June 7, 1997

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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. 135 TO FACILITY OPERATING LICENSE NO. DPR-43 - KEWAUNEE NUCLEAR POWER PLANT (TAC NO. M96562)

Dear Mr. Marchi:

The Commission has issued the enclosed Amendment No. 135 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 April 22, 1997, as supplemented on May 15, and June 2, 1997. The April 22, 1997, submittal superseded a previous submittal on this subject dated September 6, 1996, as supplemented on October 30, October 31, November 7, November 15, and November 27, 1996, and January 23 and January 29, 1997.

The amendment revises TS Section 4.2.b, "Steam Generator Tubes," and its associated Basis, by allowing a laser-welded repair of Westinghouse hybrid expansion joint (HEJ) sleeved 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.

DF01

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



cc w/encls: See next page

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

June 7, 1997

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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Division of Reactor Projects III/IV
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2. Safety Evaluation

cc w/encls: See next page

Mr. M. L. Marchi
Wisconsin Public Service Corporation

Kewaunee Nuclear Power Plant

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. 135
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 April 22, 1997, as supplemented on May 15, and June 2, 1997, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and 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. 135 , 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, 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: June 7, 1997

ATTACHMENT TO LICENSE AMENDMENT NO. 135

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 4.2-2 | TS 4.2-2 |
| 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 | TS B4.2-6 |
| TS B4.2-7 | TS B4.2-7 |
| TABLE TS 4.2-3 (page 1 of 1) | TABLE TS 4.2-3 (page 1 of 1) |

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

b. Steam Generator Tubes

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

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

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

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

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

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

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

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

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

Repaired Tube is a tube that has been modified to allow continued service consistent with plant Technical Specifications regarding allowable tube wall degradation, or to prevent further tube wall degradation. A tube without repairs is a nonrepaired tube. This definition does not apply to the portion of the tube below the F* or EF* distance provided the tube is not degraded (i.e., no detectable degradation permitted) within the F* distance for F* tubes and within the EF* distance for EF* tubes.

Laser Weld Repaired Sleeved Tube is a tube with a Westinghouse mechanical hybrid expansion joint sleeve that has been returned to operable status by use of a laser welded repair process.

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. In addition to the sample required in 4.2.b.2.a through 4.2.b.2.c, all tubes which have had the F*, or EF*, criteria applied will be inspected each outage in the uppermost tubesheet roll expanded region. These tubes may be excluded from 4.2.b.2.c provided the only previous wall penetration of >20% was located below the F* or EF* distance. F* and EF* tubes will be inspected for a minimum of 2 inches below the bottom of the uppermost roll transition. The results of F* or EF* tube inspections are not to be used as a basis for additional inspection per Table TS 4.2-2 or Table TS 4.2-3.

e. In addition to the sample required in 4.2.b.2.a through 4.2.b.2.c, all laser weld repaired sleeved tubes will be inspected at the first in-service inspection following the repair. Subsequent inspections will include a minimum sample size consistent with 4.2.b.2.a.

   During the first in-service inspection and each subsequent in-service inspection, at least 20% of the laser weld repaired sleeved tubes will be inspected using an ultrasonic inspection technique. The laser weld repaired tubes inspected with the ultrasonic technique shall be selected on a random basis. Actions based on the results of the ultrasonic inspection shall be as described in Table TS 4.2-3.

f. 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.

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

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 or for degradation except as specified in TS 4.2.b.6 for tubesheet crevice region in which the F* and EF* criteria is 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 the following:

   WCAP-14685, Revision 3, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant";

   WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes";

   WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)";

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

CEN-632-P Revision 0, "Repair of Kewaunee Steam Generator Tubes Using a Resleeving Technique"; 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 24% 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 ≥ 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 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.

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 24% is applied to the sleeve as shown on Figure TS 4.2-1. 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] and WCAP-13088[4], 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.

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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] 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[5]. 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 $\Delta D$ 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.

Recent inspection information has indicated a potential for the parent tube behind the upper HEJ region to develop service induced degradation. For parent tube degradation within or below the upper HEJ hardroll lower transition, tube operability can be restored by fusing the sleeve and tube using a laser welding process effectively isolating the degradation below the weld. The laser weld repair is performed similar to the initial installation of laser welded sleeves. The laser repair weld for degraded parent tubes with installed HEJ sleeves has been shown to meet the weld qualification, stress and fatigue requirements of the ASME code. All laser weld repaired HEJ sleeved tubes will receive a post weld stress relief at the weld location and ultrasonic inspection to verify weld quality, in accordance with the process described in WCAP-14685, Revision 3[6] and WCAP-14685, Revision 2, Addendum 1[7].

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

[6]WCAP-14685, Revision 3, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant," May 1997 (Proprietary).

[7]WCAP-14685, Revision 2, Addendum 1, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant Addendum 1: Evaluation of Weld Repaired HEJ Sleeved Tubes," April 1997 (Proprietary).

Topical CEN-629-P[8] describes 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 a full depth tubesheet sleeve which is welded near the sleeve upper end and hard rolled into the tube and tubesheet at the sleeve lower end. A variation on the tubesheet sleeve design is the use of a pre-curved sleeve which allows access to the outer periphery of the tube bundle. 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. CEN-632-P[9] describes the steps required to re-sleeve tubes which have existing HEJ sleeves. This report describes the sleeved/tube preparation, re-sleeve installation and the design of a leak tight full depth tubesheet sleeve that is up to 39 inches in length.

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.

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[8] CEN-629-P Revision 2, "Repair of Westinghouse Series 44 and 51 Steam Generator Tubes Using Leak Tight Sleeves," January 1997.

[9] CEN-632-P Revision 0, "Repair of Kewaunee Steam Generator Tubes Using a Resleeving Technique," April 1997.

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

Tubes with indications of degradation in either the original factory roll expansion in the tubesheet or the unexpanded portion of tube within the tubesheet may be dispositioned for continued service or repaired through application of the F* or EF* criteria. The F* and EF* criteria are described in WCAP-14677[10]. The F* and EF* criteria are established using guidance consistent with RG 1.121. Neither the F* or EF* criteria will significantly contribute to offsite dose following a postulated main steam line break such that contributions from these sources need to be included in offsite dose analyses. Inherent to these criteria is the ability to perform an additional roll expansion of the tube, either as an extension of the original factory roll expansion, in which case F* criteria applies, or in the area starting approximately 4" below the top of the tubesheet, in which case EF* criterion apply. The additional roll expansion procedure can be applied over existing degradation, provided the F* or EF* requirements for non-degraded roll expansion lengths of 1.12" (plus an allowance for NDE uncertainty) and 1.44" (plus an allowance for NDE uncertainty), respectively, are satisfied. The NDE uncertainty applied to the F* and EF* distance is a function of the eddy current probe and technique used. Current state-of-the art inspection technology will be used with implementation of the F* and EF* criteria. The uncertainty in such inspections has been shown to be as small as 0.06", however, for field application, an eddy current uncertainty of 0.20" will be applied. Any and all indications of degradation existing below the F* or EF* distance is acceptable for continued service.

Technical Specification 4.2.b.7

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.7.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.7.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.

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[10]WCAP 14677, F* and Elevated F* Tube Alternate Repair Criteria for Tubes With Degradation Within the Tubesheet Region of the Kewaunee Steam Generators, June 1996 (Proprietary).

## 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 or repair defective repaired tubes and inspect all remaining repaired tubes in this S.G. | C-1 | None |
| | | | C-2 | Plug or repair 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 or repair defective tubes and inspect 20% of the repaired tubes in the other S.G.<br><br>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.<br><br>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. 135 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 April 22, 1997, as supplemented on May 15, and June 2, 1997, Wisconsin Public Service Corporation (WPSC), the licensee, requested a revision to the Kewaunee Nuclear Power Plant (KNPP) Technical Specifications (TSs). The proposed amendment would revise KNPP TS Section 4.2.b, "Steam Generator Tubes," and its associated Basis, by allowing a laser welded repair of Westinghouse hybrid expansion joint (HEJ) sleeved steam generator (SG) tubes. The April 22, 1997, submittal superseded a previous submittal on this subject dated September 6, 1996, as supplemented on October 30, October 31, November 7, November 15, and November 27, 1996, and January 23 and January 29, 1997.

The May 15, and June 2, 1997, submittals provided clarifying information that did not change the initial proposed no significant hazards consideration determination published in the Federal Register on May 7, 1997 (62 FR 24988).

A significant number of Kewaunee SG tubes containing HEJ sleeves have experienced service-induced degradation in the parent tube portion of the HEJ. The degradation is due to primary water stress corrosion cracking (PWSCC). It is in the form of circumferentially oriented cracks located within the lower hardroll transition of the upper HEJ. The upper HEJ is the sleeve to tube joint located in the tube freespan above the tubesheet. Since the degradation has affected the structural integrity of the HEJ, the tubes would normally require plugging.

As an alternative to plugging, the licensee has proposed a repair for the affected HEJs. The proposed repair entails adding a laser weld within the existing HEJ. The weld would replace the hardroll portion of the HEJ as the structural boundary and essentially modify the HEJ sleeve to the configuration of a conventional laser welded sleeve. This proposed repair method is referred to as a laser welded repair (LWR).

Extensive analyses and testing were performed for the licensee by the vendor (Westinghouse) on LWR mockups to demonstrate the feasibility of the concept and verify that Regulatory and Code design criteria were satisfied under normal operating and postulated accident conditions. Details of the LWR tests and qualifications are discussed in Westinghouse report SGO-ATD-96-13, "Interim Report on Laser Weld Repair of Hybrid Expansion Joint Sleeves" (proprietary), dated April 18, 1996, Westinghouse report WCAP-14685, Revision 3, "Laser Welded Repair of Hybrid Expansion Joint Sleeves for Kewaunee Nuclear Power Plant" (proprietary), dated May 1997, Westinghouse report WCAP-14685 Revision 2, Addendum 1, "Evaluation of Weld Repaired HEJ Sleeved Tubes", dated April 1997, public meeting notes dated October 10, 1996 (proprietary), and the licensee responses of October 31, November 7, November 15, November 27, 1996, January 23 and January 29, 1997 to staff requests for additional information (RAI). Additionally, the licensee, Westinghouse, and NRR staffs discussed the status of the repair effort development at meetings that took place on October 10 and December 17, 1996, January 14, March 24, and April 14, 1997.

Additionally, the licensee conducted performance demonstrations of the LWR process. As a result of the performance demonstrations a number of improvements were made to the LWR process. The resulting modifications and improvements to the proposed repair process are discussed in the licensee letters of November 27, 1996, January 23, April 22, and May 15, 1997, and detailed in WCAP-14685, Revision 3, and WCAP-14685, Revision 2, Addendum 1.

The licensee chose to proceed with the proposed repair method at their own risk pending review of the process by the NRC staff. The LWR effort was completed with partial success (additional repair strategies were also developed and implemented).

## 2.0 BACKGROUND

The Westinghouse HEJ tube sleeve derives its name from the two step expansion sequence. After the sleeve is inserted into the tube being repaired, the sleeve ends are hydraulically expanded into contact with the tube inside diameter (ID). Then a portion of the hydraulically expanded sleeve ends are further expanded with a mechanical rolling tool to form the interference fit which becomes the structural and leak limiting joint.

After four to seven operating cycles with HEJ sleeves in service, the licensee discovered circumferential cracks in a significant number of the HEJ sleeve joints. The cracks were in the parent tube material (not the sleeve) at the diameter transition, or step, between the hydraulic and hardroll expansion. The only joints that cracked were the upper, or freespan joints above the tubesheet. The specific location where cracking occurred within the upper HEJ was primarily at the lower hardroll step (some percentage of the joints had cracks in the parent tube at the lower hydraulic step, but these cracks were of no structural or leakage significance).

Several sleeved tubes containing cracked HEJ's have been removed at various times for metallurgical examination. It was found that the parent tube material had cracked due to PWSCC. A combination of slight primary coolant leakage past the hardroll, along with the residual stresses at the hardroll step (from installation), was the cause of the localized cracking. Further tests of HEJ's (laboratory mockups) revealed that the lower hardroll step has a higher residual stress than other locations within the HEJ. This knowledge of the root cause for the HEJ cracking allowed development of an alternative load bearing joint (laser weld) that would alleviate the parent tube degradation.

The proposed repair involves making a laser weld to join the sleeve to the tube. The weld thereby replaces the HEJ as the pressure boundary transition between the sleeve and the tube. This proposal conceptually changes an HEJ into a conventional Westinghouse laser welded sleeve design such as has been licensed for use at Kewaunee and other facilities.

The technical issues connected with performing the proposed repair were generally similar to those previously evaluated by the staff for installing a new laser welded sleeve. However, a number of changes to the previously accepted method for installing a laser welded sleeve (LWS) were necessary. The principal changes were:

1. An option for performing the weld in either the hardroll or upper hydraulic expansion;

2. Addition of a preheat step to remove any possible entrapped moisture from the joint crevice; and

3. Addition of a second ultrasonic test (UT) inspection after the postweld heat treatment (PWHT).

Additionally, after the LWR effort was completed, the licensee removed six tube/sleeve assemblies for destructive examination, verification of the nondestructive examination (NDE) method, and verification of the structural and leakage integrity of production welds. In-situ hydrostatic tests (at operating and postulated accident pressures) were also performed on 14 tubes to further verify the weld integrity. This safety evaluation (SE) will detail the issues that differ from a new LWS installation (an amendment for Westinghouse LWS was approved for Kewaunee and issued September 24, 1996).

## 3.0 DISCUSSION

For a new LWS installation, an automatic autogenous laser weld joins the sleeve to the parent tube. Prior to installing the sleeve, the sleeve interior and exterior surfaces, and tube (at the desired location of the weld), are cleaned to bright metal. The sleeve is inserted into the tube and the sleeve end in the region of the intended weld is hydraulically expanded to contact the parent tube. The weld is then executed.

For the proposed LWR, the weld would be applied within the existing hardroll region above the location of the parent tube degradation (welds in this location are referred to as HR welds). The HR weld would form a new pressure boundary transition between the sleeve and the tube. Thus, the original sleeve to tube joint (the HEJ) would become redundant. Degradation in the parent tube within the HEJ would be immaterial to the structural and leakage integrity of the repaired joint.

The proposed repair was conceptually similar to a new LWS installation. To verify the technique, a series of in-situ demonstration welds on degraded HEJ sleeves were performed. After several demonstration trials, some modifications and enhancements to the process were incorporated. A significant addition was the incorporation of a welding preheat step. This would aid in removing any residual moisture that could be trapped within the creviced regions inherent in an HEJ.

The preheat addition and some welding technique changes were incorporated into the repair procedure and production welding commenced. After roughly 650 sleeves were welded, the licensee determined that the weld acceptance rate was insufficient for their purposes. A number of welds were exhibiting porosity. Reweld attempts were failing due to hot cracks. All production work was stopped. The NRC staff was appraised of the results, and the licensee and vendor commenced detailed laboratory studies to correct the welding problems.

During the laboratory studies conducted to improve HR weld quality, approximately 350 HEJ sleeved tube mockups with welds were made and tested. Test specimens were produced that duplicated the range of conditions experienced with typical field installations. After weld specimens were produced, the samples were inspected by UT and eddy current test (ECT), using different probes and/or techniques. Then samples were sectioned for metallurgical examination. The metallurgical examination results were used to evaluate both the welding technique and the capability of the various NDE methods. The weld samples included numerous intentional weld defects. The defects were produced as part of the qualification program for the NDE methods and for understanding the tube/sleeve conditions and welding parameters that could result in unacceptable welds. With knowledge of the defect causes, the intention was to revise the welding technique to alleviate the conditions that were resulting in unacceptable welds.

### 3.1 Welding Technique Modifications

When the initial demonstration HR welds showed a significant percentage of defects due to porosity, the previously noted preheat step was added because it was believed that entrapped moisture between the sleeve and tube was the cause of the porosity. Addition of the preheat step aided the situation, but not to the extent desired. It was then decided that the hardroll region was not conducive to effective drying during preheat due to the interference fit that exists between the sleeve and tube. Trapped moisture would have difficulty escaping during the preheat. Instead, it would vent through the weld puddle, creating porosity. Moving the weld location to the hydraulic expansion region (HE weld) would allow better venting of any moisture during preheat since the sleeve to tube clearance was greater here. The laboratory tests confirmed a significant decrease in porosity problems when the weld location was moved.

The metallurgical effects of the preheat step have been evaluated. The additional heating time will not cause any adverse effect on the metallurgical structure of the sleeve or tube. A time delay between the preheat step and the laser welding will be maintained to allow the metal temperature to return to ambient conditions so the expansion stresses due to preheating and welding will not be additive.

If the temperature range for the preheating is similar to the temperature used for PWHT, the subsequent ECT would not be able to differentiate between the effects of the preheat and the PWHT. In this case, process controls must be used to ensure the correct positioning of the heater and application of the qualified PWHT in lieu of the ECT verification. The licensee implemented administrative controls during the repair campaign to ensure that PWHT was performed on all welds.

The conventional LWS installation procedure allows several rewelds over an unacceptable initial weld. A reweld over the original weld can repair some weld defects such as porosity or insufficient weld width. When rewelds were attempted on the hardroll area welds, hot cracks frequently occurred in the sleeve portion of the weld. Westinghouse conducted tests of heat input (power levels) during rewelds. The weld mockups revealed that revised heat input during the weld passes would alleviate the propensity for hot cracks. Consequently, the welding procedure was modified to adopt different heat input requirements during the various passes of an initial weld or reweld. This modification had the greatest benefit when welds were performed in the hardroll region (the region with greatest propensity for hot cracks to develop).

The minimum distance between an HEJ indication and a hardroll area repair weld was decreased after it was shown that the weld could be accurately placed, and/or its position confirmed by NDE with respect to the parent tube defect. Since a parent tube defect acts as an edge or end of the tube, its proximity to a weld is immaterial structurally. Placement of the weld adjacent to a parent tube crack is principally a matter of how accurately the remote manipulator tooling is able position the laser weld head. Confirmation of the weld position can be determined by NDE measurement. Laboratory confirmation of the accuracy and repeatability of the NDE to resolve the indication from the weld permitted the smaller proposed minimum distance. This allows greater flexibility for weld placement within the hardroll, should it be needed.

### 3.2 HE Weld Experience

From the results of the HR weld experience and the laboratory tests to improve HR weld quality, the licensee decided to adopt HE welds as the preferred repair method. The LWR technique used for the remaining (approximately 850) repair welds incorporated the following changes:

1. The primary weld location was moved from the approximate mid-point of the upper hardroll expansion (HR weld) to the upper hydraulic expansion region (HE weld);

2. An optional preheat step was added to dry the crevice area of the sleeve/tube interface prior to welding;

3. An option for placing the primary weld in the hardroll region was retained; and

4. An optional repair weld location inboard of either of the two primary weld locations was added (HR or HE).

The HE welds were produced with a much greater acceptance rate. After NDE for weld acceptance and baseline data gathering, the welds were heat treated. Then the secondary side of the SG was flooded for leak testing. Several HE weld joints were discovered to be leaking very slightly. Re-inspection with NDE revealed UT signal changes in the welds with suspected leakage and in a substantial percentage of the previously acceptable welds with no apparent leakage. As a result of the number of now evidently faulty welds, along with the changes in UT signals, the licensee stopped work, advised the staff of the problem, and made plans for removing sleeve/tube assemblies for destructive examination and root cause determination.

3.3 Destructive Examination of Production Welds

Six sleeved tube assemblies were removed for laboratory examination. Five were HE welds: two had acceptable UT results and did not leak; three had unacceptable UT results after the PWHT. Of these three, two leaked in the field. An HR weld with a marginal weld width comprised the sixth sample.

Prior to sectioning for metallurgical examination, all six samples were subjected to a battery of NDE methods to fully characterize the types of indications, if any. Helium leak tests were performed to verify the field observations under more controlled conditions. One sample was archived (not sectioned).

The metallurgical examination revealed the cause of the change in UT signal and leakage to be due to hot cracks originating at the edges of the HE welds. Although very small (on the order of 0.001 inch) hot cracks are normally seen at the edge of a laser weld. Their presence in a typical new LWS installation (or the HR welds at Kewaunee) is of no significance.

The cause of the large hot cracks in the HE welds was determined to be the result of excessive stresses on the weld while it cooled. While the weld is cooling, but still at about 2,000°F, the material is susceptible to forming hot cracks, which initiate as grain boundary separations. If the stresses during cooling remain high, the grain boundary separations can grow or link up to form visible cracks. In the case of the HE welds, the cracks were formed, but grain separation did not consistently occur (due to stress variations from tube to tube). When the welds were subsequently heat treated (at a much lower temperature where the material is not susceptible to hot cracking), the differential expansion stresses from heating opened the previously formed grain boundary cracks and tore the material to form detectable cracks. Since the original grain boundary cracks were so small, they were undetectable by the initial UT performed prior to PWHT.

The stresses that formed, and subsequently enlarged the hot cracks, were determined to arise from residual stresses within the original tube/sleeve assembly. This stress results in a shear load upon the HE weld during solidification and the initial moments of the PWHT. The stresses at the HE location were compared to those at the HR weld location and those for a new LWS installation. For a new LWS, the stress is essentially zero, and any hot cracks (due to weld shrinkage stresses) at the weld edges are microscopic. In an HR weld, the constraint provided by the hardroll steps on either side of the HR weld keeps the shear load and, thus, the stress through the weld to low levels. Any weld edge cracks are microscopic, as verified by the pulled tube sample. Additional verification of the absence of significant hot cracks is the unchanged UT signal from before and after PWHT of the HR welds.

3.4 Nondestructive Examination

The baseline nondestructive examination of LWR HEJ sleeves is conducted using UT and ECT. UT is performed after welding to confirm the laser welds are consistent with critical process dimensions and are of acceptable weld quality. Westinghouse presented data on a UT system which demonstrated postweld 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 demonstrated that the UT system can confirm a continuous metallurgical bond between the sleeve and tube and that the weld width meets minimum acceptable dimensions.

Inservice inspection by UT will be in accordance with the requirements of TS 4.2.b.2.e. that specifies that at least 20% of the laser repaired sleeved tubes will be inspected at the first and each subsequent inservice inspection.

ECT is then used to supplement the UT inspection for determining acceptability of the weld and to establish baseline inspection data for every LWR for comparison against future inspections. Acceptability of the weld is determined by:

1. Verifying weld defects (i.e., porosity, inclusions, blowholes, etc.) are not present;

2. Verifying the weld is properly placed within the upper hydraulic expansion (preferred location) or the roll expansion, within the acceptance criteria for the different locations;

3. Verifying the presence of a PWHT if the preheat is not performed; and

4. Verifying the minimum required distance exists between the weld and any degradation of the parent tube below the weld.

The licensee uses the Electric Power Research Institute's, "PWR Steam Generator Tube Examination Guidelines," Appendix H, qualified ECT techniques for the purposes of performing inspections of LWR HEJ sleeves.

A second UT inspection is then performed following the postweld stress relief step. This second UT inspection was added as a result of Kewaunee field experience, which is discussed in Sections 3.2 and 3.3 of this SE. The UT inspection and qualification is discussed in Section 3.4.1.

3.4.1 Ultrasonic Examination

The UT performed following the postweld stress relief step is used to verify that the minimum acceptable fusion zone thickness of the weld is present. This minimum acceptable fusion zone thickness has been shown by analysis to satisfy the requirements of the ASME Code with regard to acceptable stress levels and fatigue usage during operation and accident conditions.

Westinghouse presented data on a UT system to demonstrate that the UT system can confirm a continuous metallurgical bond between the sleeve and tube and that the weld width meets minimum acceptable dimensions. As presented in Westinghouse WCAP-14685, the PWHT UT results were validated by destructive testing (metallography) examination results. That is, regions with UT indications of incomplete weld width (IWW), and those that were observed to have cracking or having no bond between the sleeve and the tube, were confirmed by metallography. Those areas that were acceptable by UT were confirmed by metallography to have continuous bonding between the sleeve and the tube.

3.4.2 UT Acceptance Criteria

The UT on the SG sleeve repair weld is performed to determine the quality of the weld and assure that the minimum width of the weld (0.015 inch) at the sleeve tube interfaces is met. The UT process transmits ultrasound from the piezoelectric crystal transducer through the water couplant until it strikes the ID weld surface of the sleeve. Ultrasonic energy is both reflected and transmitted through the boundary. The transmitted wave propagates through the sleeve and, theoretically, if there is full weld fusion between the sleeve and the parent tube, there will be no reflection of the UT beam at the sleeve/tube interface and the beam will pass through to the far wall and then return. However, the UT transducer employed by Westinghouse to focus the ultrasound wave at the sleeve to tube interface has a beam width of at least 0.020 inch at this location. An acceptable weld may be only 0.015 inch wide. If the beam width at the weld is larger than the weld there will be sound energy reflected because the beam spread is larger than the width of the weld and there will be a return signal from the beam overlap. The smaller the weld the more energy that will be returned. Therefore, by setting a maximum on the amplitude of the reflected signal, the required weld width can be verified. By examining a number of weld samples with marginal and narrow weld widths, a maximum reflection amplitude was established to ensure that no welds less than

0.015 inch wide (the criteria for structural integrity based on the ASME Code margins) would be accepted.

In addition, the acceptance criteria are such that there must be a calibrated amplitude indication of a continuous 360 degree UT front wall (weld surface) reflection signal. The condition of the weld surface is vital for the UT examination. A slight change in the surface profile can change the direction of both the refracted and reflected ultrasonic beams. When the reflected beam is altered, less energy is returned to the transducer and the front wall signal amplitude is reduced. It is important that enough ultrasonic energy enters the weld to have a valid test since the first criterion is based on a return signal that does not exceed a calibrated maximum amplitude of the reflected wave and a weak input signal could cause a bad weld to have the signature of a good weld.

The acceptance criteria allow for the acceptance of a weld in areas which do show reflections from the weld/tube/sleeve interface in excess of the first criterion as long as the analyst can discern a tube backwall signal. This is a tool which the analyst can utilize when good welds fail the first criterion. The first criterion may unnecessarily disqualify a good weld. Return signals from the weld/tube/sleeve interface in excess of the first criterion can occur in good welds due to increased beam spread caused by weld surface variations or variations in the arrival time of the sleeve front wall signal due to nonplanar (rough) weld surface. The basis for the "discernable backwall signal criteria" (which indicates that the signal has indeed passed through a weld fusion zone) was justified by examining weld standards and noting that a backwall signal could not be resolved for welds below 0.015 inch in width. Therefore the criteria for the acceptance of the laser weld is based upon a combination of the observed ultrasonic response from the weld surface, the sleeve/tube interface, and the tube OD backwall.

Other bases for rejecting welds were (1) surface hole indications, (2) protrusion of material on the weld surface, (3) weld mislocation or weld not found, or (4) evidence of a very low power weld pass with no fusion.

### 3.4.3 Staff Evaluation of UT Methodology

At the request of the staff, a meeting was held to review the UT methodology and actual print-outs from the inspections and qualification tests. The acceptance criteria were of special interest and were the main subject of the staff inquiry.

A UTEC computer display system was utilized by the analyst to evaluate ultrasonic inspection data records which were stored on re-writable laser disks. The displays consisted of an A-scan display with calibrated amplitude settings. The available displays were C-scans of the entire weld, with the capability to print out time step B-scans and A-Scan amplitude versus depth (time of flight) plots at any location. A-scans and B-scans were used by the analyst to confirm the presence of back reflection signals.

The staff examined a selection of UTEC computer display print outs from the equipment used to read the inspection data from sleeved tube inspections in order to verify that the methodology used was adequate. A Westinghouse analyst explained and demonstrated the methodology used to disposition the production inspection of the laser welds.

The staff reviewed the written Westinghouse UT acceptance procedure which provides the basis for acceptance of welds in the case that automatic computerized criteria would disqualify the weld. According to the licensee, the "discernible tube backwall signal" criterion was only used when, based on other information contained in the UT signals, there was justification for weld acceptance by examining all UT reflections from the joint. According to the licensee, previous testing had shown that the limited aperture of a weld with width less than 0.015 inch, combined with the defocusing of the sound beam past the narrow weld would consistently prevent a backwall signal from being resolved for welds below the acceptable 0.015 inch in width.

The technical basis for the definition and use of the "discernible tube backwall signal" criterion was evaluated by the staff by review of the basis, by discussion with a UT analyst, and by actual examination of computerized inspection print-outs of the data displays. This system and acceptance criteria were found to be essentially the same as those used previously to qualify the laser welded sleeve installation at Maine Yankee. On the basis of the licensee's reported qualification of the Kewaunee acceptance criteria, staff's experience with the onsite assessment of the UT at Maine Yankee, and the review of print-outs from the Kewaunee UT inspection, the staff finds that there is reasonable assurance of the integrity of the laser repair welds at Kewaunee based on the UT examination.

### 3.5 In-situ Hydrostatic Tests

As further verification of the weld integrity and UT inspection technique, the licensee chose to perform in-situ hydrostatic tests of a selected population of LWR's to include welds with and without UT indications or suspected leakage during a prior 175 psig secondary side hydro test. Eighteen (18) LWRs were ultimately pressure tested. Each of those were tested at three different pressures, up to 2,765 psig, with a hold-time of 10 minutes for each test. All welds accepted by UT passed the hydro test. Eight welds that were rejected by UT did not leak. Four welds with NDE indications did experience leakage of less than 0.005 gpm. These hydro test results, along with the metallurgical examinations, convincingly demonstrated that the UT was capable of detecting flaws that are far smaller than would be significant enough to cause leakage. Additionally, the licensee conservatively chose to plug those few tubes with discrepancies between the UT and the ECT.

### 3.6 Potential for Unintentional Leakage

The licensee will not place any known or suspected leaking LWR tubes in service. The LWR joint is similar to that used for laser welded sleeves. Leakage testing of 3/4 inch and 7/8 inch full-length laser welded sleeve tube assemblies, under conditions considered to be more severe than expected during all operating plant conditions, has shown that the laser weld does not introduce any primary-to-secondary leakage during a postulated steam line break event. In addition, six pulled LWRs leak tested in the laboratory, and 18 tubes that were in-situ leak tested, demonstrated that welds with acceptable ECT and UT are leak-tight at up to main steam line break differential pressures.

Of note, regarding the extensive testing that was performed on these tubes, is the observation that two of these tubes exhibited the interesting phenomenon of allowing a slight amount of leakage from the outside to the inside of the tube, but were leak-tight when pressurized from the inside of the tube, as stated above. Subsequent additional field testing, including secondary side pressure tests, performed between March and May 1997, at test pressures up to approximately 175 psig, identified two tubes in SG B, which had acceptable NDE results that showed minor leakage during the secondary side pressure test. However, neither of these two tubes showed any leakage when in-situ pressure tested at a nominal test pressure of 2,765 psig.

The licensee has concluded that the slight amount of leakage from the outer diameter to the inside of the two tubes is probably the result of a very small, below detection threshold, structurally inconsequential defect in the weld of the repair. The manner in which the weld is loaded during a secondary side pressure test is very different from the manner in which the weld is loaded during a primary side pressure test (or during operation). In particular, the primary side pressure test will tend to place the weld under compression, tending to tighten the joint and close any small defect within the weld. This compressive effect will be enhanced by the differential thermal expansions of the sleeve and tube at operating temperatures. The sleeve material expands more than the tube material, causing the tube sleeve joint to become tighter at higher temperatures. Since these pressure and temperature effects will be present under the operating and accident conditions of interest, and since these effects were not present during the secondary side pressure testing, the licensee concluded that the secondary side test is a conservative and effective way to verify the adequacy of the laser weld sleeve repairs.

These two tubes that exhibited traces of leakage under the secondary side pressure test were subsequently plugged. Based on the in-situ tests performed as well as the secondary side pressure testing, the LWRs are believed to be leak-tight. However, in the unlikely event of a leak passage through a LWR, the aforementioned in-situ hydro test demonstrated the leak rate to be negligible. Thus, should a weld contain a microscopic leak path, the maximum leak rate would be so small as to be unmeasurable by normal in-plant instrumentation. Consequently, the 10 CFR Part 100 limits for radiological release would not be impacted.

3.7 Corrosion Assessment

Thermally treated alloy 600 and alloy 690 laser welded sleeved tube assemblies have performed well historically with regard to corrosion. No service induced degradation of a sleeve weld joint has occurred to date at any plant, with service times of up to about 11 years. Accelerated corrosion tests show that the freespan laser welded joint for initially installed laser welded sleeves (with postweld stress relief) resist cracking for as much as 10 times longer than rolled tube transitions in the same test media. Accelerated corrosion tests also show that nonheat-treated LWS freespan joints exhibit corrosion resistance equal to or greater than rolled tube transitions. These factors suggest postulated sleeve joint degradation, even in a nonheat-treated condition, would occur at a relatively slow rate and be able to be detected by routine NDE inspection prior to reaching any applicable safety margins. The freespan laser welded joint heat treatment process is designed to achieve sufficient stress reduction such that rapid crack initiation and propagation in the joint is not expected.

For the purposes of LWR corrosion assessment, the LWRs are treated as two separate populations, the welds in the HE location and the welds in the HR location. The reason for this is that the stresses and loadings experienced during the welding and post-stress relief application were not the same due to the differing geometric configurations.

For the HR welds, accelerated corrosion testing (doped steam) was performed with bounding tube stress conditions applied during the test. Corrosion samples were fabricated from a tube heat of alloy 600 known to be susceptible to PWSCC. The test results, documented in Section 5 of WCAP-14685, Revision 3, demonstrate that the LWRs located in the HR location do not have a rapid corrosion potential as measured against roll transition control samples. Times to crack formation in the samples were consistent with previous tests of laser welded sleeves.

For the HE welds, a corrosion assessment was performed based on the results of the pulled tube data and information from LWRs performed at the Doel 4 plant. The corrosion assessment is documented in Section 8.0 of Addendum 1 to WCAP-14685, Revision 2. The results of pulled tube destructive examinations of the HE welds revealed there were small hot cracks present in the weld fusion zone. The hot cracks started at the sleeve/tube interface and extended into the weld metal. Results of the Kewaunee pulled tube destructive examination verified that the weldment was alloy 690, which is generally immune to PWSCC. Additionally, the cracks are not oriented such that the principal operating stresses could cause crack growth. Therefore, the cracks are not expected to propagate in-service.

4.0 <u>SUMMARY</u>

Based on the preceding analysis, the NRC staff concludes that repair of Westinghouse HEJ sleeves by laser welding is acceptable as specified in

Westinghouse WCAP-14685, Revision 3, dated May 1997, and WCAP-14685, Revision 2, Addendum 1, dated April 1997.

The licensee proposed the following changes in the TS to implement the laser weld repair methodology discussed above:

1. Proposed Changes to TS 4.2.b, "Steam Generator Tubes"

   The term "Laser Weld Repaired Sleeved Tube" would be added to the list of terms defined to clarify the requirements of the inspection program;

2. Proposed New TS 4.2.b.2.e

   A new requirement would be included to specify inservice inspection requirements for laser weld repaired sleeved tubes (to include UT inspections as discussed in Section 3.4 above);

3. Proposed Changes to TS 4.2.b.4, "Plugging Limit Criteria"

   WCAP-14685, Revision 3, "Laser Welded Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant," and Addendum 1 to WCAP-14685, Revision 2, "Evaluation of Weld Repaired HEJ Sleeved Tubes," would be added to the list of methods allowed for tube repair in TS 4.2.b.4.a.;

   The HEJ sleeve plugging limit in TS 4.2.b.4.b would be reduced from 31% to 24% throughwall due to the use of ASME Code minimum material property values for the sleeve material;

4. Table TS 4.2-3 on repaired tube inspections would be revised to allow repair or plugging of previously repaired SG tubes; and

5. Proposed Revision to Bases Section

   The Bases for TS Section 4.2 would be revised to add a description of the laser weld repair process and make reference to WCAP-14685, Revision 1.

The staff has reviewed the TS changes discussed above and finds that they consistently incorporate the laser weld repair process previously discussed in this SE and will provide adequate assurance of SG tube integrity. Therefore, the proposed changes are acceptable.

## 5.0 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.

## 6.0 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 (62 FR 24988). 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.

## 7.0 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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