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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. 128 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M95302)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 128 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP). This amendment revises the Technical Specifications (TS) in response to your application dated May 1, 1996, as supplemented on May 31, August 14, August 26 and September 11, 1996. The May 1, 1996, submittal superseded a previous submittal on this subject dated October 6, 1995, as supplemented on November 8, 1995, and January 8 and January 19, 1996.

The amendment revises KNPP TS 4.2.b, "Steam Generator Tubes," its associated bases, and Figure TS 4.2-1 by redefining the pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves.

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. 128 to License No. DPR-43
2. Safety Evaluation

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

September 25, 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

Enclosures: 1. Amendment No. 128 to License No. DPR-43
2. Safety Evaluation

cc w/encls: See next page

cc:

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P. O. Box 1497
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Chairman
Town of Carlton
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Mr. Harold Reckelberg, Chairman
Kewaunee County Board
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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. 128
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 May 1, 1996, as supplemented on May 31, August 14, August 26 and September 11, 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) <u>Technical Specifications</u>

The Technical Specifications contained in Appendix A, as revised through Amendment No. 128, 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 25, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 128

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 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 4.2-9
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	TS B4.2-6
Figure TS 4.2-1	Figure TS 4.2-1

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 hybrid expansion joint (HEJ) sleeve which, upon inspection, exhibits wall degradation of 31% or more shall be plugged or repaired prior to returning the steam generator to service. For disposition of parent tube indications (PTI), the following requirements will apply:

1. HEJ sleeved tubes with circumferential indications located within the upper hardroll lower transition shall be inspected with a non-destructive examination (NDE) technique capable of measuring the sleeve ID difference between the sleeve hardroll peak diameter, and the sleeve ID at the elevation of the PTI. If this diameter change is $\geq$ 0.003" (plus an allowance for NDE uncertainty), the indication may remain in service provided the faulted loop steam line break (SLB) leakage limit from all sources is not exceeded. A SLB leakage allowance of 0.025 gpm shall be assumed for each indication left in service regardless of length or depth. For tubes where the diameter difference is > 0.013", SLB leakage can be neglected.

2. HEJ sleeved tubes with a sleeve ID difference of < 0.003" (plus an allowance for NDE uncertainty) between the sleeve ID hardroll peak diameter and sleeve ID at the elevation of the PTI shall be plugged or repaired prior to returning the steam generator to service.

3. HEJ sleeved tubes with axial indications located within the parent tube pressure boundary as defined on Figure TS 4.2-1 shall be plugged or repaired prior to returning the steam generator to service.

4. HEJ sleeved tubes with parent tube indications located outside of the parent tube pressure boundary as defined on Figure TS 4.2-1 may remain in service.

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. <u>Tube Support Plate Plugging Limit</u>

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 ≤ 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.

Amendment No. ~~52, 73, 76, 93, 95, 103, 118, 126, 127~~, 128

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 hybrid expansion joint (HEJ) 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.

---

[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.

The pressure boundary for HEJ sleeves is shown on Figure TS 4.2-1. The pressure boundary used to disposition parent tube indications (PTIs) detected in the upper joint of HEJ sleeved tubes is discussed in WCAP-14641[6]. The pressure boundary will allow PTIs located such that there is a minimum diameter change of 0.003 inch (plus an allowance for NDE uncertainty) between the peak diameter of the sleeve hardroll, and the diameter at the elevation of the PTI, to remain in service. The 0.003 inch interference lip is derived from structural and leakage testing. When inspecting and dispositioning the PTIs, the acceptance criteria will be adjusted to account for measurement uncertainties associated with the technique used to measure the relative change in ID sleeve diameters. During field application, the PTI elevation will be measured by comparing the diameter reported at the peak amplitude of the flaw, and the diameter at the center of the plus point coil's field, and using the more conservative of the two diameters to perform the ΔD determination. Application of the pressure boundary for HEJ sleeved tubes provides allowance for leakage in a faulted loop during a postulated steam line break (SLB) event. A SLB leakage of 0.025 gpm is assumed for each applicable indication. Steam line break leakage from all sources must be calculated to be < 34 gpm in the faulted loop. Maintenance of the 34 gpm limit ensures off-site doses will remain within a small fraction of the 10 CFR Part 100 guidelines for a SLB.

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.

[6]WCAP-14641, "HEJ Sleeved Tube Structural Integrity Criteria: Diameter Interference at PTIs," April 1996.

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:

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

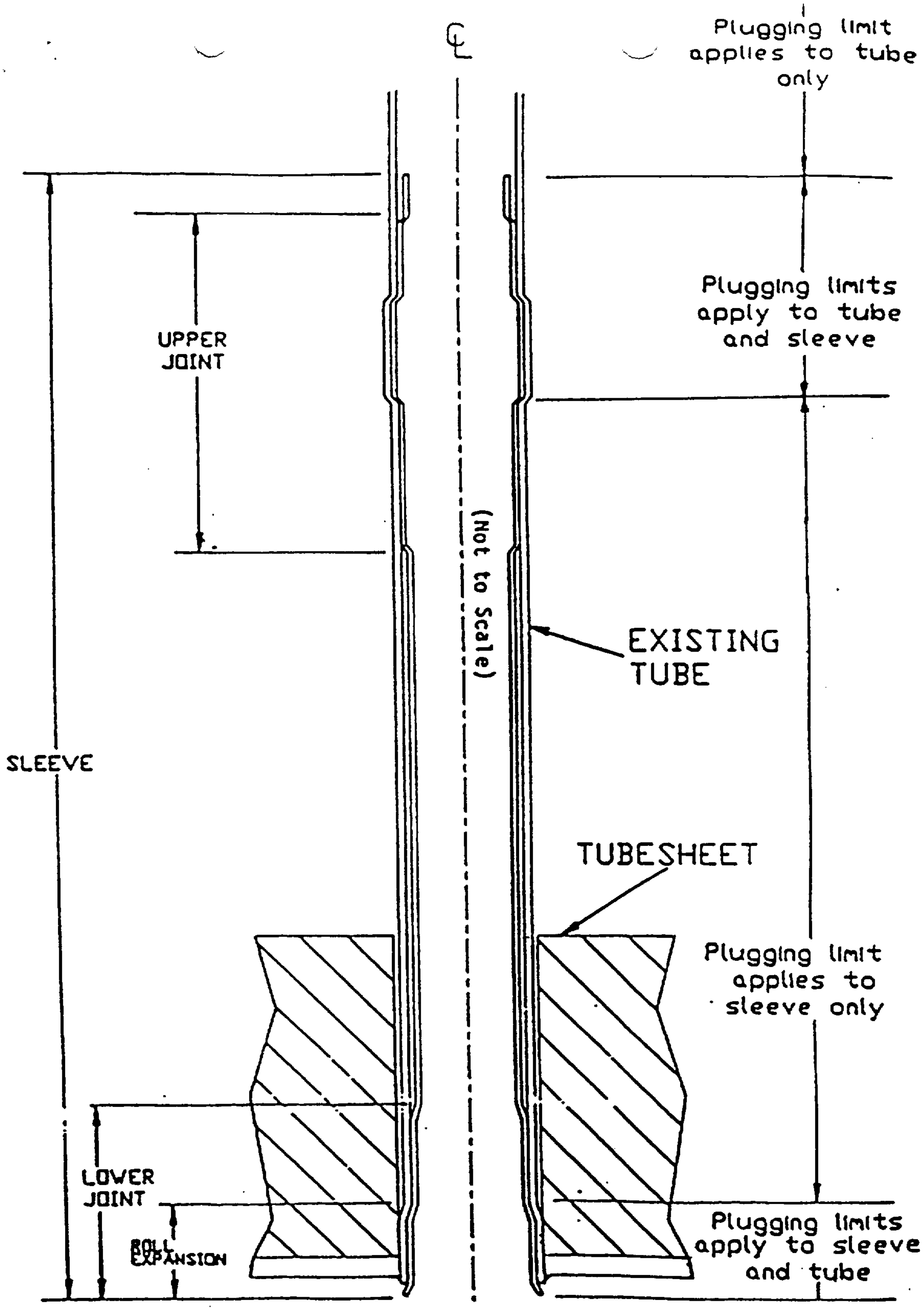
Where V_{GR} represents the allowance for flaw growth between inspections and V_{NDE} represents the allowance for potential sources of error in the measurement of the bobbin coil voltage. Further discussion of the assumptions necessary to determine the voltage repair limit are discussed in GL 95-05.

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.



* Refer to TS 4.2.b.4.b

Application of Plugging Limit for a Westinghouse Mechanical Sleeve

Figure TS 4.2-1

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO AMENDMENT NO. 128 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 May 1, 1996, as supplemented on May 31, August 14, August 26 and September 11, 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 Kewaunee Nuclear Power Plant (KNPP) Technical Specification (TS) 4.2.b, "Steam Generator Tubes," its associated bases, and Figure TS 4.2-1 by redefining the pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves.

The May 1, 1996, submittal superseded a previous submittal on this subject dated October 6, 1995, as supplemented on November 8, 1995, and January 8 and January 19, 1996. The May 31, August 14, August 26 and September 11, 1996, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination published in the May 22, 1996, Federal Register.

The proposed pressure boundary relocation is based, in part, on analytical evaluations, the results of prototypic testing, and the results of destructive examinations and tests of HEJ specimens removed from the Kewaunee SGs. The licensee has concluded that the results from these tests show that HEJs with parent tube indications (PTIs) located within the hardroll lower transition have sufficient structural integrity with regard to the requirements of RG 1.121, sufficient leakage integrity with regards to 10 CFR Part 100 guidelines, and do not compromise the safety of the SG tube bundle when there is an interference lip of 0.003 inch (3 mils) or more. The difference in diameter between the peak diameter of the hardroll expanded portion of the upper joint of the HEJ sleeve and the location of a PTI is generally referred to as the interference lip.

2.0 BACKGROUND

Westinghouse hybrid expansion joint (HEJ) sleeves have been used to repair SG tubes that have exhibited various forms of SG tube degradation thereby allowing the tube to remain in service. The HEJ sleeve assembly is generally attached to the tube by first performing a hydraulic expansion of the sleeve and the tube at the sleeve ends. A mechanical roll expansion (i.e., hardroll) is then performed at the bottom end of the sleeve followed by a mechanical roll expansion within the upper hydraulically expanded region. The sleeve lengths can vary depending on their location within the SG.

The licensee is proposing to redefine the pressure boundary at the upper joint of the sleeve. Specifically, the boundary would be at the lower expansion transition region between the hardroll region and the hydraulically expanded region. This region is referred to the hardroll lower transition (HRLT). The HRLT is a tapered or sloped transition between the hardroll and the smaller diameter hydraulic expansion of the HEJ. The HRLT varies between 0.25 and 0.5 inch in width.

A significant number of SG tubes in the Kewaunee SGs with Westinghouse HEJ sleeves have exhibited degradation, PTIs, at the upper expansion joint HRLT. The PTIs were detected by eddy current test (ET) after 4 to 7 cycles of operation for a particular sleeved joint. To characterize this degradation three tubes were removed from a Kewaunee SG in 1995. Based on the destructive evaluation of these tubes, the nature of the PTIs was determined to be circumferential intergranular corrosion cracking which initiated from the inside diameter (ID) of the parent tube. The cracks were segmented and the initiation sites were scattered slightly in elevation (i.e., the cracks were non-coplanar). All HEJ sleeved tubes with such PTI indications within the HRLT were removed from service. Rolled joints at the lower end of a sleeve that are within the tube sheet are unlikely to experience the cracking problem.

The proposed relocation of the pressure boundary would allow the majority of the HEJs with PTIs to remain in service. This is because the majority of the cracks are located at the approximate midpoint of the HRLT. At the midpoint of the HRLT, the diameter difference between the location of the indication and the hardroll straight portion exceeds the amount proposed as necessary to ensure structural integrity of the HEJ. Thus, the indications would be outside the proposed pressure boundary, and their presence would be immaterial to the pressure boundary integrity. Since the indications would be effectively removed from the pressure boundary, issues related to flaw size, potential flaw growth, growth rates, reinspection intervals, flaw evaluation methods and the impact of the flaws upon the pressure boundary integrity would be avoided.

The proposed amendment request by WPSC (and a duplicate request submitted by American Electric Power for their D.C. Cook plant) is an evolutionary proposal based upon a prior request made by Wisconsin Electric Power Co. for use at that licensee's Point Beach Nuclear Plant. The TS amendment request for Point Beach was dated August 26, 1994. The amendment request was evaluated and denied by the staff as detailed in the Safety Evaluation (SE) dated January 11, 1995. The reasons for the denial were insufficient evidence to resolve certain technical issues. The principal unresolved issues included root cause of the observed HEJ cracking, crack growth rates, and non-destructive examination (NDE) capabilities to detect and adequately size the indications.

Subsequent to the issuance of the SE denying the Point Beach amendment request, representatives from WPSC, Westinghouse, and others, met with the NRC staff on February 1, 1995, to discuss the unresolved technical issues. Additional meetings were held April 13, 1995, December 8, 1995, January 31, 1996, April 25, 1996, June 17, 1996, and August 20, 1996. These meetings, and numerous conference calls, were held as ongoing tests and analyses were performed (primarily by Westinghouse) to resolve the staff identified issues arising from the Point Beach SE. During the evolution of these additional studies, many previously identified issues were resolved and some were discarded as new approaches were developed by the licensee. This SE examines the key issues of these studies that pertain to NDE, and to the structural and leakage integrity of HEJs with PTIs in the HRLT.

3.0 DISCUSSION

A principal issue remaining after the Point Beach evaluation was the root cause of the PTIs. No ex-service tube samples with indications were available for laboratory examination to verify the accuracy of the ET method or provide a root cause determination. At the time, the evidence from the ET examinations implied the degradation mechanism to be outside diameter (OD) initiated stress corrosion cracking (SCC).

3.1 Root Cause Determination

A significant unresolved issue was the question of whether the observed indications in the HRLT were a precursor to future cracking at other locations in the HEJ. The same type of circumferential ET indications were also detected in some of the lower hydraulic transitions of the upper HEJ. The staff found that these indications had no adverse structural or leakage impact upon the HEJ, but it did heighten interest in the precursor issues. The primary unresolved questions included (1) why did the indications occur at this location, and (2) did evidence of indications at this location indicate that the same degradation mechanism could be acting anywhere in the HEJ?

Since a root cause determination was essential to resolving these and other issues, the licensee removed 3 tubes, each containing an HEJ with PTIs, from the hot leg of Kewaunee SG "B" during April 1995. The sample tubes were extensively examined and tested at the Westinghouse metallurgical facility. Third party (peer) review was provided by staff metallurgists of the Electric Power Research Institute (EPRI). The results of the root cause investigation were detailed in Westinghouse WCAP-14446, "Repair Boundary for Parent Tube Indications Within the Upper Joint Zone of Hybrid Expansion Joint (HEJ) Sleeved Tubes," dated August 1995 (proprietary). The principal findings of this report were correlation of NDE results with actual flaws, flaw characterization, root cause determination, and leak rate and structural integrity of flawed HEJs.

Prior to sectioning the samples for metallurgical analyses, each sample was subjected to a number of nondestructive examinations including ET, radiography, dimensional characterization, and visual examination. The NDE results were carefully indexed so that any indications could be directly compared with subsequent metallurgical examination results. Results from the various ET probe types could also be used to provide a better correlation of ET signals against field data.

Room temperature tensile testing was conducted on two of the HEJs, as well as on three free span sections, one from each tube. The bulk of the third tube/HEJ section was retained intact as an archive specimen. The free span tensile tests were for confirmation of the sample material properties prior to further destructive tests. The mechanical properties derived from the tensile tests of the free span tube material were normal, thus the tensile (pull out) test of the rolled joint would give typical joint strengths.

The two HEJs were fixtured and loaded until failure. Both HEJs had high separation loads, in excess of 10,000 pounds. These loads are far in excess of the most limiting criteria of RG 1.121 (minimum load of 1516 pounds for Kewaunee).

The two HEJs which were tensile tested were then destructively examined using metallographic and scanning electron microscope (SEM) fractography techniques to characterize the fracture faces and any corrosion. An analysis of the OD and ID deposits, ID oxide films, and fracture face oxide films was performed using a variety of energy dispersive analytical techniques. In addition, ion chromatography and capillary electrophoresis were performed on soluble ID deposits obtained by water leaching.

Post-tensile test visual examination showed that ID originating, circumferentially oriented, corrosion cracks were present continuously around the circumference of the tube fracture faces of both HEJs that were separated by tensile testing. The tube tensile separations occurred in circumferential macrocracks that were composed of numerous circumferentially oriented intergranular microcracks of ID origin that were aligned in a single narrow band (less than 0.12 inch height). A large fraction of the many ligaments separating the microcracks had ductile features. Many other ligaments had only intergranular features, indicative of SCC. No corrosion degradation was observed at any other location of the ID or OD of the tubes, and no corrosion was noted at any location of the alloy 690 sleeve material.

All intergranular features were confined to the fracture zone of the parent tube HRLT regions where the separation occurred (no cracking occurred in the alloy 690 sleeve material). The fracture faces exhibited a widely varying depth of intergranular cracks. The intergranular fracture faces had a maximum of 92% and an average of about 60% throughwall depth. The morphology of the cracks was that of primary water SCC (PWSCC), rather than that of secondary side corrosion that typically occurs in caustic crevices.

Following SEM examination of the fracture surfaces and deposit analysis, a narrow axial metallographic section was cut from each tube through the HEJ region. Microhardness measurements were made at selected locations near the intergranular cracks, away from the tensile shear faces. The microhardness next to the intergranular fracture faces were generally similar to or higher than microhardness values measured elsewhere. The inner-most microhardness values included the highest measured.

The chemical analyses of the tube ID surface below the fracture face of the HRLT region revealed high concentrations of boron and lithium. The presence of these two elements demonstrates primary water intrusion into the creviced area below the HRLT. Since it is known that rolled joints are not necessarily leak tight, it was concluded that the observed cracking was due to PWSCC.

With the finding of PWSCC as the root cause, the staff posed the question of why the lower transitions cracked in preference to the upper transitions that were, regardless of leakage past the hard roll, always in contact with the primary side environment. Since an HEJ is symmetrical, it is logical to assume that the residual stresses would be the same at both the upper and lower transitions.

To answer this question, Westinghouse constructed a number of full size mock-ups consisting of alloy 600 tubes with alloy 690 sleeves. Prior to installing the sleeves, the tubes were instrumented with strain gages to measure any far field stresses that might result from the sleeve installation process. The sleeves were installed following normal field practice, with field tooling. The mock-ups of the first tube support plate above the tubesheet were constructed two ways. Some provided no constraint. Some provided constraint as in a tube support plate lock-up situation.

The test results showed that the net effect of the installation of a sleeve is a compressive stress in the sleeve and a tensile stress in the tube below the HEJ. This finding was shown for both conditions of tube constraint in the tube support plate. The difference in far field stresses was roughly 10 to 15 ksi: higher below the HEJ compared to above it. When considered in conjunction with the local plastic deformation residual stresses in the transitions themselves, the tests demonstrated why the lower transitions were the preferred crack sites. When the effects of thermal expansion were added, it was noted that the lower transitions experienced additional tensile stress. From these tests and analyses, the staff concluded that an acceptable root cause determination had been made.

The staff notes that the root cause determination does not rule out the possibility for future degradation of the upper portions of the HEJ. However, based upon operating experience with sleeves of all types, the root cause determination, and the far field stress measurements, the staff concludes that any potential cracking that could occur at these other locations will not progress at a rate greater than that previously experienced at the tube transition zone at the top of the SG tube sheet. Crack growth rates in this zone have been noted to be relatively slow, taking several operating cycles to progress from initiation to a size that could adversely impact the pressure boundary. Therefore, it is reasonable to expect that any SG with a substantial number of HEJ sleeves in service can be adequately inspected by maintaining the currently mandated inspection schedule for SG tubes.

3.2 Leak Rate Determination

One sample was leak rate tested prior to being sectioned for metallography. The leak rate test was conducted at elevated temperatures and pressures simulating normal operating and design conditions. No leakage occurred. This result confirmed previous laboratory tests on mock-ups that were performed to support the Point Beach amendment request. These tests were discussed in Westinghouse WCAP-14157, "Technical Evaluation of Hybrid Expansion Joint (HEJ) Sleeved Tubes With Indications Within the Upper Joint Zone," dated August 1994 (proprietary), and WCAP 14157, Addendum 1, "Supplemental Leak and Tensile Test Results for Degraded HEJ Sleeved Tubes in Model 44/51 S/Gs," dated September 1994 (proprietary).

At the time of the Point Beach SE, the staff found that the leak rate tests were insufficient because of the unanswered question regarding postulated multiple flaw sites at different locations in a single HEJ. Potential multiple flaw sites implied a higher leak rate than that measured in mock-up tests and by calculation. With the subsequent finding of root cause, the staff re-reviewed these previous leak rate tests.

For the leak rate tests, HEJs were produced with the tube completely removed at various locations within the HEJ and leak rate tested at elevated temperatures and pressures simulating steam line break (SLB) conditions. Most of the samples exhibited insignificant or no leakage. The sample with the highest measured leak rate was used to establish a conservative bounding value of 0.025 gpm. The staff compared this bounding value to other independent leak rate test data and found it to be conservative.

As applied to the proposed TS change, the total number of indications remaining in service will be limited such that the primary-to-secondary leakage from a postulated SLB will not exceed a small fraction of the 10 CFR Part 100 guidelines. For Kewaunee, this has been calculated to be 34 gpm per faulted loop.

3.3 Structural Integrity of Revised HEJ Pressure Boundary

As originally submitted (October 6, 1995), the proposed TS amendment request approached the pressure boundary relocation based upon a fixed measurement (about 1.1 inch) between the upper roll transition and an indication within the HRLT. As the multiple studies conducted in support of the amendment request evolved, it was determined that a different approach to defining the edge of the pressure boundary (and thus, flaw location) was needed. Consequently, an investigation into the contribution of various portions of the HEJ to the overall joint strength was conducted. The results of this investigation led to the adoption of a differential diameter based definition of the pressure boundary edge (and thus, allowable indication location).

As detailed in Westinghouse WCAP-14157 and WCAP-14157, Addendum 1, and discussed in the Point Beach SE dated January 11, 1995, an HEJ is capable of withstanding tensile loads far in excess of design requirements provided some portion of the HRLT lip is present. The structural tests were extended for the Kewaunee submittal and discussed in Westinghouse WCAP-14446 and WCAP-14641, "HEJ Sleeved Tube Structural Integrity Criteria: "[Delta] D Diametral Interference at PTIs," dated April 1996, (non-proprietary). WCAP-14641 details the tests conducted to support the final version of the Kewaunee amendment request that establishes a pressure boundary lip as the controlling parameter for structural integrity assurance. Additionally, WCAP-14641 reviews prior tensile test results and specimens that were discussed originally in WCAP-14157 and Addendum 1 for the Point Beach amendment request and in WCAP-14446 for Kewaunee.

The structural capability of an HEJ under different conditions was investigated by constructing laboratory duplicates of typical field installed joints using production methods. Numerous test specimens were fabricated to study the effect of different conditions of the HEJ and their resulting effect on strength. Among the varieties of samples fabricated and tested were those with one or more of the following test variables:

1) oxidized versus clean tubing with various machined in defects
2) full circumferential through-wall breaks at differing locations within the HRLT and the lower portion of the hard roll straight region
3) part circumferential through-wall breaks with remaining ligaments of various dimensions
4) samples with various defects (as above) tested with internal pressure applied to represent various design conditions
5) samples with varying degrees of "rolldown" and a range of defect types (rolldown refers to a dimensional variance in the length of taper in the HRLT)
6) minimum and maximum dimensional tolerance joints with various defects as above

Samples were tensile tested to destruction. The amounts of joint slippage (if any), elongation and ultimate load at failure were recorded.

The results of all the various tests revealed several previously unrecognized aspects of the parameters that contribute to (or have minimal impact upon) rolled joint strength. During the earlier stages of the investigation (for the Point Beach amendment request) it was consistently noted that a HRLT with a remaining ligament in only a small portion of the circumference (the rest being completely parted) had greater tensile strength than the free span straight portion of the parent tube or sleeve. Thus, the structural capability of the joint far exceeded requirements. Conversely, variables with little effect upon HEJ strength were minimum/maximum joint dimensions (as reflected in the final expansion diameter) and the degree of "rolldown."

The discovery of the contribution of a small transition lip resulted from tests of HEJs that had the entire HRLT machined away, leaving only the interference fit of the straight roll portion remaining. These joints exhibited significant scatter in tensile strength. Some would slip under loads less than those resulting from normal operating pressure. Some exceeded normal operating loads, but would be challenged under certain design basis accident conditions.

After this, a series of samples with varying amounts of HRLT lip were fabricated and tensile tested. The effect of this lip was to create a small overlap that provided a few thousandths of an inch of diametral interference fit when the HEJ was pulled apart. Upon a slight slippage of the joint, the lip would attempt to ride up over the larger diameter straight rolled portion of the joint. It was found that only 0.001 inch or 0.002 inch of interference was needed to satisfy the design structural requirements. To ensure conservatism, a value of 0.003 inch, plus an amount for NDE measurement uncertainty in field applications, was adopted as an acceptance criterion. The 0.003 inch value tested in the laboratory samples exceeded design requirements by more than a factor of 2.

The conclusion of the structural tests showed that any HRLT with an ET indication at any location in the transition where the ID was at least 0.003 inch less than the hard roll straight section would meet the structural requirements regardless of the circumferential extent or depth of the PTI. In other words, the effect of that portion of the HRLT that had a greater than 0.003 inch diameter difference was immaterial to the integrity of the HEJ and its presence was unnecessary.

3.4 Nondestructive Examination

As discussed above, the structural and leakage integrity of the sleeve joints with PTIs in the HRLT is ensured by verifying that the difference in diameter between the hardroll region and the location of the parent tube indication is greater than 3 mils. This diameter difference, referred to as delta-D, is verified through eddy current testing as discussed below.

For tubes in which PTIs are detected, the licensee has proposed the following eddy current technique for measuring the delta-D. A combination eddy current probe which contains two bobbin coils spaced approximately 1.25 inches apart with a plus-point coil placed between the two bobbin coils, equidistant from each bobbin coil at a distance of 0.625 inch will be used to determine the delta-D. The bobbin coils, operating in the absolute mode, are used to verify consistent translation speeds and to perform profilometry (i.e., diameter) measurements of the sleeve hardroll and the HRLT where the PTI is located. The plus-point coil is used to determine the location of the PTI. The data from the probe is collected while the probe is being pushed through the sleeve.

The location of the PTI is the more conservative diameter measurement obtained from the location corresponding to the peak amplitude of the PTI or the center of the plus-point coil's field (i.e., the center plus-point scan line within the indication). The more conservative diameter measurement will be obtained from the higher PTI elevation since the diameter increases as the axial elevation within the HRLT increases. The larger diameter at the HRLT reduces the delta-D. As discussed below, the peak amplitude for the PTI does not necessarily coincide with the location of the flaw depending on whether or not roll down exists; therefore, the more conservative delta-D is selected for comparison with the acceptance criteria. For tubes without roll down, the peak amplitude flaw signal is not necessarily at the center of the plus-point coil's field. This condition results from the geometry difference between the plus-point coil's gimballed shoe and the non-roll down HRLT geometry. The gimballed shoe is approximately 0.3125 inch wide and a non-roll down transition is approximately 0.375 inch long which results in liftoff as the plus-point coil rides upwards from the HRLT into the hardroll. Since the signal amplitude decreases as liftoff occurs, PTIs actually located in the upper half of the transition will have signal peak amplitudes which occur prior to the center scan line response of the plus-point coil.

With a correct data slew[1], the analyst locates the PTI and performs the diameter measurements with the bobbin coils at the PTI location and at the maximum hardroll diameter. The maximum hardroll diameter is determined by scrolling through the hardroll region and locating the region with the maximum diameter. The difference between the hardroll and PTI diameter measurements, referred to as delta-D, is then calculated for comparison to the acceptance criteria which has been adjusted for an allowance for NDE uncertainty. NDE uncertainty is discussed further at the end of this section.

To determine the diameter of the tube at the PTI, the data from the plus-point coil and bobbin coils must be slewed so that they correspond to the same axial elevation. The location of the PTI is determined from the plus-point coil operated at a frequency of 100 kHz within a C-scan plot. Axial and circumferential cursors are positioned within the C-scan plot to intersect over the peak amplitude of the PTI or are positioned at the center of the plus-point coil's field. With the cursors properly positioned, a diameter measurement is recorded for the sleeve inner diameter using the bobbin coils operating in an absolute mode at 600 kHz (this measurement is made after the data from all three coils has been slewed such that each coil is at the flaw location). A frequency of 600 kHz was selected to minimize any influence from the parent tube. The slewing process will be performed for each sample (i.e., sleeve joint) in a calibration group rather than just on the initial sample in a calibration group. Slewing performed for each sample was verified by the licensee to give less variation in the diameter measurements than if slewing were performed only on the initial sample in a calibration group.

Variables affecting the slewing process include:

(1) variations in the translation speed which will be controlled by (a) overlaying the leading and trailing bobbin coil's strip chart response (i.e., if a constant axial speed is maintained, the data from the two bobbin coils should overlap), (b) by using an axial encoder and ensuring that the number of revolutions per pulse is maintained in the range of 1.2 to 3.2, (c) by using a probe tensioner which applies a constant force to the probe's poly tubing minimizing the slippage of the probe caused by slack in the probes's poly tubing within the conduit and, (d) by collecting the data on the push which reduces the gravitational effects which might affect the speed of the probe if the probe were pulled through the tube/sleeve geometry;

(2) variations in the digitization rate (i.e., the frequency at which data points are obtained - typically measured in samples per second) will be controlled through procedures;

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[1] Slewing, in this context, is a process of repositioning or realigning the eddy current data so that each coil appears to be at the same location at the same time (this process is necessary since the eddy current probe contains coils positioned at different axial locations along the probe head).

(3) probe coil spacing which (a) will be controlled during the manufacturing process through QA/QC procedures, (b) will be limited to a tolerance of ± 0.002 inch, and (c) will be tested in a three hole standard prior to use in the field to verify proper coil spacing (the standard has three holes spaced 0.625 ± 0.002 inch apart to match the spacing of the probe coils);

(4) backwards tilting of the gimballed plus-point coil as it rides up (i.e., pushed up) the slope of the HRLT which results in the slewed plus-point coil data lagging in time the slewed data from the two bobbin coils. Not accounting for this effect results in a conservative estimate of the diameter of the tube at the location of the PTI. In addition, since the qualification program used to estimate the uncertainty in determining the delta-D (discussed below) did not involve adjusting for this effect (i.e., the lag of the plus point data), it will not be factored into the slewing process in order to maintain consistency.

(5) probe wear which will be controlled by limiting the variation in the amplitude response of the circumferentially sensitive channel to an EDM notch in a standard to 15% of the coil's original response to the notch (the coils response to the notch will be evaluated after every 10 tubes inspected and if it exceeds 15%, the tubes inspected since the last successful probe wear check will be reinspected);

(6) calibration standard accuracy which is controlled by specifying the distance between the holes as 0.625 inch with a tolerance of ± 0.002 inch; and,

(7) analyst variability which will be controlled (a) by providing detailed analysis guidance to the analysts on settings to be used during the slewing process (e.g., span and zoom settings); (b) by having two independent analysts determine the delta-D with any discrepancy in the PTI diameter or hardroll diameter measurement of greater than 0.001 inch being resolved by the level III shift lead analysts; and (c) by providing software that simplifies the slewing process.

To summarize the slewing process, the data from the two bobbin coil's response are slewed to overlay each other. The number of data points used to slew the response of the trailing bobbin coil over the leading bobbin coil is then divided by two to assist in determining the amount to slew the plus-point data (the value is divided by two since the plus-point coil is equidistant between the two bobbin coils). However, as described above, the amount to slew the plus-point data will not be adjusted to account for the tilt backwards of the gimballed plus-point since the qualification program to demonstrate the performance of this technique did not involve adjusting the data to account for this phenomenon. This results in slewing the plus-point data by half of the number of data points used to slew the bobbin data. In the event of an unacceptable slew (e.g., the bobbin coil response from one coil either leads or lags the response from the other coil after the slew has been performed), the area of interest will be reexamined. If subsequent reexaminations still do not provide an acceptable offset between the slewed bobbin profiles, the tube will be removed from service.

To verify the performance of this technique, tests were performed by the licensee on actual HEJ specimens which contained flaws in the HRLT. The samples were fabricated to be representative of the Kewaunee Nuclear Power Plant sleeves which have experienced roll down of the HRLT, along with samples fabricated with no roll down. The samples were examined with the combination probe discussed above and estimates of the delta-D were made. After the non-destructive examination was completed, the samples were analyzed to determine the actual delta-D. A total of ten HEJ specimens with simulated PTIs were tested. Based on the results of these analyses, the licensee concluded that a conservative estimate of the uncertainty associated with determining the delta-D is 0.004 inch. The value of 0.004 inch is based on statistical analyses of the data (specifically, 99% of the errors in the delta-D measurements should be less than 0.004 inch with 95% confidence). As a result, the 0.003-inch acceptance criteria, discussed below, will be adjusted upwards by 0.004 inch to arrive at a final acceptance criteria of 0.007-inch. The non-destructive examination delta-D measurements will be compared to this final acceptance criteria.

With respect to the inspection scope and method for examination of the sleeve joints, the licensee recently modified the TS to incorporate specific sleeve/tube inspection scope and expansion criteria as documented in Amendment No. 127 dated September 24, 1996. Essentially, the licensee will perform a minimum initial inspection sample of 20% and will expand the examination based on the results. The probe to be used during application of the proposed tube repair criteria was described above. The analysts performing diameter measurements will be qualified in accordance with the site specific performance demonstration program. Detailed procedures and testing governing the analysis of the data will be prepared prior to field implementation to ensure the analysts are familiar with the process for measuring diameters. Two analysts will perform diameter measurements for each sleeve with a PTI and the results will be compared. Discrepancies in diameter measurements of 1 mil between independent analysts will be reviewed by a resolution analyst.

The staff has reviewed the licensee's proposal for determining the delta-D of Westinghouse HEJ sleeves. The staff concludes that this technique is acceptable provided it is applied to HEJ joints that are bounded by the testing programs performed by the licensee including the testing program used in determining the uncertainty adjustment of 4 mils.

4.0 CHANGES TO THE TECHNICAL SPECIFICATIONS

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

TS 4.2.b.4.b would be revised by adding specifications 4.2.b.4.b.1 through 4.2.b.4.b.4 to specify acceptance criteria for the disposition of PTIs in SG tubes with Westinghouse HEJ sleeves.

Figure TS 4.2-1 would be revised to reflect the revised boundaries for determining whether the plugging limit applies to the tube only, to the tube and sleeve, or to the sleeve only.

The basis for TS Section 4.2 would also be revised to add discussions consistent with the changes described above.

The staff has reviewed the TS changes discussed above and finds that they consistently incorporate the methodology for relocating the pressure boundary for HEJ sleeves as previously discussed in this safety evaluation and will provide adequate assurance of SG tube integrity. Therefore, the proposed changes are acceptable.

5.0 SUMMARY

Based upon the root cause determination, bounding leak rate tests, structural tests and the capability of the NDE probes to accurately define the flaw location within the HRLT, the staff finds the licensee's proposed amendment, to relocate the pressure boundary of the upper HEJ, acceptable.

6.0 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.

7.0 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 25715). 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.

8.0 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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