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Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

SUBJECT: AMENDMENT NO. 127 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M95713)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 127 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. This amendment revises the Technical Specifications (TS) in response to your application dated June 10, 1996 as supplemented on August 27, and September 5, 1996.

The amendment revises TS 4.2.b, "Steam Generator Tubes," and its associated basis, by allowing the use of Westinghouse laser-welded sleeves to repair defective steam generator tubes.

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:

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Amendment No. 127 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME: G:\KEWAUNEE\KEW95713.AMD

* see previous concurrence

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

September 24, 1996

Mr. M. L. Marchi
Manager - Nuclear Business Group
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, WI 54307-9002

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Office of Nuclear Reactor Regulation

Docket No. 50-305

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cc w/encls: See next page

cc:

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

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. 127
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 June 10, 1996, as supplemented on August 27, and September 5, 1996, 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. 127 , 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

Richard J. Laufer

Richard J. Laufer, Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: September 24, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 127

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 vertical lines indicating the area of change.

REMOVE	INSERT
TS v	TS v
TS 4.2-3	TS 4.2-3
TS 4.2-4	TS 4.2-4
TS 4.2-5	TS 4.2-5
TS 4.2-6	TS 4.2-6
TS 4.2-7	TS 4.2-7
TS 4.2-8	TS 4.2-8
TS B4.2-3	TS B4.2-3
TS B4.2-4	TS B4.2-4
TS B4.2-5	TS B4.2-5
---------	TS B4.2-6
Table TS 4.2-2	Table TS 4.2-2
---------	Table TS 4.2-3

LIST OF TABLES

TABLE	TITLE
1.0-1 . . .	Frequency Notations
3.1-1 . . .	WPS (136) Reactor Vessel Toughness Data
3.1-2 . . .	Reactor Coolant System Pressure Isolation Valves
3.5-1 . . .	Engineered Safety Features Initiation Instrument Setting Limits
3.5-2 . . .	Instrument Operation Conditions for Reactor Trip
3.5-3 . . .	Emergency Cooling
3.5-4 . . .	Instrument Operating Conditions for Isolation Functions
3.5-5 . . .	Instrument Operation Conditions for Safeguards Bus Power Supply Functions
3.5-6 . . .	Accident Monitoring Instrumentation Operating Conditions for Indication
4.1-1 . . .	Minimum Frequencies for Checks, Calibrations and Test of Instrument Channels
4.1-2 . . .	Minimum Frequencies for Sampling Tests
4.1-3 . . .	Minimum Frequencies for Equipment Tests
4.2-1 . . .	Deleted
4.2-2 . . .	Steam Generator Non-Repaired Tube Inspection
4.2-3 . . .	Steam Generator Repaired Tube Inspection

1. <u>Steam Generator Sample Selection and Inspection</u>

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. <u>Steam Generator Tube Sample Selection and Inspection</u>

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 20% 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 below and in TS 4.2.b.2.b.

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

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.

   Implementation of the steam generator tube/tube support plate repair criteria requires a 100% 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% random sampling of tubes inspected over their full length.

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. For non-repaired tubes, actions shall be taken as described in Table 4.2-2. For repaired tubes, actions shall be taken as described in Table 4.2-3.

Category	Inspection Results
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. Inspection Frequencies

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 TS 3.1.d and TS 3.4.a.1.C 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. Plugging Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied.

a. Any tube 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. Tube repair shall be in accordance with the methods described in WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)," CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves," or WCAP-13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report."

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

c. Any Combustion Engineering leak tight sleeve which, upon inspection, exhibits wall degradation of 40% or more shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld region.

d. Any Westinghouse laser welded sleeve which, upon inspection, exhibits wall degradation of 25% or more, shall be plugged prior to returning the steam generator to service. This plugging limit applies to the sleeve up to and including the weld.

5. Tube Support Plate Plugging Limit

The following criteria are 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 intersection, the repair limit is based on maintaining steam generator tube serviceability as described below:

a. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltage $\leq$ 2.0 volts will be allowed to remain in service.

b. Degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts will be repaired or plugged except as noted in TS 4.2.b.5.c below.

c. Indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage > 2.0 volts but $\leq$ the upper voltage repair limit, may remain in service if a rotating pancake coil inspection does not detect degradation. Indications of outside diameter stress corrosion cracking degradation with a bobbin voltage > the upper voltage repair limit will be plugged or repaired.

d. If an unscheduled mid-cycle inspection is performed, the following repair limits apply instead of TS 4.2.b.5.a, b and c. The mid-cycle repair limits are determined from the following equation:

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

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

Where:

V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle
V_{SL} = structural limit voltage
NDE = 95% cumulative probability allowance for NDE uncertainty
Gr = average growth rate per cycle length
CL = cycle length (time between scheduled inspections)
Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
V_{URL} = upper voltage repair limit

Implementation of these mid-cycle repair limits should follow the same approach as in TS 4.2.b.5.a, b and c.

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

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.

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).

d. 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:

   1. 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.

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

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

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

   5. 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.

Technical Specification 4.2.b.4

Steam generator tubes found with less than the minimum wall thickness criteria determined by analysis, as described in WCAP-7832[1][2],, 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 the ASME Code and is consistent with Draft Regulatory Guide 1.121 (August 1976).

For the Westinghouse mechanical sleeves, the sleeve plugging limit of 31% is applied to the sleeve as shown on Figure TS 4.2-1. For the Combustion Engineering leak tight sleeves, a plugging limit of 40% is applied to the sleeve and weld region. The sleeve plugging limits allow for eddy current testing inaccuracies and continued operational degradation per Draft Regulatory Guide 1.121 (August 1976).

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[3], CEN-413-P[4], and WCAP-13088[5], has been evaluated and analyzed as acceptable. The Westinghouse mechanical sleeve spans the degraded area of the parent tube in the tubesheet region. The sleeves are either 36", 30" or 27" to allow access permitted by channel head bowl geometry. The sleeve is hydraulically expanded and hard rolled into the parent tubing.

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(1) WCAP 7832, "Evaluation of Steam Generator Tube, Tube Sheet, and Divider Plate Under Combined LOCA Plus SSE Conditions."

(2) E. W. James, WPSC, to A. Schwencer, NRC, dated September 6, 1977.

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

(4) CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves," January 1992 (Proprietary).

(5) WCAP 13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report," January 1994.

There are three types of Combustion Engineering leak tight sleeves. The first type, the straight tubesheet sleeve, spans the degraded area of the parent tube in the tubesheet crevice region. The sleeve is welded to the parent tube near each end. The second type of sleeve is the peripheral tubesheet sleeve. The sleeve is initially curved as part of the manufacturing process and straightened as part of the installation process. The third type of sleeve, the tube support plate sleeve, spans the degraded area of the tube support plate and is installed up to the sixth support plate. This sleeve is welded to the parent tube near each end of the sleeve.

Two types of Westinghouse laser welded sleeves can be installed, tube support plate sleeves and tubesheet sleeves.

The tube support plate sleeve is 12" long and spans the degraded area of the tube adjacent to the support plate intersection. The tube support plate sleeve is hydraulically expanded and laser welded at each end. The pressure boundary portion of the tube support plate sleeve is the weld and the sleeve section between the welds. Tubesheet sleeves extend from the tube end to above the top of the tubesheet. Standard and bowed or peripheral tubesheet sleeves can be installed. The upper or free span joint is hydraulically expanded and laser welded. The lower joint is hydraulically expanded and roll expanded. Standard tubesheet sleeves extend from 27" to 36" in length while bowed tubesheet sleeves extend from 30" to 36" in length. The pressure boundary portion of the tubesheet sleeve is the weld and below, down to the tubesheet primary face.

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. 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 repair limit of tubes with degradation attributable to outside diameter stress corrosion cracking contained within the thickness of the tube support plates is conservatively based on the analysis documented in WCAP-12985, "Kewaunee Steam Generator Tube Plugging Criteria for ODSCC at Tube Support Plates" and EPRI Draft Report TR-100407, Rev.1, "PWR Steam Generator Tube Repair Limits - Technical Support Document for Outside Diameter Stress Corrosion Cracking at Tube Support Plates." Application of these criteria is based on limiting primary-to-secondary leakage during a steam line break to ensure the applicable 10 CFR Part 100 limits are not exceeded.

The voltage-based repair limits of TS 4.2.b.5 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 tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and 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 GL 95-05.

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

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).

TS 4.2.b.6.d implements several reporting requirements recommended by GL 95-05 for situations which NRC wants to be notified prior to returning the steam generators to service. For TS 4.2.b.6.d.3 and 4, indications are applicable only where alternate plugging criteria is being applied. 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 GL 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 GL Sections 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per GL Section 6.b(c) criteria.

TABLE TS 4.2-2

STEAM GENERATOR NON-REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION		3RD SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required	Result	Action Required
A minimum of S Tubes per S.G.	C-1	None	N/A	N/A	N/A	N/A
	C-2	Plug or repair defective tubes and inspect additional 2S tubes in this S.G. (2)	C-1	None	N/A	N/A
			C-2	Plug or repair defective tubes and inspect additional 4S tubes in this S.G. (2)	C-1	None
					C-2	Plug or repair defective tubes
					C-3	Perform action for C-3 result of first sample
			C-3	Perform action for C-3 result of first sample	N/A	N/A
	C-3	Inspect all tubes in this S.G., (2) plug or repair defective tubes and inspect 2S tubes in the other S.G. (2) Prompt notification of the Commission. (1)	The other S.G.'s are C-1	None	N/A	N/A
			Some S.G.'s C-2 but no additional S.G. are C-3	Perform action for C-2 result of second sample	N/A	N/A
			Additional S.G. is C-3	Inspect all tubes in each S.G. and plug or repair defective tubes. Prompt notification of the Commission. (1) (2)	N/A	N/A

S=6%/n Where n is the number of steam generators inspected during an inspection.

Notes: 1. Refer to Specification 4.2(b)(6)(c)

2. As allowed by TS 4.2.b.2.d, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.

TABLE TS 4.2-3

STEAM GENERATOR REPAIRED TUBE INSPECTION

1ST SAMPLE INSPECTION			2ND SAMPLE INSPECTION	
Sample Size	Result	Action Required	Result	Action Required
A minimum of 20% of Repaired Tubes (1)	C-1	None	N/A	N/A
	C-2	Plug defective repaired tubes and inspect all remaining repaired tubes in this S.G.	C-1	None
			C-2	Plug defective repaired tubes
			C-3	Perform action for C-3 result of first sample
	C-3	Inspect all repaired tubes in this S.G., plug defective tubes and inspect 20% of the repaired tubes in the other S.G. Notification to NRC pursuant to 50.72(b)(2)(i) of 10 CFR Part 50	The other S.G. is C-1	None
			The other S.G. is C-2	Perform action for C-2 result of first sample
			The other S.G. is C-3	Inspect all repaired tubes in each S.G. and plug defective tubes. Notification to NRC pursuant to 50.72(b)(2)(i) of 10 CFR Part 50

(1) Each repair method is considered a separate population for determination of scope expansion.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 127 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

By letter dated June 10, 1996, as supplemented on August 27, and September 5, 1996, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TS). The proposed amendment would revise KNPP TS Section 4.2.b, "Steam Generator Tubes," and its associated basis, by allowing the use of Westinghouse (W) laser-welded sleeves to repair defective steam generator (SG) tubes.

The proposal addressed the use of two types of laser welded sleeves; a tubesheet sleeve and a tube support plate sleeve. Both types of sleeves are first secured by hydraulically expanding the upper and lower portions of the sleeve. The hydraulic expansion brings the sleeve ends into contact with the parent tube in preparation for subsequent welding or rolling. The tubesheet sleeve is installed by means of two different joint types: a rolled joint (mechanically expanded) in the tubesheet end and an autogenous laser weld at the freespan end. The tube support sleeves are laser welded to the SG tube in the freespan near each end of the sleeve. The material of construction for the sleeves is a nickel-iron-chromium alloy, alloy 690, a Code approved material (ASME SB-163), incorporated in ASME Code Case N-20.

Extensive analysis and testing were performed on the W sleeves and sleeve/tube joints to demonstrate that Regulatory and Code design criteria were satisfied under normal operating and postulated accident conditions. The details of the sleeve qualifications are discussed in report WCAP-13088, Revision 3, "Laser Welded Sleeves," Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report, dated January 1994 (proprietary). This generic sleeving report presents the technical bases supporting the licensing of laser welded sleeves for use in 7/8 inch diameter SG tubes such as those at Kewaunee.

The staff previously reviewed identical and closely similar W documents supporting requests for changes to the TS at other plants. The bulk of the technical and regulatory issues for the present request are identical to those reviewed in previous Safety Evaluations (SEs) concerning the use of W laser welded sleeves. This SE discusses only those issues warranting revision, amplification, or inclusion based upon current experience. A summary of the principal technical issues regarding the design and use of W laser welded sleeves follows. Details of the prior staff evaluation of W sleeves may be found in SEs for Calvert Cliffs Nuclear Power Plant Units 1 and 2, Docket Nos. 50-317 and 50-318, dated March 22, 1996; DC Cook Nuclear Power Plant Unit 1, Docket No. 50-315, dated January 4, 1996; Maine Yankee Nuclear Power Plant, Docket No. 50-309, dated May 22, 1995; and Joseph M. Farley Nuclear Power Plant, Units 1 and 2, Docket Nos. 50-348 and 50-364, dated October 22, 1990. These evaluations apply to the proposed Kewaunee license amendment.

The August 27, 1996, submittal increased the TS required sample size for in-service inspection of repaired tubes in both SGs. The September 5, 1996, submittal incorporated the EPRI guidelines for SG inspection scope expansion for repaired SG tubes into the TS. These submittals clarified the initial request and did not change the initial proposed no significant hazards consideration determination published in the July 3, 1996, Federal Register.

2.0 BACKGROUND

A sleeve is a tube slightly smaller in diameter than a SG tube that is inserted into a SG tube to bridge a degraded or susceptible section of tube. The length of a sleeve is selected according to the individual installation circumstance. Generally, they vary in length between one and three feet. The sleeve becomes the pressure boundary and thereby restores the structural integrity of a degraded or potentially degraded portion of the original SG tube.

Prior to the development of sleeve technology, a defective SG tube was removed from service by plugging. However, this reduces the heat transfer area. The reduction in heat transfer (or other thermal-hydraulic operating parameters) can be tolerated up to a point before other system consequences of the reduced SG performance become limiting. Beyond this limit, a utility has to make operational changes resulting in reduced electrical generating capacity of the affected unit.

Because sleeves have minimal effect upon the thermal-hydraulics of a SG, their use is essentially unrestricted. This means a licensee may restore degraded sections of SG tubes to like new condition without experiencing a serious penalty with regard to unit generating capacity. This has led to increased use of sleeves versus plugs where practical. Recently, some foreign and domestic plants have installed sleeves in previously unprecedented numbers, up to nearly 100 percent of the SG tubes on a single unit.

About 29,000 W laser welded sleeves have been installed in foreign and domestic plants since 1988. Over 8 years of operating experience with W sleeves has shown the technology to be highly reliable. No operationally induced degradation or leakage has occurred in any W laser welded sleeves.

3.0 SUMMARY OF PREVIOUS REVIEWS

Previous staff evaluations of W sleeves addressed the technical adequacy of the sleeves in the principal areas of pressure retaining component design: structural requirements, material of construction, welding, nondestructive examination and sleeve plugging limit. The staff found the analyses and tests submitted to address these areas of component design and inspection, as summarized below, to be acceptable.

3.1 Structural Requirements

The function of sleeves is to restore the structural integrity of the tube pressure boundary. Consequently, structural analyses were performed for a variety of loadings including design pressure, operating transients, and other parameters selected to envelope loads imposed during normal operating, upset, and accident conditions. Stress analyses of sleeved tube assemblies were performed in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III. These analyses, along with the results of qualification testing and previous plant operating experience, were cited to demonstrate the sleeved tube assembly is capable of restoring steam generator tube structural integrity.

3.2 Material of Construction

The sleeves are fabricated from thermally treated alloy 690, a Code approved material (ASME SB-163) covered by ASME Code Case N-20. The staff found the use of alloy 690 is an improvement over the alloy 600 material used in the original SG tubing. Corrosion tests conducted under Electric Power Research Institute (EPRI) sponsorship confirmed test results regarding the improved corrosion resistance of alloy 690 over that of alloy 600. Accelerated stress corrosion tests in caustic and aqueous chloride solutions also indicated alloy 690 resists general corrosion in aggressive environments. Isothermal tests in high purity water have shown that, at normal stress levels, alloy 690 has high resistance to intergranular stress corrosion cracking (IGSCC) in extended high temperature exposure. The NRC concluded, as a result of these laboratory corrosion tests, that alloy 690 is acceptable for use in nuclear power plants. The NRC endorsed the use of Code Case N-20 in Regulatory Guide 1.85, "Materials Code Case Acceptability, ASME Section III, Division 1." The NRC staff has approved use of alloy 690 tubing in replacement steam generators as well as sleeving applications.

3.3 Welding and Post Weld Heat Treatment

Automatic autogenous laser welding is employed to join the sleeve to the parent tube in the freespan regions. The application of this process to the W sleeve design was specifically qualified and demonstrated during laboratory tests employing full scale sleeve/tube mockups. Qualification of the welding procedures and welding equipment operators was performed in accordance with the requirements of the ASME Code, Section IX.

Accelerated corrosion tests have confirmed that a post weld heat treatment (PWHT) significantly improves the IGSCC resistance of the alloy 600 parent tube material in the weld zone. A PWHT reduces the residual stresses resulting from welding. Residual stresses from forming operations (such as bending, welding, etc.) are known to be a principal contributor to IGSCC in alloy 600. Performance of a PWHT greatly reduces the residual stresses from welding thereby enhancing the IGSCC resistance of the alloy 600 portion of the weld zone. (The alloy 690 sleeve material is highly resistant to IGSCC either with or without PWHT.) In its June 10, 1996 submittal, the licensee committed to performing PWHT of the laser welded joint in accordance with the W generic sleeving report.

The rolled joint used to join the sleeve to the tube within the tubesheet effectively isolates the alloy 600 of the parent tube from the environment and thus is not susceptible to IGSCC. Stress relief of these joints is unwarranted.

3.4 Nondestructive Examination

The baseline nondestructive examination of sleeved tubes is conducted using ultrasonic testing (UT) and eddy current testing (ECT). UT is performed after welding to confirm the laser welds are consistent with critical process dimensions and are of acceptable weld quality. W presented data on a UT system demonstrating post weld examinations of the sleeve/tube assembly will be adequate. Standards which included undersized welds were used in the qualification of the UT technique. The results of the qualification tests demonstrate the system can confirm there is a continuous metallurgical bond between the sleeve and tube and that the weld size (width) meets minimum acceptable dimensions.

ECT is then used to establish baseline inspection data for every installed sleeve/tube. In performing the inspection, the licensee will use Electric Power Research Institute (EPRI) "PWR Steam Generator Tube Examination Guidelines" Appendix G qualified personnel and Appendix H qualified ECT techniques. For future sleeve/tube inspections, the licensee committed to following the most current revision of the EPRI guidelines in terms of inspection scope and expansion criteria as well as personnel and technique qualifications. The licensee also modified the TS to incorporate sleeve/tube inspection scope and expansion criteria.

3.5 Sleeve Plugging Limits

The sleeve minimum acceptable wall thickness is determined using the criteria of Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and ASME Code Section III allowable stress values and pressure stress equations. According to RG 1.121 criteria, an allowance for nondestructive evaluation (NDE) uncertainty and postulated operational growth of tube wall degradation within the sleeve must be accounted for when using NDE to determine sleeve plugging limits. Therefore, a conservative tube wall combined allowance for postulated degradation growth and eddy current uncertainty of 20% throughwall per cycle were assumed for the purpose of determining the sleeve plugging limit. The sleeve plugging limit, which was calculated based on the most limiting of normal, upset, or faulted conditions for Series 44 and 51 steam generators, was determined to be 25% of the sleeve nominal wall thickness based on ASME Code minimum material properties in accordance with staff positions. Removal of tubes and/or sleeves from service when degradation reaches the plugging limit provides assurance the minimum acceptable wall thickness will not be violated during the next subsequent cycle of operation.

4.0 DISCUSSION

Experience with all types of SG tube sleeves has led to several areas of concern outside the scope of basic sleeve design and qualification discussed above. These include instances of cracking in sleeved SG tubes, service life predictions for sleeved SG tubes, application of PWHT and the effect of tube lockup, and primary-to-secondary leakage limits.

4.1 Cracking in Sleeved SG Tubes

Recent experiences at two U.S. plants indicated the alloy 600 steam generator tube may be susceptible to IGSCC at the freespan joint of a tubesheet sleeve. The affected joints are of the mechanically expanded type. These employ a hydraulic expansion followed by a hard roll in the center of the hydraulically expanded region. The hard roll forms the structural joint and leak limiting seal. Inner-diameter initiated cracks have been detected in the alloy 600 parent tube material at the lower hard roll transition and lower hydraulic transition of the freespan joints. The cracks were detected after 4 to 7 years of service. Since a number of sleeved tubes with this joint type have operated up to 14 years in one of the affected units with no such degradation, it is clear that not all such sleeved tubes are likely to develop cracks after a given service interval.

Accelerated corrosion tests of laser welded sleeve joints have shown the hydraulic transition to have little or no susceptibility to IGSCC. Service times exceeding 8 years have been achieved for sleeved tubes with laser welded joints at U.S. plants. No instances of service induced IGSCC have occurred in any of these joints. The staff is monitoring these developments for potential impact on welded sleeve installations.

4.2 Service Life Predictions for Sleeved SG Tubes

The staff position on sleeving considers the method unable to assure an unlimited service life for a repaired tube. The conservative view is sleeving creates new locations in the parent tube which may be susceptible to IGSCC after new incubation times are expended. Incubation times are not quantified. They are observed to vary between individual steam generators and the various tubes within, based upon prior experiences with U-bend and roll transition cracking.

This staff position that sleeving has limited service life is due to the circumstances of the sleeving processes. Sleeve installation methods can enhance one or two of the conditions necessary for IGSCC. The primary contributor is the residual stress resulting from the various joining methods. Secondarily, the local environment of the tube may be altered as a result of the formation of a wetted crevice between the tube and sleeve. Remediation of these contributors would benefit sleeved tube life. Of the two, stress relieving may be the most beneficial given the underlying causes of IGSCC and present sleeve designs. As discussed earlier, the sleeve installation procedure includes a PWHT of the weld joints to increase the resistance to IGSCC.

4.3 PWHT and Tube Lockup

Recent field experience with the installation of welded sleeves with PWHT indicated SG tubes may be constrained ("tube lockup") in their tube support plates. The result of such tube locking is distortion of the tube (bowing or bulging) during the PWHT. After the heat treatment is completed, the bow or bulge remains. Measurements of the bowing and bulging have shown them to be of negligible values. These distortions have been analyzed and found to be immaterial to the examination, operation, and safety of the sleeved tubes.

Along with the observed distortion (bowing or bulging) is a residual stress remaining after the heat treatment is completed. Strain gage measurements of this residual stress have shown it to be moderate compared to that resulting from welding without subsequent PWHT. This issue was the subject of additional testing and analysis related to the use of laser welded sleeves at the Maine Yankee facility during a sleeve installation project. Based upon the finding that many tubes are fixed in the tube support plates, W modified their sleeve installation procedure on the assumption that all tubes are locked. The modified installation procedure thereby minimizes the residual stress of PWHT regardless of tube condition.

4.4 Primary-to-Secondary Leakage Limits

While a laser weld should be inherently leak-tight, the lower (rolled) joint of a tubesheet sleeve may not be leak tight. W analyzed the effects of an abnormal lower joint seal. The analysis shows that even under extreme postulated conditions, it will have satisfactory leakage integrity.

With respect to the staff position regarding primary-to-secondary leakage limits, the licensee already implemented a change to its TS adopting a 150 gpd per SG leakage limit.

5.0 CHANGES TO THE TECHNICAL SPECIFICATIONS

The licensee's proposal would revise TS Section 4.2 as follows:

TS 4.2.b.2.a would be revised to change the in-service inspection sample size for repaired SG tubes in both steam generators to 20%.

TS 4.2.b.2.e would be revised to reflect the addition of TS Table 4.2.3 which will be used to classify the inspection results for repaired SG tubes.

TS 4.2.b.4.a would be revised to add sleeving in accordance with WCAP-13088, Revision 3, "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report," as an acceptable method for SG tube repair.

TS 4.2.b.4.d would be added to specify that any Westinghouse laser welded sleeve which upon inspection exhibits wall degradation of 25% or more, shall be plugged prior to returning the SG to service.

Table TS 4.2-2 would be revised to specify its use for non-repaired SG tubes only.

Table TS 4.2-3 would be added to specify the inspection scope expansion criteria for repaired SG tubes.

The TS Bases and the Table of Contents would also be revised consistent with the changes described above.

The staff has reviewed the TS changes discussed above and finds that they consistently incorporate the Westinghouse laser welded sleeving methodology previously discussed in this safety evaluation and will provide adequate assurance of steam generator tube integrity. Therefore, the proposed changes are acceptable.

6.0 SUMMARY

Based on the preceding analysis, the NRC staff concludes the repair of SG tubes using laser welded sleeves designed by W is acceptable, as supplemented by additional licensee commitments, as discussed above, to 1) perform PWHT of the freespan joints, and 2) to use enhanced and improved ECT inspection techniques as they are developed and verified for use.

7.0 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.

8.0 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 changes 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 effluent 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 (61 FR 34902). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). 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.

9.0 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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Date: September 24, 1996