title	Page TS
4.2.b.3	Inspection Frequencies	4.2-4
4.2.b.4	Plugging Limit Criteria	4.2-5
4.2.b.5	Hot Leg Tubesheet Crevice Plugging Limit Criteria	4.2-6
4.2.b.6	Reports	4.2-6
4.3	Deleted	4.3-1
4.4	Containment Tests	4.4-1
4.4.a	Integrated Leak Rate Tests (Type A)	4.4-1
4.4.b	Local Leak Rate Tests (Type B and C)	4.4-3
4.4.c	Shield Building Ventilation System	4.4-6
4.4.d	Auxiliary Building Special Ventilation System	4.4-7
4.4.e	Containment Vacuum Breaker System	4.4-7
4.5	Emergency Core Cooling System and Containment Air Cooling System Tests	4.5-1
4.5.a	System Tests	4.5-1
4.5.a.1	Safety Injection System	4.5-1
4.5.a.2	Containment Vessel Internal Spray System	4.5-2
4.5.a.3	Containment Fan Coil Units	4.5-2
4.5.b	Component Tests	4.5-2
4.5.b.1	Pumps	4.5-2
4.5.b.2	Valves	4.5-3
4.6	Periodic Testing of Emergency Power System	4.6-1
4.6.a	Diesel Generators	4.6-1
4.6.b	Station Batteries	4.6-2
4.7	Main Steam Isolation Valves	4.7-1
4.8	Auxiliary Feedwater System	4.8-1
4.9	Reactivity Anomalies	4.9-1
4.10	Deleted	
4.11	Deleted	
4.12	Spent Fuel Pool Sweep System	4.12-1
4.13	Radioactive Materials Sources	4.13-1
4.14	Testing and Surveillance of Shock Suppressors (Snubbers)	4.14-1
4.15	Deleted	4.15-1
4.16	Reactor Coolant Vent System Tests	4.16-1
4.17	Control Room Postaccident Recirculation System	4.17-1

4.2 ASME CODE CLASS IN-SERVICE INSPECTION AND TESTING

APPLICABILITY

Applies to in-service structural surveillance of the ASME Code Class components and supports and functional testing of pumps and valves.

OBJECTIVE

To assure the continued integrity and operational readiness of ASME Code Class 1, 2 and 3 components.

SPECIFICATION

a. ASME Code Class 1, 2 and 3 Components and Supports

1. In-service inspection of ASME Code Class 1, Class 2 and Class 3 components and supports shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i). The testing and surveillance of shock suppressors (snubbers) is detailed in Technical Specification Sections 3.14 and 4.14.

2. In-service testing of ASME Code Class 1, Class 2 and Class 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR Part 50, Section 50.55a(g), except where relief has been granted by the Commission pursuant to 10 CFR Part 50, Section 50.55a(g)(6)(i).

3. Surveillance testing of pressure isolation valves:

   a. Periodic leakage testing[1] on each valve listed in Table TS 3.1-2 shall be accomplished prior to entering the operating mode after every time the plant is placed in the cold shutdown condition for refueling, after each time the plant is placed in a cold shutdown condition for 72 hours if testing has not been accomplished in the preceding 9 months, and prior to returning the valve to service after maintenance, repair, or replacement work is performed.

---

[1] To satisfy ALARA requirements, leakage may be measured indirectly (as from the performance of pressure indicators) if accomplished in accordance with approved procedures and supported by computations showing that the method is capable of demonstrating valve compliance with the leakage criteria.

b. Whenever integrity of a pressure isolation valve listed in Table TS 3.1-2 cannot be demonstrated, the integrity of the remaining pressure isolation valve in each high pressure line having a leaking valve shall be determined and recorded daily. In addition, the position of the other closed valve located in the high pressure piping shall be recorded daily.

b. Steam Generator Tubes

Examinations of the steam generator tubes shall be in accordance with the in-service inspection program described herein. The following terms are defined to clarify the requirements of the inspection program.

Imperfection is an exception to the dimension, finish, or contour required by drawing or specification.

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

% Degradation is an estimated % of the tube wall thickness affected or removed by degradation.

Degraded Tube means a tube contains an imperfection ≥20% of the nominal wall thickness caused by degradation.

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

Distorted Indication is a possible tube wall loss condition that is unquantifiable with a numeric percent call due to the existing signal characteristics.

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

Tube is the Reactor Coolant System pressure boundary past the hot leg side of the tubesheet and before the cold leg side of the tubesheet.

Tubesheet Crevice Region is, for the purposes of applying the in-service inspection program plug and repair criteria, the area from the tube end to 5 inches below the top of the tubesheet.

Plugged Tube is a tube intentionally removed from service by plugging in the hot and cold legs because it is defective, or because its continued integrity could not be assured.

Repaired Tube is a tube that has been modified to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube.

Squirrel Indications are generally multiple stress corrosion cracks in the roll transition area and mid span of the tubesheet.

1. Steam Generator Sample Selection and Inspection

   The in-service inspection may be limited to one steam generator on a rotating schedule encompassing the number of tubes determined in TS 4.2.b.2.a provided the previous inspections indicated that the two steam generators are performing in a like manner.

2. Steam Generator Tube Sample Selection and Inspection

   The tubes selected for each in-service inspection shall:

   a. Include at least 3% of the total number of nonrepaired tubes, in both steam generators, and 3% of the total number of repaired tubes in both steam generators. The tubes selected for these inspections shall be selected on a random basis except as noted in 4.2.b.2.b.

   b. Concentrate the inspection by selection of at least 50% of the tubes to be inspected from critical areas where experience in similar plants with similar water chemistry indicates higher potential for degradation.

   c. Include the inspection of all non-plugged tubes which previous inspections revealed in excess of 20% degradation. The previously degraded tubes need only be inspected about the area of previous degradation indication if their inspection is not employed to satisfy 4.2.b.2.a and 4.2.b.2.b above.

   d. The second and third sample inspections during each in-service inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tubesheet array and on those portions of the tubes where tubes with imperfections were previously found.

   e. If a tube does not permit the passage of the eddy current inspection probe the entire length and through the U-bend, this shall be recorded and an adjacent tube shall be inspected. The tube which did not allow passage of the eddy current probe shall be considered degraded.

The results of each sample inspection shall be classified into one of the following three categories, and actions taken as described in Table 4.2-2.

<u>Category</u>	<u>Inspection Results</u>
C-1	Less than 5% of the total tubes inspected are degraded tubes, and none of the inspected tubes are defective.
C-2	One or more tubes, but not more than 1% of the total tubes inspected are defective, or between 5% and 10% of the total tubes inspected are degraded tubes.
C-3	More than 10% of the total tubes inspected are degraded tubes or more than 1% of the inspected tubes are defective.

NOTE: In all inspections, previously degraded tubes must exhibit significant (>10%) further wall penetrations to be included in the above percentage calculations.

3. <u>Inspection Frequencies</u>

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

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service 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.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.2-2 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 a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

c. Additional, unscheduled in-service inspections shall be performed on each steam generator in accordance with the first sample inspection specified in Table 4.2-2 during the shutdown subsequent to any of the following conditions:

   1. Primary-to-secondary tube leaks (not including leaks originating from tube-to-tubesheet welds) in excess of the limits of Specifications 3.1.d and 3.4.a.4, or

   2. A seismic occurrence greater than the Operating Basis Earthquake, or

   3. A loss-of-coolant accident requiring actuation of the engineering safeguards, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr, or

   4. A main steam line or feedwater line break, where the cooldown rate of the Reactor Coolant System exceeded 100°F/hr.

d. If the type of steam generator chemistry treatment is changed significantly, the steam generators shall be inspected at the next outage of sufficient duration following 3 months of power operation since the change.

4. <u>Plugging Limit Criteria</u>

The following criteria apply independently to tube[2] and sleeve wall degradation:

a. Any tube[2] which, upon inspection, exhibits tube wall degradation of 50% or more shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Repair methods will be submitted under 10 CFR 50.90 to be incorporated as an amendment in the facility license. The Commission will review the repair method, issue a significant hazards determination, and amend the facility license.

b. Any sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged prior to returning the steam generator to service. Figure 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

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[2] For the 1991-1992 operating cycle only, Specification 4.2.b.4 applies to the tube excluding the hot leg tubesheet crevice region. Refer to Specification 4.2.b.5 for the hot leg tubesheet crevice region criteria.

5. Hot Leg Tubesheet Crevice Plugging Limit Criteria[3]

The following criteria applies to indications in the hot leg tubesheet crevice region:

a. Any tubesheet crevice indication which:

1. Exhibits tube wall degradation of 50% or more with the bobbin coil exam, or

2. Is identified as a multiple circumferential indication or single circumferential indication with the motorized rotating pancake coil (MRPC) exam, or

3. Is identified as a multiple axial indication (MAI) or single axial indication (SAI) with the MRPC exam and is repairable by sleeving within the 27-inch sleeving boundary, or

4. Is identified as a MAI or SAI with MRPC exam and the corresponding bobbin call was either a distorted roll indication, distorted crevice indication or squirrel,

shall be plugged or repaired prior to returning the steam generator to service. If significant general tube thinning occurs, this criterion will be reduced to 40% wall degradation. Repair methods will be submitted under 10 CFR 50.90 to be incorporated as an amendment in the facility license. The Commission will review the repair method, issue a significant hazards determination, and amend the facility license.

b. Any tubesheet crevice indication which is not categorized in Specification 4.2.b.5.a may be left in service provided that the number of crevice indications left in service does not exceed a total of 388 tubes per steam generator.

6. Reports

a. Following each in-service inspection of steam generator tubes, if there are any tubes requiring plugging or repairing, the number of tubes plugged or repaired shall be reported to the Commission within 30 days.

[3] Specification 4.2.b.5 is applicable for the 1991-1992 operating cycle only.

b. The results of the steam generator tube in-service inspection shall be included in the Annual Operating Report for the period in which this inspection was completed. This report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of a degradation.

   3. Identification of tubes plugged.

   4. Identification of tubes repaired.

c. Results of a steam generator tube inspection which fall into Category C-3 require prompt (within 4 hours) notification of the Commission consistent with 10 CFR 50.72(b)(2)(i). A written follow up report shall be submitted to the Commission consistent with Specification 4.2.b.6.a, using the Licensee Event Report System to satisfy the intent of 10 CFR 50.73(a)(2)(ii).

BASIS

The plant was not specifically designed to meet the requirements of Section XI of the ASME Code; therefore, 100% compliance may not be feasible or practical. However, access for in-service inspection was considered during the design and modifications have been made where practical to make provisions for maximum access within the limits of the current plant design. Where practical, the inspection of ASME Code Class 1, Class 2 and Class 3 components is performed in accordance with Section XI of the ASME Code. If a code required inspection is impractical, a request for a deviation from the requirement is submitted to the Commission for approval.

The basis for surveillance testing of the Reactor Coolant System pressure isolation valves identified in Table T.S. 3.1-2 is contained within "Order for Modification of License" dated April 20, 1981.

Technical Specification 4.2.b

These Technical Specifications provide the inspection and repair/plugging requirements for the steam generator tubes at the Kewaunee Nuclear Power Plant. Fulfilling these specifications will assure the KNPP steam generator tubes are inspected and maintained in a manner consistent with current NRC regulations and guidelines including the General Design Criteria in 10 CFR Part 50, Appendix A.

General Design Criterion (GDC) 14 "Reactor Coolant Pressure Boundary," and GDC 31, "Fracture Prevention of Reactor Coolant Pressure Boundary," require that the reactor coolant pressure boundary have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. Also, GDC 15, "Reactor Coolant System Design," requires that the Reactor Coolant System and associated auxiliary, control, and protection systems be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. Furthermore, GDC 32 "Inspection of Reactor Coolant System Pressure Boundary," requires that components that are part of the reactor coolant pressure boundary be designed to permit periodic inspection and testing of critical areas to assess their structural and leak tight integrity.

The NRC has developed guidance for steam generator tube inspections and maintenance including Regulatory Guides 1.83 and 1.121. Regulatory Guide 1.83, "In-service Inspection of Pressurized Water Reactor Steam Generator Tubes," forms the basis for many of the requirements in this section and should be consulted prior to any revisions. Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes," defines the minimum wall thickness in a steam generator tube, and may be applied to tube sleeves in determining their minimum wall thickness.

Technical Specification 4.2.b.1

If the steam generators are shown to be performing in a like manner, it is appropriate to limit the inspection to one steam generator on a rotating schedule. Economic savings as well as reductions in personnel exposure and outage duration can be realized.

Technical Specification 4.2.b.2

Periodic inspection of the steam generator tubes allows evaluation of their service condition. As operational experience has become available it is evident that certain types of steam generators are susceptible to generic degradation mechanisms. Site specific steam generator tube degradation has also occurred throughout the industry. The inspection program at Kewaunee is designed to identify both generic and site specific tube degradation mechanisms.

Steam generator tube surveillance at Kewaunee is generally performed using eddy current techniques. Various methods of eddy current (EC) testing are used to inspect steam generator tubes for wall degradation. EC methods have improved considerably since Kewaunee began commercial operation in 1974. Single frequency EC testing with a single probe and X-Y plotter have evolved into multifrequency techniques with assorted probe types and sophisticated software to allow more accurate volumetric tube examinations. Profilometery techniques are also being developed which detect imperfections in a tube's original geometry. WPSC is committed to utilize advancing EC testing technology, as appropriate, to assure accurate determination of the steam generator tubes' service condition.

Technical Specification 4.2.b.3

Steam generator tube inspections are generally scheduled during refueling outages at the Kewaunee Nuclear Power Plant. The tubes scheduled for a given inspection are based upon their service condition determined during previous inspections, and operational experience from other plants with similar steam generators and water chemistry. Identification of degraded steam generator tube conditions results in augmentation of the inspection effort as well as increasing the frequency of subsequent inspections. In this manner, steam generator tube surveillance is consistent with service conditions.

There are several operational occurrences or transients that will require subsequent steam generator tube inspections. These inspections are required as a result of excessive primary-to-secondary leakage or transients imposing large mechanical and thermal stresses on the tubes.

Technical Specification 4.2.b.4

Steam generator tubes[4] found with less than the minimum wall thickness criteria determined by analysis, as described in WCAP 7832[5][6], must either be repaired to be kept in service or removed from service by plugging.

Steam generator tube plugging is a common method of preventing primary-to-secondary steam generator tube leakage and has been utilized since the inception of PWR nuclear reactor plants. This method is relatively uncomplicated from a structural/mechanical standpoint as flow is cut off from the affected tube by plugging it in the hot and cold leg faces of the tubesheet.

To determine the basis for the sleeve plugging limit, the minimum sleeve wall thickness was calculated in accordance with Draft Regulatory Guide 1.121 (August 1976). In addition, a combined allowance of 20% of wall thickness is assumed for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

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[4] For the 1991-1992 operating cycle only, Specification 4.2.b.4 applies to the tube excluding the hot leg tubesheet crevice region. Refer to Specification 4.2.b.5 for the tubesheet crevice region criteria.

[5] WCAP 7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions."

[6] E. W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

Repair by sleeving, or other methods, has been recognized as a viable alternative for isolating unacceptable tube degradation and preventing tube leakage. Sleeving isolates unacceptable degradation and extends the service life of the tube, and the steam generator. Tube repair, by sleeving in accordance with WCAP 11643[7] has been evaluated and analyzed as acceptable. This WCAP establishes hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

Technical Specification 4.2.b.5

The purpose of Specification 4.2.b.5 is to clarify the repair criteria for ambiguous eddy current indications in the hot leg tubesheet crevice region and is applicable for the 1991-1992 operating cycle only. During the spring 1990 refueling outage, eddy current inspections using a rotating pancake coil found axial indications in the tubesheet crevice region which were not discernible using the standard bobbin coil eddy current probe. A metallurgical exam of two tubes pulled from the Steam Generator B hot leg revealed the presence of axial cracks within the tubesheet crevice area which could not be reliably detected and sized with the standard bobbin coil technology.

An evaluation of tube integrity and associated radiological consequences was performed to show that continued operation of the plant with these indications in service provided an adequate margin of safety. This evaluation was based on the pulled tube exam and leakage rate testing of crevices restricted by top-of-tubesheet dents analogous to those present in the Kewaunee steam generators. The details of this evaluation are documented in WCAP 12790[8].

The results of this evaluation conservatively demonstrate that with an operating leak rate limit of 200 gallons per day (administrative limit imposed for the 1991-1992 operating cycle), a total of 388 tubes per steam generator with through wall cracks in the tubesheet crevice region can be in service without exceeding 10% of the 10 CFR Part 100 guidelines during a postulated steam line break.

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[7]WCAP 11643, Kewaunee Steam Generator Sleeving Report, Revision 1, November 1988 (Proprietary).

[8]WCAP 12790 "Kewaunee Steam Generator Mid-Cycle Report," December 1990 (Proprietary).

During the 1991 refueling outage, 27-inch sleeves will be installed in addition to the 30- or 36-inch sleeves which were used in previous outages. The 27-inch sleeves expand the current sleeving boundary to cover approximately 84% of the tube bundle. During the 1992 refueling outage, flexible sleeving technology may be used which will extend the sleeving boundary to all but the outermost tubes. Therefore, this specification is an interim measure for the 1991-1992 operating cycle until the sleeving boundary is extended.

Technical Specification 4.2.b.6

Category C-3 inspection results are considered abnormal degradation to a principal safety barrier and are therefore reportable under 10 CFR 50.72(b)(2)(i) and 10 CFR 50.73(a)(2)(ii).

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 93 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER AND LIGHT COMPANY
MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

On March 19, 1991 Wisconsin Public Service Corporation (WPSC), the licensee, requested exigent license amendment approval for a revision to Technical Specification (TS) Section 4.2. The exigent nature of the licensee's request was based principally on the proximity to and the need for the TS change prior to plant restart from the current refueling outage which is scheduled to be completed by May 4, 1991. The staff, in its initial screening of the proposed amendment, made the decision to process the licensee's request with a normal 30-day notice period. The staff's decision to process this request within the normal time frame was based on providing the most appropriate public notice period.

The amendment provides clarification on how motorized rotating pancake coil (MRPC) eddy current indications in the steam generator (SG) hot leg tubesheet crevice area will be dispositioned during the licensee's spring 1991 refueling outage. Other changes to the TS include administrative changes to the current TS Section 4.2 to the Word Perfect format and to correct minor typographical and format inconsistencies.

After discussions with the staff, WPSC, on April 26, 1991, supplemented their original application to address staff concerns regarding the dispositioning of circumferential indications in the SG crevice region. The supplements to the technical specifications (TSs) do not change the intent of the original TSs, and therefore need not be renoticed. To date, no SG tubes with circumferential indications have been left in service in either SG.

2.0 DISCUSSION

The Kewaunee Nuclear Power Plant (KNPP) steam generators are Westinghouse Model 51. Each SG contains 3388 tubes constructed of Mill Annealed Alloy 600 (MA600). The tubes are partial depth rolled to a depth of approximately

2.5 inches, leaving a crevice depth of approximately 18 inches in the tube sheet. The tube support plates and the tube sheets are made of carbon steel.

The SGs were operated for about 6 months during the initial fuel cycle with Phosphate Chemistry and have since been operated with All Volatile Treatment (AVT) Chemistry. During the past two operating cycles, boric acid has been added to the secondary side of the SGs to act as a buffer to the caustic conditions in the SG crevice and tube support plate regions.

Augmentation of the bobbin coil SG eddy current inspections by MRPC inspections began in 1987. During the 1990 refueling outage, the licensee supplemented the bobbin coil inspections in the SG hot leg tubesheet, hot leg and cold leg tube support plates, and low row (Rows 1 and 2) U-bend areas. During a post test review of the MRPC results for the 1B SG, the licensee discovered a small distorted indication (DI) in the hot leg crevice region with significant axial orientation. The MRPC DI had not been previously detected by the bobbin coil eddy current inspection. This finding raised the possibility that the bobbin coil exam may not have identified all the axial indications present in the SG hot leg crevice region.

The licensee chose to increase the sample of tubes with no bobbin indications in the hot leg from the worst area of degradation in both SGs for MRPC inspection. The sample expansion resulted in KNPP examining more than 300 tubes by MRPC in the tubesheet crevice region. After completing the data review, nearly 70 of these tubes revealed the presence of axially oriented indications not previously identified with the bobbin coil. The indications were generally located 5 inches or more below the top of the tubesheet.

Prior to plant restart from the 1990 refueling outage, KNPP submitted its return to power report. The KNPP return to power report, dated April 6, 1990, identified a number of commitments made by WPSC. The commitments included an administrative primary-to-secondary leakage limit of 200 gallons per day (GPD), an examination of two tubes pulled from the 1B SG, and submittal of the pulled tube evaluation to the NRC staff for review.

The KNPP updated return to power report, dated December 27, 1990, contains the results of the pulled tube examination and leakage rate tests performed on crevices restricted by top of tubesheet dents analogous to those present in the KNPP SGs. The examination of the two tubes pulled from the 1B SG revealed the presence of alkaline induced stress corrosion cracking (SCC) with minor intergranular attack (IGA). The leakage and crevice evaluations indicate that the KNPP can operate with a total of 388 unrepaired tubes with crevice indications.

During the 1991 refueling outage, KNPP will use 27-inch sleeves in addition to the 30- or 36-inch sleeves that were installed in previous outages. The 27-inch sleeve will expand the current sleeving boundary to cover approximately 84% of the tube bundle. During the 1992 refueling outage, KNPP plans to utilize flexible sleeving technology to extend the sleeving boundary to all but the outermost rows of tubes. Therefore, the proposed TS change is only an interim measure for the 1991-1992 operating cycle.

The licensee has committed to continue the administrative primary-to-secondary leakage limit of 200 GPD and continue the enhanced leakage monitoring program for NRC Bulletin 88-02, "Rapid Propagating Fatigue Cracks in Steam Generator Tubes."

3.0 EVALUATION

3.1 General

The two accidents addressed in the KNPP USAR Chapter 14 that consider primary-to-secondary leakage are the SG tube rupture (SGTR) and the steam line break (SLB). Of these, the licensee has indicated that the SLB is the most limiting with regard to leakage due to the differential pressure across the SG tubes.

The licensee's pulled tube examination confirmed the presence of axial cracks in the area of the tubesheet crevice region. The evaluation, WCAP-12790 "Kewaunee Steam Generator Mid-Cycle Report," December 1990 (Proprietary), concluded that alkaline SCC was responsible for the observed corrosion degradation on the two pulled tubes. Neither tube exhibited Primary Water Stress Corrosion Cracking (PWSCC) on the inner diameter surface of the mechanical expansion transition.

3.2 Tube Rupture

The licensee has indicated that the presence of a small annular gap around the tube within the crevice region would limit the tube expansion and thus preclude a tube rupture. Therefore it follows that primary-to-secondary leakage would be characterized by a leakage path which would be restricted by the tube/tubesheet crevice and tube denting rather than a rupture characterization should a through-wall crack occur.

3.3 Radiological Consequences

Although tubes with indications of cracking within the tubesheet crevice region are not subject to burst, it certainly cannot be assured that the cracks will not leak as a result of an SLB due to the large, sudden differential pressure which is developed during the initial blowdown portion of the accident. The licensee determined the acceptable number of tubes with known and projected MRPC indications, assuming all develop through-wall cracks in the crevice during an SLB, by dividing the total permissible leak rate during an SLB by the maximum possible leak rate for one tube with a worst-case projected MRPC indication.

The licensee calculated the total permissible leak rate, 260 gallons per minute (GPM), based on not exceeding 30 rem to the thyroid at the site boundary;

30 rem to the thyroid is 10 percent of the 10 CFR Part 100 limit. The assumptions used in the analysis include 1 percent fuel defects deposited into the SG for the faulted loop, an iodine plate-out factor of 0.1, and an iodine decontamination factor of 1.0.

The staff performed an independent radiological consequences analysis using: (1) 260 gpm primary-to-secondary leak rate during an SLB accident; (2) reactor coolant specific fission product activities shown in Table D.4-1 of the Kewaunee USAR; (3) the maximum primary coolant activity limit specified in Section 3.1.c of the Kewaunee Technical Specifications; and (4) a 30-minute operator action time to identify and isolate the faulty steam generator. The staff neither used an iodine plate-out factor nor an iodine decontamination factor during an SLB accident. The staff's calculated doses at the site boundary are 0.38 rem to the whole body and 14.6 rem to the thyroid. These dose values are less than 10 percent of dose reference values specified in 10 CFR Part 100 and within the acceptance criteria specified in the Standard Review Plan Section 15.1.5. Therefore, the staff finds that the total permissible primary-to-secondary leak rate of 260 gpm during an SLB accident is acceptable.

A computer code called CRACKFLO has been developed for predicting leak rates through cracked SG tubes. The analytical model assumes one-dimensional flow and accounts for crack entrance pressure losses, tube wall friction, and flashing. The model also considers the fact that all unsleeved tubes at Kewaunee have been found to be dented at the top of the tubesheet. To simplify the study of leakage through a crack in series with denting corrosion, an equivalent flow channel of constant diameter is assumed by the licensee's analysis. This means that the added hydraulic resistance of the denting corrosion can be characterized by a resistance coefficient, KD. The licensee determined this coefficient by test and correlated it with past plant operating history. The empirical KD value chosen for the leak analysis bounds the KNPP operationally determined KD for past operating history.

The results of the leak rate assessment indicate that, with the presence of denting at the top of the tubesheet, up to 1250 unsleeved tubes with long (greater than 1 inch) axial cracks could leak without exceeding the calculated steam line break (SLB) leak rate limit of 260 gpm discussed above. Based on the leakage test data for a "non-dented" configuration, up to 388 tubes could leak without exceeding the SLB leak rate limit of 260 gpm. Under the proposed Technical Specification change, the plugging criteria for the tubesheet crevice indications is conservatively applicable to no more than 388 tubes. In a phone conversation with the staff on April 18, 1991, the licensee indicated that, after applying the proposed criteria to SG tubes with eddy current indications in the crevice region, approximately 12 tubes will be left in service. This is a considerably smaller number of tubes than are allowed to be left in service.

Based on the above, the staff concludes that any leakage during an SLB during the upcoming cycle is conservatively bounded by the 260 gpm allowable SLB leakage.

3.4 Proposed Technical Specifications

Pages TS ii and TS iii

The TS Table of Contents pages are administratively modified to include the TS Section 4.2 changes. Since these changes are purely administrative in nature, the staff finds them acceptable.

Pages TS 4.2-1, 4.2-2, 4.2-2.A, 4.2-3 and 4.2-4

The TS changes include: (1) adding a hyphen in TS 4.2.a.1 and 2 for the word "In-service;" (2) adding a comma on the fourth line in TS 4.2.a.1.; and (3) moving type, including a footnote from old TS page 4.2-2, to new TS page 4.2-1. Page TS 4.2-2.A did not have any text on it and was therefore removed from the TSs. TS 4.2.b. was moved from TS page 4.2-3 and now begins on TS page 4.2-2. The above TS changes are all administrative in nature. Therefore, the staff finds them acceptable.

TS 4.2.b. has several administrative changes and clarifications. The three term definitions added to TS 4.2.b. are for Distorted Indication (DI), Tubesheet Crevice Region, and Squirrel Indications. The three added definitions are all consistent with current industry practices and the licensee's safety evaluation, therefore, they are acceptable to the staff. The administrative changes to TS 4.2.b. include: (1) replacing a hyphen with the word "is" and capitalizing the words "Reactor Coolant System" in the definition of "Tube," and (2) capitalizing the words "Technical Specifications" in the definition of "Repaired Tube." These changes are administrative in nature and, therefore, are acceptable to the staff.

TS 4.2.b.2 has been moved from old TS page 4.2-4 and 4.2-5, and now begins on new TS page 4.2-3 and ends on TS page 4.2-4. Administrative changes to TS 4.2.b.1. and 2. include: (1) heading format changes, (2) TS 4.2.b.2.c. was modified to make the word "tube" plural, (3) TS 4.2.b.2.d. was modified to de-hyphenate the word "tubesheet" and hyphenate the word "in-service," and (4) a comma was added to the C-1 category Inspection Results column after the word "tubes." Since these changes are all administrative in nature, they are all acceptable to the staff.

TS 4.2.b.3. was shifted from old TS pages 4.2-5 and 4.2-6 to TS pages 4.2-4 and 4.2-5. Administrative changes to TS 4.2.b.3. include: (1) a heading format change; (2) hyphenation of the words "in-service," "40-month" and "pre-service"; (3) a semicolon was added after the word "category" in TS 4.2.b.3.a; (4) the words "Rector Coolant System" were capitalized in TS 4.2.b.3.c.3. and 4.; (5) the word "or" was added to the end of TS 4.2.b.3.c.1., 2. and 3. for clarification purposes; and (6) "(3)" was removed from TS 4.2.b.3.d. Since all these modifications are administrative in nature, the staff finds them acceptable.

TS 4.2.b.4 has been moved from old TS page 4.2-6 to new TS page 4.2-5. Administrative changes to this TS include a change in the heading format and removal of two commas in TS 4.2.b.4.a. A footnote has been added to this TS indicating that TS 4.2.b.4 as it applies to unsleeved tubes is not applicable within the hot leg tubesheet crevice region. The footnote refers to TS 4.2.b.5 for indications within the hot leg tubesheet crevice region. Since the above changes are all administrative in nature and the following evaluation accepts TS 4.2.b.5, the staff finds these changes acceptable.

TS 4.2.b.5, "Hot Leg Tubesheet Crevice Plugging Limit Criteria," has been added and old TS 4.2.b.5 is renumbered as TS 4.2.b.6. The new TS provides clarification to the dispositioning method of indications discovered within the hot leg tubesheet crevice region for the 1991-1992 operating cycle. TS 4.2.b.5 is split into two parts: (1) 4.2.b.5.a indicates criteria for the repair or plugging of tubes examined by the bobbin and/or the MRPC, and (2) 4.2.b.5.b. specifies that no more than 388 tubes per SG with crevice indications be left in service. Also, a footnote has been added, specifying that the proposed TS 4.2.b.5 is applicable only for the 1991-1992 operating cycle.

TS 4.2.b.5.a.1. specifies that any tube bobbin coil exam which exhibits tube wall degradation of 50 percent or more shall be repaired or plugged. This is a standard practice to the industry and is acceptable to the staff. TS 4.2.b.5.a.2. specifies that any tube with an indication identified as a multiple circumferential or single circumferential indication shall be plugged or repaired prior to returning the steam generator to service. TS 4.2.b.5.a.3 specifies that any tube with an indication identified as a multiple axial indication (MAI), or single axial indication (SAI), with the MRPC exam and is repairable by sleeving shall be repaired or plugged. The term "repairable by sleeving" indicates that those tubes for which sleeving is the appropriate repair method (for example, the tubes within the 27-inch sleeving boundary) should be repaired. Tubes within the 27-inch sleeving boundary which display MAI or SAI within the partial depth roll by MRPC and have no distorted indication by bobbin, may be left in service but are subject to specification TS 4.2.b.5.b. This clarification to the TS is acceptable to the staff and is more conservative than previous TS for the KNPP.

TS 4.2.b.5.a.4. specifies that any tube with an indication identified as an MAI or SAI with MRPC exam and the corresponding bobbin evaluation indicated either a distorted roll indication, distorted crevice indication or squirrel shall be plugged or repaired. This is a standard industry practice and is acceptable to the staff. TS 4.2.b.5.a further clarifies dispositioning of tubes which exhibit significant general tube thinning by applying a 40 percent through-wall criterion to specification 4.2.b.5.a.1.

Again, specification 4.2.b.5.b. limits the number of tubes which exhibit a crevice indication to a total of 388 tubes per steam generator. Based on the discussions above, the staff finds the licensee's proposed TS change acceptable.

As stated above, old TS 4.2.b.5, "Reports", has been renumbered to TS 4.2.b.6 and is now located on TS pages 4.2-6 and 4.2-7. Administrative changes to the specification include: (1) hyphenating the word "in-service, " (2) adding spaces to correct previous problems, and (3) changing a reference to TS 4.2.b.6.a. from old TS 4.2.b.5.a.

The KNPP TS 4.2 basis section now begins on TS page 4.2-7 vice TS page 4.2-8 and still ends on TS page 4.2-11. Administrative changes to the TS 4.2 basis include: (1) system capitalization changes, (2) spacing changes, (3) hyphenation changes, (4) heading format and numbering changes to correspond to the TS changes described above, and (5) placement of references as footnotes vice listing at the end of TS 4.2. The above mentioned changes are all administrative in nature and, therefore, are acceptable to the staff.

Basis section Technical Specification 4.2.b.5 was added to edify the reasoning for and provide the historical background behind TS 4.2.b.5. It also references WCAP-12790, "Kewaunee Steam Generator Mid-Cycle Report," December 1990 (Proprietary) as the evaluation for tube integrity. Since the TS 4.2 basis changes provide clarification only and reflect the discussions above, the staff finds them acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). This amendment also involves changes in recordkeeping, reporting or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR §51.22(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 this amendment.

6.0 CONCLUSION

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

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