ATTACHMENT 1

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

September 6, 1996

Proposed Amendment 144

Description of Proposed Change

Safety Evaluation

Significant Hazards Determination

Environmental Considerations

INTRODUCTION

Wisconsin Public Service Corporation (WPSC) is submitting this proposed Technical Specification (TS) amendment to allow a laser-welded repair (LWR) of hybrid expansion joint (HEJ) sleeved steam generator (SG) tubes. The proposed laser welded repair process, illustrated on Figure 1, is similar to the laser welded sleeving technique that has been licensed and successfully installed at plants with both 3/4 inch and 7/8 inch OD SG tubing. The proposed LWR process can be performed on Westinghouse HEJ sleeved SG tubes that have indications of parent tube degradation in order to return the tube to service, or the LWR can be performed on non-degraded HEJ sleeved tubes as a preventative measure.

The repair would be performed by fusing the tube and sleeve using an autogenous laser welding technique in the approximate mid-point of the existing upper HEJ hardroll expansion. By fusing the tube and the sleeve, parent tube degradation below the weld is isolated and a new pressure boundary is formed. This philosophy is consistent with that used in establishment of the tube/sleeve inspection requirements for the initial installation of laser welded sleeves.

The new pressure boundary (the weld) satisfies both the structural and leakage integrity requirements of the sleeved tube assembly with no change in the flow or heat transfer characteristics of the sleeved tube. A series of fabrication and corrosion tests have been performed to support implementation of the proposed HEJ repair process. The results of the fabrication testing were submitted to the NRC in report SGO-ATD-96-13, titled "Interim Report on Laser Weld Repair of Hybrid Expansion Joint Sleeves," dated April 4, 1996. The results of the corrosion testing program, as well as supporting analytical work, are discussed in WCAP-14685, titled "Laser Weld Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant." WCAP-14685 is included as Attachment 3. The corrosion testing and analytical work demonstrate that production of a laser weld in the upper hardroll region will not result in parent tube stress conditions significantly greater than the traditional laser welded sleeve installation and that these stresses and the associated fatigue usages are bounded by the ASME Code.

BACKGROUND

The KNPP has two Westinghouse Model 51 SGs with 7/8" OD tubing. The KNPP SGs have been experiencing tube wall degradation attributed to outside diameter intergranular attack and outside diameter stress corrosion cracking (ODSCC). As a result of this degradation significant tube plugging and sleeving has been required. Tube plugging initially began in 1983 as a corrective action measure. During the 1988 and 1989 outages a large scale preventative sleeving program was implemented in the hot leg tubesheet crevice region. All of the sleeves installed were Westinghouse mechanical HEJs. Additional sleeving occurred in 1991 using the Westinghouse HEJs, and in 1992 with the Combustion Engineering welded sleeves. As a result of these sleeving programs a total of 2195 HEJs and 12 CE sleeves were installed in SG A, and 2133 HEJs and 4 CE sleeves were installed in SG B.

During the 1994 refueling outage the upper HEJs were inspected using the I-coil, a motorized rotating pancake coil (MRPC) probe developed for sleeve inspections. A total of 77 circumferential crack-like indications were detected in the parent tube, i.e., PTIs. Sixty-six of the indications were within the pressure boundary defined in the KNPP TSs and were removed from service by plugging the tube. The remaining 11 indications were below the upper joint pressure boundary. In 1995, the upper HEJs were examined with the MRPC +Point probe. During this inspection 753 PTIs were detected; 657 in the TS defined pressure boundary and 92 below the pressure boundary. The 657 PTIs located within the defined pressure boundary were removed from service by plugging the tubes. Three specimens were removed from SG B for testing and destructive evaluation.

On October 6, 1995, WPSC submitted a TS amendment request to the NRC to redefine the pressure boundary for the HEJ sleeved tubes. Based on discussions with the NRC staff, this submittal was subsequently revised and resubmitted to the NRC staff on May 1, 1996. The proposed pressure boundary criterion requires a minimum change in sleeve inner diameter between the maximum point of the upper hardroll, and the elevation at the PTI. Upon NRC approval, application of the revised pressure boundary will allow a number of the HEJ sleeved tube with PTIs to be unplugged and returned to service.

The purpose of this TS amendment request is to allow a laser-welded repair of or as a preventative measure HEJ sleeved tubes for which the revised pressure boundary criteria can not be applied. The HEJ laser weld repair process would use the same equipment and welding procedures used for laser welded sleeves. Laser welded repair of degraded HEJ sleeved tubes is considered an extension of the existing sleeving technology. On June 10, 1996, WPSC submitted a TS amendment request to the NRC staff to allow repair of degraded SG tubes with the installation of Westinghouse laser welded sleeves (LWS). The proposed TS change was reviewed in accordance with the requirements of 10 CFR 50.92 and it was shown that no significant hazards exist with regard to repairing the KNPP SG tubes by the use of LWSs. A complete description of the LWS design and qualification testing is provided in WCAP-13088, Revision 3, titled "Westinghouse Series 44 and 51 Steam Generator Generic Sleeving Report," dated January 1994. This WCAP has previously been submitted to the NRC staff and approved as a basis for repairing degraded SG tubes for other Model 51 SGs.

Welding parameters used for the laser weld repair of degraded HEJ sleeved tubes are within the bounds of the original LWS qualification program. For the HEJ repair process, a laser weld will be produced in the approximate center of the upper HEJ hardroll expansion. The integrity of the weld will be verified by ultrasonic (UT) examination and the weld stress relieved by heat treatment.

The LWR satisfies both the structural and leakage integrity requirements of the sleeved tube assembly with no change in the flow or heat transfer characteristics of the sleeved tube. A series of fabrication and corrosion tests have been performed to support the implementation of the proposed HEJ repair process. The results of the fabrication testing were submitted to the NRC in report SGO-ATD-96-13, titled "Interim Report on Laser Weld Repair of Hybrid Expansion Joint Sleeves," dated April 4, 1996. The results of corrosion testing program, as well as supporting analytical work, are discussed in WCAP-14685, titled "Laser Weld Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant". WCAP-14685 is included as Attachment 3. The corrosion testing and fabrication samples verified that production of a laser weld in the upper hardroll region will not result in the potential for rapid corrosion of the repaired sleeve/tube assembly.

Provided below is a description of the proposed change, a safety evaluation, a 10 CFR 50.92 significant hazards determination and an environmental considerations statement. Attachment 2 contains the affected TS pages.

DESCRIPTION OF PROPOSED CHANGE

This proposed amendment request will modify KNPP TS 4.2.b, "Steam Generator Tubes," and the associated bases to permit laser-welded repair of HEJ sleeved tubes. Specifically, TS 4.2.b.4.a will be revised to allow tube repair in accordance with WCAP-14685, and the basis section will be revised to include a discussion on the HEJ repair technique.

SAFETY EVALUATION

Currently, SG tubes at KNPP with indications of degradation in excess of the repair criteria are removed from service by plugging or repaired by sleeving. Removal of tubes from service results in a reduction of reactor coolant flow through the SG. This reduction in flow can have a significant impact on the margins of reactor coolant flow through the SG in the loss-of-coolant accident (LOCA) analyses and on the heat transfer efficiency of the SG.

WPSC has previously submitted a TS amendment request to the NRC staff that justifies the operability of certain HEJ sleeved tubes with PTIs by establishing their structural and leakage integrity consistent with existing NRC guidance. Implementation of the relocated pressure boundary TS amendment would also allow a number of the HEJ sleeved tubes with PTIs to be unplugged and returned to service. The purpose of this amendment request is to allow laser-welded repair of HEJ sleeved tubes for which the revised pressure boundary criterion can not be applied or, as a preventative measure for non-degraded HEJ sleeved tubes. Repair of the HEJ sleeved tubes maintains both the tube heat transfer area and reactor coolant flow area. Therefore, the laser-welded repair process helps to assure that minimum flow rates are maintained.

The proposed amendment will modify TS 4.2.b, "Steam Generator Tubes" and the associated bases to permit the laser-welded repair process. A complete description of the repair process and process qualification are contained in Westinghouse Report SGO-ATD-96-13 and WCAP-14685.

In support of this proposed TS amendment request, this safety evaluation includes the following summary discussions:

- Generic Structural Assessment
- Leakage Assessment
- Integrity Assessment
- Corrosion Assessment
- Repair of Previously Plugged Indications

Generic Structural Assessment

The weld repair process, described in WCAP-14685, has been evaluated to Section III, Subsection NB-3200, of the 1989 Edition of the ASME Code, and includes both fatigue and stress analyses of the repaired HEJ sleeved tube assembly. The sleeve and weld structural analysis uses a generic set of design and transient loading inputs which are intended to bound all plants with Westinghouse Model 51 SGs. The temperature and pressure variances used in the assumed operating conditions and generic transients are conservative when compared to the KNPP operating conditions. The structural evaluation considers the effects of operation on the assembly by considering cases of free and fixed tube support plate conditions and intact and separated tube conditions upon the applied stress. The results of the primary stress evaluation, primary plus secondary stress intensity range evaluation and fatigue evaluation indicate that the ASME Code allowable limits for the repair weld, sleeve and tube are not exceeded. That is, stress intensities are bounded by the Code minimum limits for SB-163 material and the cumulative fatigue usage is less than 1.0.

Draft Regulatory Guide 1.121, "Bases for Plugging Degraded PWR Steam Generator Tubes" and ASME Code are used to develop the plugging limit of the sleeve determined by NDE, should sleeve wall degradation occur. Potentially degraded sleeves are shown (by analysis) to retain burst strength in excess of three times the normal operating pressure differential at the end of cycle conditions. No credit for the presence of the parent tube behind the sleeve is assumed when performing the minimum wall/burst evaluation.

An ultrasonic inspection will be performed of the repair weld prior to placing the tube into operation. The UT inspection is used to verify that the minimum acceptable fusion zone thickness of the weld is achieved. This minimum weld 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 operating and accident conditions. The loading cycles that were applied

to the repaired sleeve assembly analysis were those for a 40 year plant life cycle. Therefore, the fatigue analysis is conservative for an operating plant. The results of the fatigue analysis indicate acceptable usage factors for the entire range of permitted weld thicknesses.

The plugging limit for the HEJ sleeve was determined by updating the minimum wall thickness calculations using ASME Code minimum material properties and the Kewaunee plant specific steam pressure and temperature. At the time of the licensing of HEJ sleeves for Kewaunee, the accepted practice used the lower tolerance limit material properties for Alloy 690, as determined by Westinghouse. Since then, sleeve plugging limit methodology has been to use ASME Code minimum material property values for the plugging limit determination. Using conservative allowances growth (even through no degradation has been detected in domestically installed thermally treated Alloy 690 tube or sleeve material) and uncertainty, the repaired HEJ sleeve plugging limit has been determined to be 24% throughwall.

Leakage Assessment

The LWR joint uses an autogenous tube-to-sleeve weld and is inherently leaktight. Leakage testing of 3/4 inch and 7/8 inch full length laser welded sleeve assemblies under conditions considered to be more severe than expected during all operating plant conditions has shown that the laser weld does not introduce additional primary-to-secondary leakage during a postulated steam line break event.

Previously submitted data regarding the lower HEJ indicates that the lower joint will remain leak tight during all operating and faulted plant conditions. Therefore, the repair process does not introduce additional primary-to-secondary leakage during faulted conditions, and laser weld repaired HEJ sleeved tubes do not need to be considered as leakage contributors in any offsite dose analysis. Evaluation of tube sheet deflections for the Model 51 SG at KNPP operating conditions indicate that the joint will not experience a reduction in radial contact pressure due to tube sheet bow effects as can be experienced in some SGs.

Corrosion Assessment

Thermally treated Alloy 600 and Alloy 690 laser welded sleeved tube assemblies have performed well historically with regard to corrosion. Accelerated corrosion test results show that the free span laser welded joint for initially installed LWS (with post weld heat treatment) are capable of exhibiting a resistance to corrosion of greater than 10 times that of rolled tube transitions. Accelerated corrosion tests also show that non-heat treated LWS free span joints exhibit corrosion resistance equal to or greater than rolled tube transitions. These factors suggest postulated sleeve joint degradation, even in a non-heat treated condition, would occur at a

relatively slow rate, and be able to be detected by routine eddy current inspection prior to reaching any applicable safety margins. The free span 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.

Accelerated corrosion testing (doped steam) of LWR HEJ sleeved tubes was performed with bounding tube stress conditions applied through the use of a specially designed test fixture. This fixture is used to apply an axial load to the tube during the corrosion test, thereby simulating locked tube condition stresses. Corrosion test preloads were determined by using the peak installation stresses for a LWR HEJ tube for the area above the weld and including the stress input due to pressure end cap loads. Pressure end cap loads would contribute if the tubes were not locked. For Kewaunee this condition is conservative since it is felt that nearly every tube has experienced some degree of fixity at the TSPs. Corrosion samples were fabricated from a tube heat of Alloy 600 known to be susceptible to PWSCC. Both stress relieved and non-stress relieved LWR HEJ samples were used. Roll transition control samples placed in the corrosion media failed in the range of 42 to 62 hours. Of the 8 stress relieved samples, 4 showed no indication of cracking either by NDE or by destructive examination, with approzimatley 600 hours of exposure, or approximately 12 times the reported mean time-to-crack for the roll transition control samples. The remaining four showed crack indications at 2.7 to 6 times the mean time-to-crack for the roll transition control samples. All cracks were located in the upper hydraulic transition, and the crack morphology consisted of either circumferential (3 samples) or axial (1 sample) degradation. No degradation was detected in the weld regions. The non-stress relieved samples showed crack indications at 1.4 to 4 times the mean time-to-crack for the roll transition control samples. These data, in particular the results of the non-stress relieved samples indicate that the laser weld repair of degraded HEJ sleeved tubes will not result in a rapid corrosion potential, as measured against roll transition control samples and the reported times to crack of the roll transitions within the industry and at Kewaunee.

The results of the corrosion testing and analytical work, as discussed in WCAP-14685, demonstrate that production of a laser weld in the upper hardroll region will not result in parent tube stress conditions significantly greater than the traditional laser welded sleeve installation.

Integrity Assessment

The weld process to be used for the HEJ repair is identical to that used for installation of LWSs and was qualified in accordance with ASME Code Section IX and XI. The feasibility of the laser weld repair process was verified through laboratory testing with sleeve tube geometries consistent with those of the upper HEJ in 7/8 inch tubing. After welding was completed, the welds were metallographically sectioned and evaluated. The geometry of the welds were within

the requirements for a LWS application. The welds were sound with no cracking, porosity, or indications on the weld ID when subject to liquid dye penetrant inspection. Field experience indicates that the weld fusion zone width is typically twice the minimum acceptable thickness used in the stress/fatigue evaluation.

Part of the HEJ repair verification process included the performance of repair welds in HEJ mockups in a fixed condition test fixture. The fabrication of the mockups was twofold. The first was to determine the magnitude of the residual stresses imparted on the sleeve and parent tube during the installation of the HEJ sleeve under conditions of tube lockup, and the stress conditions during and follow welding and post weld stress relief. Prior to welding both the parent tube and the sleeve material were oxidized to simulate potential contaminants present in the joint region. The type of contaminants introduced was determined from analyses of HEJ sleeved tube samples removed from KNPP. The second purpose in fabricating the mockups was to provide samples for the accelerated corrosion testing program.

The residual stresses were determined through strain gages attached to the tube at locations above and below the upper HEJ. Stresses were determined at each fabrication step from formation of the HEJ through laser welded repair and post weld stress relief. All of the fabrication steps were consistent with the field process used in-situ. The stress relief step which was used, and will be employed in the field, was at a nominal temperature of 1350° F with a soak period of five minutes. The laser welded repair, including stress relief, did not result in any change in tube diameter nor the introduction of any significant tube bowing or localized tube bulging from the process.

The weld process verification testing included the use of preconditioned, or oxidized tube/sleeve lengths, which provide for a representive in-situ condition. The quality of the welds was not affected by the oxidation. The HEJ installation process included a wet bone cleaning step, which sufficiently removes all oxidation/deposits from the tube ID in the joint area. The LWR power level is within the qualified power level range for the laser welded sleeves, therefore, no formal qualification of the LWR is required per the ASME Code. The essential welding variables to be used for LWR are within the range of variables defined in Code Case N-395. Visual examination of LWRs showed the welds to be sound, with no evidence of blowholes or cracking. UT examination showed the welds to be of good quality and within the desired range of weld width at the sleeve-to-tube interface. Sectioning of repair welds verified that the minimum weld widths were consistently greater than the analysis minimum weld thickness which satisfies the ASME Code stress analysis.

Repair of Previously Plugged Indications

The inspection requirements applicable to the laser weld repair of previously plugged HEJ sleeved tube assemblies are similar to the inspection requirements for the installation of original LWS. A new baseline inspection of the entire sleeve/tube length must be performed prior to

repair to ensure that the degradation is confirmed to the upper HEJ lower hardroll transition, or lower. Tubes with HEJ indications above the proposed repair weld location will be removed from service by plugging.

The HEJ repair weld will be formed at the approximate mid-point of the upper hardroll and should not be affected by any previously identified parent tube degradation in the lower hardroll transition. If shallow degradation is present in the weld region, it will be detected during the UT inspection.

SIGNIFICANT HAZARDS DETERMINATION

This proposed change was reviewed in accordance with the provisions of 10 CFR 50.92 to show no significant hazards exists.

1) Operation of the KNPP in accordance with the proposed license amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

The laser weld repair of HEJ sleeved tubes will not affect the tube, sleeve or weld stress conditions or fatigue usage factors such that the limits of the ASME Boiler and Pressure Vessel Code are exceeded. Strain gauge testing of the laser-weld repaired HEJ sleeved tubes indicates tube far field stresses above and below the upper HEJ are similar. The magnitude of these stresses are slightly less than those associated with far field residual stresses for the two most recent domestic LWS programs. Accelerated corrosion testing of the prototypic repair welds in special fixtures designed to simulated locked tube configuration show that the expected lifetime of the repair welds exceeds the current license. Therefore, use of the laser-welded repair process will not result in an increased probability of an accident previously evaluated.

A post weld UT inspection will be required to verify minimum acceptable weld thickness to ensure that weld stresses do not exceed ASME Code limits for both stress intensity and fatigue usage. Leakage testing of LWS joints at pressure conditions far exceeding any plant normal or faulted conditions indicate the weld is leaktight at all plant conditions. Mechanical testing of 7/8 inch laser welded tubesheet sleeves installed in roll expanded tubes has shown that the individual joint structural strength of Alloy 690 laser welded sleeves under normal, upset and faulted conditions provides margin to acceptable limits. These acceptance limits bound the most limiting (3 times normal operating pressure deferential) recommended by RG 1.121.

The HEJ sleeve plugging limit is currently defined in the KNPP TSs is reduced from 31% to 24% throughwall due to the use of ASME code minimum material properly values for the sleeve material. A parent tube plugging limit of 50% continues to apply to the tube length adjacent to and above the weld. Minimum wall thickness requirements (used for developing the depth based

plugging limit for the sleeve) are determined using the guidance of RG 1.121 and the pressure stress equation of Section III of the ASME Code.

The hypothetical consequences of failure of the laser-welded repaired HEJ would be bounded by the current SG tube rupture analysis covered in the KNPP Updated Safety Analysis Report. Due to the slight reduction in diameter caused by the sleeve wall thickness, primary coolant release rates would be slightly less than assumed for the SGTR , and therefore would result in lower primary fluid mass release to the secondary system. For a postulated break location immediately above the repair weld the metal-to-metal interference fit provided by the original roll expansion would greatly reduce leak rates. Tube fixity conditions in the Kewaunee SGs at the support plate intersections are such that sufficient resistance to end cap loads are believed to be provided, therefore, axial motion of the postulated separated tube is not expected to occur, and leak rates would be expected to be well below make-up capacity.

The laser weld repair process does not change existing reactor coolant system flow conditions, therefore, existing LOCA analysis results will be unaffected. Plant response to design basis accidents for the current tube plugging and flow conditions are not affected by the repair process; no new tube diameter restrictions are introduced. Therefore, the application of the repair weld will not increase the consequences of a previously evaluated accident.

2) The proposed license amendment request does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Application of laser-welded repair of HEJ sleeved tubes will not introduce significant or adverse changes to the plant design basis. The general configuration of the HEJ sleeve is unaffected by the repair weld process. The repair process also does not represent a potential to affect any other plant component. Stress and fatigue analysis of the repair has shown that the ASME Code and Regulatory Guide 1.121 criteria are not exceeded. Application of the laser weld repair of HEJ sleeved tubes maintains overall tube bundle structural and leakage integrity at a level consistent to that of the originally supplied tubing during all plant conditions. The laser weld repair process does not provide a mechanism resulting in an accident outside of the area affected by the repair. Any hypothetical accident as a result of potential tube or sleeve degradation in the repaired portion of the joint is bounded by the existing tube rupture accident analysis. Therefore, use of the laser-welded repair process will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3) The proposed license amendment does not involve a significant reduction in the margin of safety.

The laser weld repair of the HEJ sleeved tubes has been shown to restore integrity of the tube bundle consistent with its original design basis conditions, i.e., tube/sleeve operational and faulted load stresses and cumulative fatigue usage are bounded by the ASME Code requirements

and the repaired tubes are leaktight under all plant conditions. Application of the laser-welded repair will not result in a significant reduction in the margin of safety.

ENVIRONMENTAL CONSIDERATIONS

This proposed amendment request involves a change to the inspection requirements with respect to the installation or use of a facility component located within the restricted area. WPSC has determined that the proposed amendment request involves no significant hazards consideration and no significant change in the types of effluent that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, this proposed amendment meets the eligibility criteria for categorical exclusions set forth in 10 CFR 51.22(c)(9). This proposed amendment also involves changes in recordkeeping, reporting or administrative procedures or equipment. Accordingly, with respect to these items, the amendments meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

ATTACHMENT 2

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

September 6, 1996

Proposed Amendment 144

Affected TS Pages:

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TS 4.2-6
TS 4.2-7
TS 4.2-8
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4.4.e Containment Vacuum Breaker System 4.4-7

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

b. Steam Generator Tubes

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

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

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

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

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

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

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.

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.

1. Steam Generator Sample Selection and Inspection

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

2. Steam Generator Tube Sample Selection and Inspection

The tubes selected for each in-service inspection shall:

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

   Tubes left in service as a result of application of the tube support plate plugging criteria shall be inspected by bobbin coil probe during all future REFUELING outages.

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

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

   Implementation of the steam generator tube support plate voltage-based plugging 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 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 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).

e. 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.

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

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

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

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

3. Inspection Frequencies

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

a. In-service inspections shall be performed at refueling intervals not more than 24 calendar months after the previous inspection. If two consecutive inspections following service under AVT conditions, not including the pre-service inspection, result in all inspection results falling into the C-1 category; or if two consecutive inspections demonstrate that previously observed degradation has not continued and no additional degradation has occurred, the inspection interval may be extended to a maximum of once per 40 months.

b. If the results of the in-service inspection of a steam generator conducted in accordance with Table 4.2-2 fall in Category C-3, the inspection frequency shall be increased to at least once per 20 months. The increase in inspection frequency shall apply until a subsequent inspection meets the conditions specified in 4.2.b.3.a and the interval can be extended to a 40-month period.

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

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

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

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

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

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

4. Plugging Limit Criteria

The following criteria apply independently to tube and sleeve wall degradation except as specified in TS 4.2.b.5 for the tube support plate intersections for which voltage-based plugging criteria are applied.[2]

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-14685, "Laser Weld Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant," WCAP-11643, "Kewaunee Steam Generator Sleeving Report (Mechanical Sleeves)," or CEN-413-P, "Kewaunee Steam Generator Tube Repair Using Leak Tight Sleeves."

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[2]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

b. Any Westinghouse mechanical sleeve which, upon inspection, exhibits wall degradation of 24% or more shall be plugged prior to returning the steam generator to service. Figure TS 4.2-1 illustrates the application of tube, sleeve, and tube/sleeve joint plugging limit criteria.

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

5. Tube Support Plate Voltage-Based Plugging Criteria[3]

The following criteria are used for the disposition of a steam generator tube for continued service that is experiencing 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 ≤ 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 ≤ 5.6 volts 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 > 5.6 volts will be plugged or repaired.

d. If, as a result of leakage due to a mechanism other than ODSCC at the tube support plate intersection or some other cause, an unscheduled mid-cycle inspection is performed, the following repair criteria apply instead of TS 4.2.b.5.c. If bobbin voltage is within expected limits, the indication can remain in service. The expected bobbin voltage limits are determined from the following equation:

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[3]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

$$V < \frac{\frac{\Delta t}{CL}(V_{SL} - V_{BOC}) + V_{BOC}}{1 + (.2)\left(\frac{\Delta t}{CL}\right)}$$

Where:

V = measured voltage
V_{BOC} = voltage at BOC
Δt = time period of operation to unscheduled outage
CL = cycle length (full operating cycle length where operating cycle is the time between two scheduled steam generator inspections)
V_{SL} = 9.6 volt for 7/8 inch tubes

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 actual measured end-of-cycle voltage distribution would have exceeded the leak limit (for the postulated main steam line break utilizing licensing basis assumptions) during the previous 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 the calculated conditional burst probability exceeds the threshold value, notify the NRC and provide an assessment of the safety significance of the occurrence.

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 be fusing the sleeve and tube using a laser welding process at the approximate midpoint of the upper HEJ hardroll region, effectively isolating the degradation below the weld. The laser weld repair is performed identical 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, consistent with the process described in WCAP-14685[5].

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.

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[6]

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.

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[5]WCAP 14685, "Laser Weld Repair of HEJ Sleeves for Kewaunee Nuclear Power Plant," September 1996 (Proprietary).

[6]The tube support plate voltage-based repair criteria is applicable for the 1995 to 1996 operating cycle only.

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).

ATTACHMENT 3

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

September 6, 1996

Proposed Amendment 144

AFFIDAVIT
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WCAP-14686