ATTACHMENT 1

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To

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Proposed Amendment 155

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INTRODUCTION

Wisconsin Public Service Corporation (WPSC) is submitting this proposed Technical Specification (TS) amendment to redefine the parent tube pressure boundary for Westinghouse mechanical hybrid expansion joint (HEJ) steam generator (SG) tube sleeves. The existing TS, illustrated on Figure TS 4.2-1, defines the parent tube pressure boundary for the upper HEJ sleeve joint as the hardroll lower transition (HRLT). The HRLT on Figure TS 4.2-1 refers to TS 4.2.b.4.b which discusses the disposition of parent tube indications (PTIs) within the HRLT based on the amount of interference lip between the hardroll expansion and the elevation of the PTI. The existing TS allows PTIs located within the HRLT to remain in service if there is a minimum diameter difference, or interference lip, of 0.003 inch (not including non-destructive examination (NDE) uncertainty).

The proposed amendment will change the parent tube pressure boundary from a minimum required interference lip to a minimum required length of non-degraded hardroll engagement. The proposed parent tube pressure boundary is based on analytical evaluations and testing of representative samples. The results show that tubes with PTIs located 0.92 inch or more below the bottom of the hardroll upper transition (HRUT), as measured on the inside of the sleeve, have sufficient structural and leakage integrity and do not compromise the safety of the SG tube bundle. Parent tube indications located 0.92 inch (plus an allowance for NDE uncertainty) or more below the bottom of the HRUT will be allowed to remain in service. The bottom of the HRUT is more specifically defined in the safety evaluation as well as in Attachment 5.

BACKGROUND INFORMATION

The Kewaunee Nuclear Power Plant (KNPP) has two Westinghouse Model 51 SGs with 7/8" OD tubing. The KNPP SGs have experienced 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 have 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 and 1997 with the Combustion Engineering welded sleeves. As a result of these sleeving programs a total of 2195 HEJs and 440 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 rotating probe developed for sleeve inspections. A total of 77 circumferential crack-like parent tube indications were detected. Sixty-six of the indications were within the parent tube pressure boundary that was defined by TS at that time. The 1994 TS defined the parent tube pressure boundary as the entire upper joint including both the hydraulic and hardroll expansions. The tubes with the PTIs were removed from service by plugging. The remaining 11 indications were below the upper joint parent tube pressure boundary. In 1995, the upper HEJs were examined with the rotating +Point probe. During this inspection 753 PTIs were detected; 657 in the TS defined parent tube pressure boundary and 92 below the parent tube pressure boundary. The 657 PTIs located within the TS defined parent tube pressure boundary were removed from service by plugging. Three specimens were removed from SG B for testing and destructive evaluation.

When the PTIs were first detected in 1994, analysis and testing of representative samples were performed to characterize the effects of the degradation on the structural and leakage integrity of the upper joint. WPSC presented this information to the NRC staff in April of 1994 as a basis to revise the TS parent tube pressure boundary. The NRC staff responded that there was insufficient time to review a TS change within the schedule constraints of the 1994 refueling outage. Therefore, WPSC did not submit an amendment request to the NRC.

On October 6, 1995, WPSC submitted an amendment request to revise the location of the HEJ parent tube pressure boundary (reference 1). The proposed parent tube pressure boundary was based on analytical evaluations, the testing of representative samples, and the results of the destructive examinations and tests of the HEJ specimens removed from the KNPP SGs. The culmination of these tests and evaluations showed that tubes with PTIs located 1.1 inches or further below the bottom of the HRUT had sufficient structural and leakage integrity and would not compromise the safety of the SG tube bundle. The technical basis is documented in WCAP-14446, "Repair Boundary for Parent Tube Indications Within the Upper Joint Zone of the Hybrid Expansion Joint (HEJ) Sleeved Tubes," dated August 1995.

During a meeting in January of 1996, the NRC staff indicated their preference to have a parent tube pressure boundary acceptance criterion based on the amount of interference lip remaining in the HRLT. This preference was predicated on uncertainties in the hardroll length. The discussion between the NRC and WPSC focused on providing a TS criterion based on a change in diameter between the maximum point of the hardroll and the elevation of the PTI. Subsequent to this meeting, additional work was performed by both Westinghouse and Zetec to develop this criterion as well as the ability to measure it. Westinghouse performed additional structural testing to determine the amount of interference lip necessary to satisfy the requirements of Regulatory Guide (RG) 1.121. Based on these results, it was proposed that the parent tube pressure boundary be changed to allow PTIs with a minimum diameter change of 0.003 inch (not including an allowance for measurement uncertainty) to remain in service (reference 4). For leakage considerations, the total number of PTIs remaining in service was to be limited such that the primary-to-secondary leakage from a postulated steam line break (SLB) event would not exceed a small fraction of the 10 CFR 100 guidelines. The results of this structural testing were documented in WCAP-14641, "HEJ Sleeved Tube Structural Integrity Criteria: 'ΔD' Diametral Interference at PTIs," dated April 1996. The NRC approved these changes to the KNPP TS on September 25, 1996.

On September 21, 1996, the KNPP shut down for a refueling and SG tube inspection outage. As a part of the planned outage work scope, plugs were removed from 550 previously plugged HEJ sleeved tubes. All of the inservice and unplugged HEJs were inspected using the rotating +Point probe. Applying the minimum diameter difference criterion, only 127 HEJ sleeved tubes were acceptable out of a total of 2044 HEJ sleeved tubes inspected. Based on the number of tubes affected, KNPP elected to perform a laser welded repair (LWR) on all tubes with PTIs. Due to the limited success of the LWR effort, KNPP then implemented the resleeving process. The resleeving repairs were ultimately successful and the KNPP was returned to service in June 1997.

During October 1996, seven HEJ sleeved tubes with circumferential PTIs in the HRLT were removed from the KNPP SGs for examination at Westinghouse. Examinations included lab eddy current measurements for comparison to field eddy current measurements, leak tests, and tensile tests. Destructive examination results from HEJ joints that did not meet the ΔD criterion showed that the HEJs continued to provide adequate structural and leakage integrity.

After the 1996 outage, further analyses, evaluations, and tests were completed and have shown that a specific length of hardroll engagement relative to the bottom of the HRUT will provide adequate structural and leakage integrity. During the Fall of 1997 and the Winter of 1997/1998, testing was performed to expand the database of leak rate and burst tests of the HEJ sleeved tubes. The tests concluded there was sufficient load capacity and leak resistance if the PTI is located greater than or equal to a minimum distance of non-degraded hardroll engagement as measured from the bottom of the HRUT (i.e., the presence of a lip is not necessary). This distance was determined to be 0.92 inch as measured on the inside of the sleeve. The information from the Fall 1997 testing was presented to the NRC on December 9, 1997 (reference 2). On April 9, 1998, WPSC and the NRC met again to discuss the Winter 1997/1998 testing and the evaluation of the data (reference 3).

This proposed change in the parent tube pressure boundary takes its technical basis from the tests and evaluations presented to the NRC during these two meetings. The technical basis is documented in Attachment 4, WCAP-15050, "HEJ Sleeved Tube Length Based Degradation Acceptance Criterion," dated May 1998. NRC staff questions from the April 9 meeting are addressed in Section 8 of WCAP-15050. The NDE inspection technique and associated uncertainties are described in Attachment 5.

Provided below is a description of the proposed TS changes, a safety evaluation, a 10 CFR 50.92 significant hazards determination, and an environmental considerations statement. Attachments 2 and 3 contain the affected TS pages.

DESCRIPTION OF PROPOSED CHANGE

This proposed amendment request will modify KNPP TS Section 4.2 to redefine the parent tube pressure boundary in the upper HEJ sleeve and incorporate inspection requirements for HEJs left in service by the new criterion. The proposed changes are as follows:

1) The definition of "repaired tube" in Section 4.2.b is being modified to reference the repair methods used. The definition will now reference the repairs described in TS 4.2.b.4.a.

2) Section 4.2.b.2, "Steam Generator Tube Sample Selection and Inspection," will add a new inspection criterion specifically for the upper joint of the Westinghouse HEJ sleeved tubes. The new inspection criterion is TS 4.2.b.2.f and is inserted between current TS 4.2.b.2.e and TS 4.2.b.2.f. The new inspection criterion will require all Westinghouse mechanical HEJ sleeves allowed to remain in service by the new parent tube pressure boundary criterion of TS 4.2.b.4.c to be inspected during each inservice inspection outage in the upper HEJ joint region.

3) TS 4.2.b.4.b will be separated into two specifications. The first specification, TS 4.2.b.4.b, will describe the requirement for Westinghouse mechanical HEJs that exhibit sleeve wall degradation. This specification is for the entire sleeve. The second specification, TS 4.2.b.4.c, will outline the requirements for dispositioning parent tube indications in the upper joint of Westinghouse HEJs. This second specification is described in more detail in 4) below.

4) New TS 4.2.b.4.c will be added to describe the requirements for dispositioning parent tube indications in Westinghouse HEJ sleeved tubes. This specification, formerly part of TS 4.2.b.4.b, for disposition of parent tube indications is being revised from a minimum diameter difference criterion to a criterion which requires a minimum engagement length of non-degraded hardroll. The text of the new specification will be as follows:

   For disposition of parent tube indications in the upper joint of Westinghouse HEJ sleeved tubes, as depicted in Figure TS 4.2-1, the following requirements will apply:

   a) HEJ sleeved tubes shall be inspected with a non-destructive examination technique capable of locating the bottom of the hardroll upper transition. HEJ sleeved tubes with circumferential parent tube indications located greater than or equal to 0.92 inch (plus an allowance for NDE uncertainty) below the bottom of the hardroll upper transition, as measured on the inside of the sleeve, may remain in service.

   b) HEJ sleeved tubes with circumferential parent tube indications located less than 0.92 inch (plus an allowance for NDE uncertainty) from the bottom of the hardroll upper transition, as measured on the inside of the sleeve, shall be plugged or repaired prior to returning the steam generator to service.

c) HEJ sleeved tubes with axial parent tube indications located in the parent tube pressure boundary, as depicted in Figure TS 4.2-1, shall be plugged or repaired prior to returning the steam generator to service.

5) Figure TS 4.2-1 will be revised to depict the bottom of the hardroll upper transition, to revise the TS referenced at the bottom of the figure, and to define the sleeve and parent tube pressure boundaries more clearly.

6) Notes 1 and 2 of Table TS 4.2-2 will be updated to reference the correct TS.

The basis for TS Section 4.2 will be modified along with the above changes to discuss the revised parent tube pressure boundary and to reference WCAP-15050 (Attachment 4) and the NDE technique and qualification (Attachment 5) reports which contain the technical basis for this change. Administrative changes throughout Section 4.2 and its bases are also being made as part of this proposed amendment request.

SAFETY EVALUATION

The evaluation of circumferential parent tube indications in HEJ sleeved tubes supports the conclusions that tube structural integrity consistent with RG 1.121 is provided. Additionally, off-site doses are maintained within a small fraction of the 10 CFR 100 guidelines and control room doses are maintained within GDC-19 criteria in the event of a postulated SLB. In support of this proposed TS amendment request, the following areas will be summarized in this safety evaluation:

- HEJ sleeve design and terminology,
- a discussion of the HEJ sleeved tubes removed from the KNPP SGs,
- a discussion of sleeve/tube joint structural integrity issues,
- a discussion of the 1997/1998 structural testing results,
- a discussion of sleeve/tube joint leakage integrity issues,
- a discussion of the 1997/1998 leak rate testing results, and
- defense-in-depth considerations.

HEJ Sleeve Design and Terminology

The Westinghouse HEJ sleeve assembly extends entirely through the tubesheet and is attached to the tube by first performing a simultaneous hydraulic expansion of the sleeve into the tube at the sleeve ends. A mechanical roll expansion, i.e., hardroll, is then produced at the lower end of the sleeve followed by a mechanical roll expansion within the upper hydraulically expanded region.

The HEJ sleeves installed at KNPP are 36 inches, 30 inches, or 27 inches depending on their location within the SG.

Starting from the upper end of the sleeve and moving downwards, the regions of the upper HEJ sleeve-to-tube joint assembly are as follows (refer to figure 1, page 16):

(1) The non-expanded safe end, approximately 0.5 inch long,
(2) The upper transition from the non-expanded to the hydraulically expanded region (HEUT),
(3) The upper hydraulically expanded region, approximately 1 inch long,
(4) The expansion transition from the hydraulically expanded region to the upper hardroll joint (HRUT), approximately 0.25 inch long,
(5) The upper hardroll (HR) joint, approximately 1 inch long,
(6) The expansion transition from the hardroll joint to the hydraulically expanded region (HRLT), approximately 0.25 inch long without roll down, or approximately 0.5 inch with roll down,
(7) The lower hydraulically expanded region, approximately 1.5 inches long, and
(8) The lower transition from the hydraulically expanded to the non-expanded portion of the sleeve (HELT).

Discussion of the HEJ Sleeved Tubes Removed from the KNPP SGs

In April of 1995, three HEJ sleeved tube samples were removed from SG B. Field eddy current results identified a 300 to 360 degree circumferential parent tube indication in the HRLT in one tube (R2C32) and 360 degree circumferential indications in the remaining two tubes (R2C54 and R2C61). Tube R2C54 was leak tested with no leakage detected up to SLB pressures. The joints in tubes R2C32 and R2C54 were both tensile tested. Maximum loads of 10359 lbf and 10700 lbf were achieved (respectively). The end cap load associated with three times the design differential pressure at KNPP is 2450 lbf. The third specimen was archived. No Alloy 690 sleeve degradation was identified in these specimens.

Seven HEJ sleeved tubes with circumferential PTIs in the upper portion of the HRLT were removed from the KNPP SGs during October of 1996. Two of the tubes were removed from SG A (R2C21 and R2C25) and five from SG B (R2C21, R2C51, R2C58, R2C65, and R2C69). None of these tubes met the minimum diameter difference criterion used for disposition of PTIs during the inspections performed in October 1996. Non-destructive and destructive examinations were performed on these tubes at Westinghouse. All tubes exhibited low leak rates and high strengths. In addition, no Alloy 690 sleeve degradation was identified.

Three of the seven pulled HEJ specimens were leak tested at differential pressures representative of normal operation and steam line break conditions. Two specimens were tested at room temperature while the remaining was tested at an elevated temperature. Only two of the specimens leaked and both exhibited very low leakage rates at SLB conditions. The tube tested at elevated temperatures experienced a SLB leak rate of 3.5×10^{-5} gpm. The other tube was tested at room temperature and had a SLB leak rate of 1.45×10^{-6} gpm.

Four of the specimens were tensile tested. The objective of these tests was to determine the load required to slide the portion of the tube above the circumferential indications over the sleeve. It was anticipated that following fracture of the parent tube in the HRLT, significant loads would be required to pull the fractured portion over the approximate 1.0 inch HR region of the sleeve. These tests were conducted at 600°F since elevated temperatures are more representative of SG conditions during normal operation. As suspected, high pull loads were required to fracture the tubes. Maximum sliding loads occurred in the first 0.5 inch of travel and ranged from 4250 lbf to 5400 lbf. (Note: The end cap load associated with three times the differential pressure for KNPP is approximately 2450 lbf.) These tests led to concept, and eventually qualification, testing aimed directly at determining the leak resistance and strength of HEJ joints without the presence of an interference lip between the tube and sleeve.

Discussion of the Sleeve/Tube Joint Structural Integrity Issues

In accordance with RG 1.121, tube rupture should be precluded at the pressure loading equal to three times normal operating pressure differential, or 1.43 times the SLB pressure differential, depending on which is more limiting. For KNPP, the most limiting RG 1.121 loading is the three times normal operating pressure differential (4800 psi as compared to 3660 psi for 1.43 times the SLB pressure). Tube rupture is normally thought of as a double-ended guillotine break of a SG tube, or a failure of a tube involving localized burst with ductile tearing of the tube material at the edges of the crack. In the case of a sleeved tube with postulated circumferential degradation in the parent tube at the HEJ HRLT, tube rupture will be considered as an uncontrolled release of reactor coolant exceeding the normal makeup capacity (consistent with NUREG-0844). The tube will not experience failure or burst as a result of internal pressurization resulting in tube failure with crack tearing, but from separation of the severed tube end from the sleeve by axial displacement of the postulated circumferentially separated parent tube.

For the configuration of a HEJ sleeved tube with a postulated 100% throughwall, 360 degree circumferential PTI in the HEJ HRLT, tube rupture can only occur if the tube is displaced axially by a distance of approximately 3 to 3.25 inches. Tube displacement of this magnitude would result in a complete separation between the severed tube end from the sleeve resulting in a sufficient flow area such that a release rate exceeding makeup capacity would be realized. Testing performed with specimens and HEJ samples removed from KNPP support the conclusion that indications will retain structural integrity such that significant tube slippage and axial displacement will not be expected. As a defense in depth backup to these data, analysis of SG dimensions, fit up, and manufacturing practices was performed in 1995 to determine the lengths over which a postulated circumferentially separated tube could become displaced. This axial displacement can only occur if the hardroll interaction friction force and packed TSP crevice friction forces are neglected. Therefore, the assessment was conservative. The postulated levels of tube displacement were calculated for plant operation during: 1) normal operation, 2) a postulated SLB and 3) at pressure differential loadings equal to the RG 1.121 safety limits. The joint was able to resist the potential loadings described above.

A number of structural tests were performed to determine the structural characteristics of a degraded HEJ joint. Since the TSs consider sleeved tubes no different than non-sleeved tubes, and considering that RG 1.121 supplies structural integrity recommendations independent of sleeved or non-sleeved tubes, the applicability of redefined parent tube pressure boundary criterion can be established using current NRC documents and regulations.

Discussion of 1997/1998 Structural Testing Results

Since PTIs were first detected in April of 1994, many series of tests have been performed on HEJ sleeved tube specimens. Tests performed in 1996 showed that HEJs with PTIs in the lower hardroll transition would provide structural integrity with a minimum measured amount of interference lip. The results of this testing were submitted to the NRC and provided the basis for the current TS that defines the HEJ parent tube pressure boundary. However, due to the addition of measurement uncertainties applied in the field, less tubes were retained inservice than anticipated when the criterion was applied. This led to additional testing to determine if the interference lip was necessary for structural integrity.

Concept tests were performed in late 1997 to establish whether an HEJ sleeve with a minimum length of non-degraded hardroll engagement (i.e., without an interference lip), could provide adequate structural and leak integrity. This testing attempted to simulate the increase in interface pressure between the tube/sleeve joint resulting from:

1) the difference in coefficients of thermal expansion between the tube and sleeve materials,
2) the result of the temperature gradient across the sleeve/tube interface and its effect on thermal expansion, and
3) the force due to the pressure on the inside of the sleeve.

Each of the above had been simulated singly during previous testing. It should be noted that the concept test data were not taken for the purpose of quantifying the strength or leak tightness of the sleeve/tube joints. The results were intended to confirm or deny the expectation of joint strength.

Ten specimens were pressurized to 1600, 3000, and 4800 psi for the concept testing. The pressure test results of the specimen tubes, which were severed at the top of the HRLT, demonstrated significant strength and low leak rates. It was apparent that joint strength could meet the RG requirements. Leak rates at the SLB differential pressure were small. The concept test data was presented at the December 1997 meeting between the NRC and WPSC. At the close of the

December meeting, plans for additional qualification testing to support a new length criterion for definition of the upper HEJ parent tube pressure boundary were presented.

In parallel with the concept testing, field data was reviewed to establish the length of hardroll and the potential changes in hardroll length as a result of rolldown. A sample of 500 tubes were reviewed and analysis determined that the length of hardroll flat is independent of any rolldown which may occur during the installation process. This information was also presented at the December 1997 meeting to resolve the NRC staff concern related to uncertainties in hardroll length. The average length of hardroll inside the sleeve was found to be 0.98 inch.

During the winter of 1997/1998, 40 qualification test specimens were made. Attention was paid to the production process in order to obtain specimens that were similar to the HEJ sleeved tubes that are currently inservice at KNPP. Eddy current exams were completed to locate the HRUT and the PTI. Test flaws were then machined 100 percent throughwall and 360 degrees around the tube. Data was taken from a total of 25 no lip specimens that were tested at elevated temperatures. This data was analyzed for pressure failure and leak rate. Of the remaining 15 specimens, five were tested at ambient conditions, two were destroyed during testing, and eight were archived.

The qualification testing protocol called for pressurizing the specimens to 1600, 2560, and 4800 psi. Deflection and leak rate were measured at both the 1600 and 2560 psi hold points. Each hold point was maintained for a period of two minutes to obtain a steady-state leak rate and temperature measurement. Failure pressure of each specimen was also recorded. The test setup is described in detail in the presentation materials from the April 9, 1998 meeting between WPSC and the NRC (reference 3). The test protocol and the results are described in detail in WCAP-15050 (Attachment 4).

The purpose of the data analysis of WCAP-15050 was to quantify potential acceptance criteria for HEJs with PTIs in the hardroll lower transition. The test results demonstrate that an HEJ sleeved tube with a 360 degree circumferential, 100 percent deep PTI located as low as 0.92 inch or greater below the bottom of the HRUT, as measured from the inside of the sleeve, would meet the structural guidelines of RG 1.121. The data was evaluated using three different approaches which are described below. Each approach was then compared to determine if all supported the continued operation of sleeve tubes with PTIs located greater than or equal to 0.92 inches from the bottom of the HRUT. The results of the statistical analysis are described in detail in the supporting WCAP-15050.

The first approach to analyzing the test data used all of the valid, raw failure data (22 specimens). By keeping the data unfiltered, some data which could be expected to be non-conservative or suspect was used. The as-measured failure pressures of all these specimens averaged 5571 psi with a standard deviation of 486 psi. The mean for the first approach of analyzing the data meets the three times differential pressure criterion from the RG.

The second approach filtered the failure test data to include only specimens which had a sleeve temperature less than or equal to 600°F and a sleeve/tube temperature gradient of less than or equal to 100°F. All of these specimens (15) had a sleeve temperature lower than the normal operating and SLB temperature and a smaller temperature gradient than the normal operating and SLB temperature gradients. Therefore, the failure pressure results for this set of data are conservative with regard to normal operating and SLB failure pressures because increasing the primary side temperature or the temperature gradient increases the interface pressure. The mean failure pressure of the 15 specimens was 5611 psi with a standard deviation of 461 psi. The mean again meets the three times the differential pressure criterion of 4800 psi. These results are quite similar to those from using all the raw data.

The third approach adjusted all of the applicable specimens' (22) failure pressures to correspond to the sleeve and tube conditions at normal operating and SLB temperatures. The mean failure pressure of the adjusted data at normal operating temperatures was 7851 psi with a standard deviation of 2206 psi. For SLB temperature conditions, the mean failure pressure was 9673 psi with a standard deviation of 2721 psi. The mean again meets the criterion of 4800 psi.

The adjusted failure pressure was also reviewed for the effect of precycling the specimens. Thirteen of the 22 applicable specimens were precycled (i.e., brought to 1600 psi then depressurized), prior to being pressurized to 1600, 2560, and 4800 psi for data collection. It is noted that precycling has a significant effect on the failure pressure. From data tables in WCAP-15050, it can be seen that the expected failure pressures increase from 6668 (no precycling) to 8670 (precycled) psi for normal operating thermal conditions. The standard deviation decreases slightly from 2682 to 2336 psi. At SLB conditions, the failure pressures for specimens without and with precycling are 8214 and 10684 psi with standard deviations of 2682 and 2336 psi. Since the sleeved tubes in the KNPP SGs have been precycled by plant operation, the precycled test results should be preferred for the failure pressure evaluation using the adjusted data.

All failure pressure data analyzed by the three approaches meet the three times the normal operating differential pressure acceptance criterion. The first makes use of all raw data, whether or not it is conservative. The second analyzes only the data which would be known to be conservative relative to the normal and faulted SG conditions. The third attempts to adjust failure pressures to correspond to the normal and faulted SG conditions. The second and third approaches assume that there is a correlation between the strength of the joint, the temperature of the sleeve and tube, and the thermal gradient across the joint. The performance of the joint should improve due to these factors and their effect on the radial interference between the tube and sleeve. An increase in radial interference would be brought about by increasing the temperature or increasing the thermal gradient across the joint. By adjusting the data to the normal operating or faulted conditions, the increase in interference was simulated (by calculation). The second approach uses only data that corresponds to conditions of less interference than during normal or faulted operation. Therefore, the second approach is preferred because of its conservatism, but it should be noted that the results from all three of the analysis evaluations support the continued operation of sleeved tubes with PTIs located 0.92 inch (plus an allowance for uncertainty) or below the bottom of the HRUT as measured on the inside of the sleeve.

A range of lengths from 0.90 to 1.01 inches below the bottom of the HRUT, as measured on the inside of the sleeve, were tested as part of the qualification tests. All of the qualification tests yielded acceptable results with regard to the guidelines of RG 1.121. Additionally, there appears to be no dependence of failure pressure on the length to the PTI relative to the bottom of the HRUT over the tested range. In other words, the joints with indications at elevations above the estimated upper end of the HRLT, greater than or equal to 0.90 inch below the bottom of the HRUT, exhibit failure pressures commensurate with PTIs at other relatively close elevations. Although the test data suggests that PTIs as near as 0.90 inch from the bottom of the HRUT provide adequate structural integrity, a value of 0.92 inch was conservatively selected for determining the minimum acceptable physical length of non-degraded hardroll.

The overall conclusion from the analysis of the test results is that a sleeved tube with a 360 degree, 100 percent throughwall PTI located 0.92 inch or greater from the bottom of the HRUT will meet the requirements of RG 1.121. This length (0.92 inch) does not include an allowance for NDE uncertainty as is measured on the inside of the sleeve. The allowance for NDE examination uncertainty is discussed in Attachment 5, "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes With Parent Tube Indications."

Discussion of Sleeve/Tube Joint Leakage Issues

Leakage from the sleeve/tube joint must be small such that normal operating leakage is kept below the current TS allowable limits for normal operation. The TS limit for normal operation is 150 gpd or 0.104 gpm. For SLB conditions, off-site doses are to be maintained within a small fraction of 10 CFR 100 guidelines and control room doses are to be maintained within GDC-19 criteria. The most current calculated KNPP maximum allowable primary-to-secondary leakage limit during a SLB is 12.85 gpm. By maintaining the primary-to-secondary leakage during a SLB below this limit, 10 CFR 100 and GDC-19 criteria are met. The 12.85 gpm for maximum allowable primary-to-secondary leak rate is currently under review by the NRC. The proposed license amendment for the 12.85 gpm leak rate limit was submitted on April 6, 1998.

Discussion of 1997/1998 Leak Rate Test Results

As described earlier, ten specimens were pressurized to 1600, 3000, and 4800 psi for the concept testing. The test data of the specimen tubes, which were severed at the top of the HRLT, demonstrated significant strength and low leak rates. Leak rates at the 3000 psi differential pressure ranged from 0.004 to 0.032 gpm with a mean value of 0.022 gpm. These values were consistent with results from previous representative specimen testing and pulled tube testing. The concept test data was presented at the December 1997 meeting between the NRC and WPSC. At the close of the December meeting, plans for additional qualification testing to support a new length criterion for definition of the upper HEJ pressure boundary were presented.

There were 25 results from the qualification testing for the evaluation of leakage. Of those, only 20 leak rate values were recorded at a differential pressure of 1600 psi. This occurred during the first days of testing when the leak rate at the postulated SLB differential pressure was of most interest. Data from all 25 specimens were available at 2560 psi. The average leak rate at a differential pressure of 1600 psi (20 specimens) was 0.012 gpm with a standard deviation of 0.013 gpm. The average leak rate at a differential pressure of 2560 psi from the 25 specimens was 0.013 gpm with a standard deviation of 0.012 gpm. The leak rate distributions are essentially the same at both pressure differentials.

The leak rates were also selectively evaluated using the second approach considered for analyzing the failure pressure data. For estimating the leak rate at 1600 psi, specimens with sleeve temperatures of ≤ 600°F and a sleeve-to-tube temperature gradient of ≤ 100°F were selected. The average leak rate from these 13 specimens was 0.014 gpm with a standard deviation of 0.015 gpm. For the SLB leak rate, data were selected that had a sleeve temperature of ≤ 600°F, regardless of the thermal gradient. The mean leak rate from the 16 specimens meeting the criteria was 0.013 gpm with a standard deviation of 0.009 gpm. The individual ratios of leak rate at SLB to normal operating were also examined and found to range from 0.233 to 3.90. Statistical analysis shows that the maximum ratio expected at a 99 percent confidence level is 9.3. Therefore, the leak rate during a SLB event would be expected to be less than one order of magnitude greater than the leak rate during normal operation. Normal operating primary-to-secondary leakage is limited to 0.104 gpm (150 gpd). Therefore, the maximum primary-to-secondary leak rate during a SLB is conservatively assumed to be one gpm (0.104 x 9.3). This is below the allowable primary-to-secondary leak rate allowed at KNPP during SLB event. As stated earlier, the most recent calculation limits primary-to-secondary leakage to 12.85 gpm during a SLB. This limit is used as the bounding leakage when estimating leakage from all sources during a SLB. When this estimation occurs, one gpm will be added to encompass all HEJs left in service by the revised parent tube pressure boundary.

Primary-to-secondary leak rate history for the KNPP was presented to the NRC staff and is documented in reference 3. It should be noted that at the end of September 1996, when the plant was about to shut down for a refueling outage, the cumulative primary-to-secondary leak rate for normal plant operation was two (Argon 41 measurement) to five (tritium measurement) gpd. During the refueling outage it was determined that approximately 1500 HEJ sleeved tubes with PTIs within the parent tube pressure boundary had been inservice. Converting these actual leak rates to 0.0014 and 0.0035 gpm respectively, the leak rate from the 1500 sleeved tubes was less than might be expected from the lower regime of the test data for a single, 360 degree, throughwall, sleeved tube PTI. This is consistent with the observed leak rate from one of the pulled HEJ sleeved tubes from KNPP.

Safety Evaluation Conclusion

The overall conclusion from the analysis of the failure pressure and leak rate tests is that a sleeved tube with a 360 degree, 100 percent throughwall PTI located 0.92 inch or greater from the bottom of the HRUT will meet the structural requirements of RG 1.121. Furthermore the operational limit of 150 gpd and the assignment of one gpm to the allowable primary-to-secondary leak rate during a SLB will ensure off-site doses do not exceed the small fraction of 10 CFR 100 and will ensure control room doses remain within GDC-19 criteria. Therefore, implementation of a length criterion with an adjustment for NDE uncertainty does not jeopardize public health or safety.

Defense-In-Depth Considerations

1) The location of PTIs from the pulled tube data is predominately below the top of the HRLT. Therefore, in most cases a lip is present. The strength of such tubes has been demonstrated in prior documents and by the results from testing of the pulled tube specimens.

2) The range of test data showed that PTIs located 0.90 inch or greater below the bottom of the HRUT met the structural requirements of RG 1.121. However, 0.92 inch was conservatively selected for determining the physical length measurement.

3) The leak rate test results show that leak rates at normal operating and SLB conditions are essentially the same. However, statistical analysis shows that for a 99 percent confidence level, the ratio of leak rate at SLB to normal operating is 9.3. To bound the SLB leak rate, the assumption is made that the SLB leak rate is one order of magnitude greater than the normal operating leak rate. Therefore, at SLB conditions, the bounding leak rate would be expected to be approximately one gpm. This leak rate will be included in the calculation for estimating primary-to-secondary leakage from all sources.

4) For the configuration of a HEJ sleeved tube with a postulated 100% throughwall, 360 degree circumferential PTI in the HEJ HRLT, tube rupture can only occur if the tube is displaced axially by a distance of approximately 3 to 3.25 inches. Tube displacement of this magnitude would result in a complete dissociation between the separated tube end and sleeve resulting in a sufficient flow area such that a release rate exceeding makeup capacity would be realized. Testing performed with specimens and HEJ samples removed from KNPP support the conclusion that indications will retain structural integrity such that tube slippage and axial displacement will not be expected.

SIGNIFICANT HAZARDS DETERMINATION

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

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.

   Mechanical testing shows inherent structural integrity of the HEJ upper joint such that the requirements of RG 1.121 are met even for 360 degree, 100 percent throughwall PTIs. Structural test results are documented in WCAP-15050. Based on the test data, the structural recommendations of RG 1.121 are satisfied when there is a minimum length of non-degraded hardroll which measures 0.92 inch (plus an allowance for NDE measurement uncertainty) or more from the bottom of the HRUT, as measured on the inside of the sleeve. Based on the structural integrity of the HEJ upper joint, it can be concluded that application of the revised parent tube pressure boundary will not result in a significant increase in the probability of an accident previously evaluated.

   A conservatively bounding primary-to-secondary SLB leak rate of one gpm will be applied to the calculation for postulated SLB leakage. This leak rate encompasses all HEJs left inservice with PTIs located outside the revised parent tube pressure boundary. This one gpm is based on a normal operating leakage limit of 150 gpd. This leak rate is based on tests and analysis documented in WCAP-15050. Application of this leak rate to the postulated leakage calculation will ensure primary-to-secondary leakage will not exceed the current maximum allowable during a SLB event. Maintenance of the current maximum allowable primary-to-secondary leak rate during a SLB event ensures off-site doses will not exceed a small fraction of 10 CFR 100 and control room doses will not exceed GDC-19 criteria. Therefore, it can be concluded that the application of the revised parent tube pressure boundary will not increase the consequences of an accident previously evaluated.

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

   Implementation of the revised parent tube pressure boundary will not introduce a change to the design basis or operation of the plant. The configuration of the currently installed sleeves is not physically changed. As with the initial installation of the sleeves and previous changes to the parent tube pressure boundary for HEJs, implementation of the revised parent tube pressure boundary does not interact with other portions of the reactor coolant system. Neither the sleeve design nor the implementation of the revised parent tube pressure boundary affects any other component or location of the tube outside of the immediate repaired area. Mechanical testing of representative specimens supports the conclusions that the joint retains structural integrity consistent with RG 1.121 and leakage integrity with regards to 10 CFR 100 and GDC-19. Any hypothetical accident as a result of potential PTIs is bounded by the existing tube rupture analysis. Therefore, application of the revised parent tube pressure boundary will not create the possibility of a new or different kind of accident from any previously evaluated.

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

   The safety factors used in establishment of the HEJ sleeved tube pressure boundary are consistent with safety factors in the ASME Boiler and Pressure Vessel Code used in the SG design. Based on the sleeve-to-tube geometry, it is unrealistic to consider that application of the revised parent tube pressure boundary could result in single tube leak rates exceeding the normal makeup capacity during normal operating conditions. The parent tube pressure boundary developed in WCAP-15050 has been developed using the methodology of RG 1.121. The performance characteristics of postulated degraded parent of HEJ sleeve/tube joints have been verified through testing to retain structural integrity and preclude significant leakage during both normal operating and SLB conditions. The existing off-site and control room dose evaluation performed for KNPP established a faulted loop primary-to-secondary leak rate of 12.85 gpm. Combined leakage from all sources including the assumed leak rate for the voltage based repair criteria and for HEJs with PTIs that are left inservice will not exceed 12.85 gpm in the faulted loop. Maintenance of this limit will ensure off-site doses will not exceed a small fraction of the 10 CFR 100 guidelines nor will it exceed the GDC-19 criteria for control room dose. Therefore, the application of the revised parent tube pressure boundary 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 involves no significant hazards consideration and no significant change in the types of effluent that may be released off-site 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 record keeping, reporting, or administrative procedures or requirements. Accordingly, with respect to these items, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with this proposed amendment.

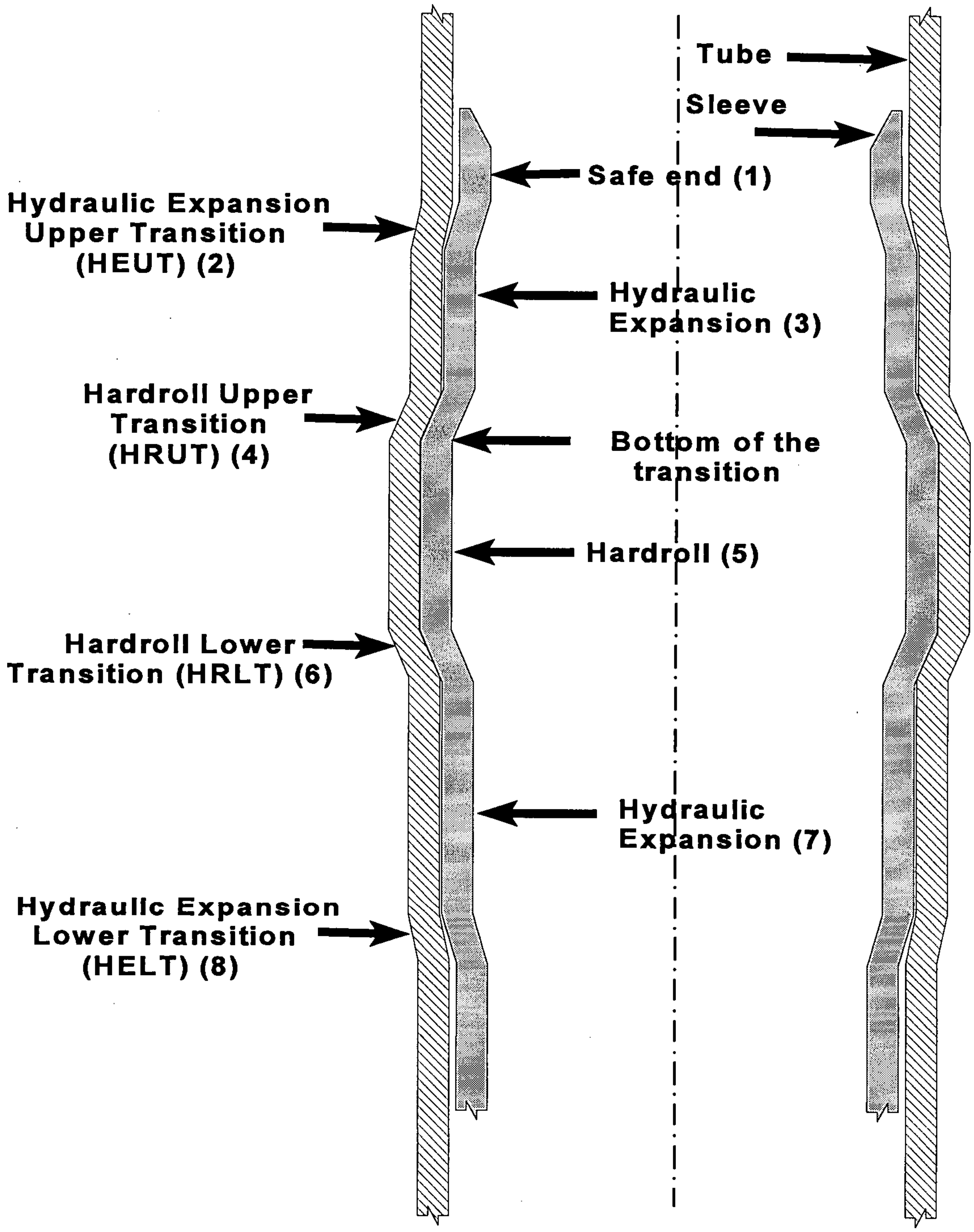

Figure 1 - HEJ Sleeve Terminology (correspouds to numbered text, pg 6)

ATTACHMENT 2

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

May 14, 1998

PROPOSED TS AMENDMENT NO. 155

Strikeout TS Pages:

TS 4.2-2
TS 4.2-4
TS 4.2-5
TS 4.2-6
TS 4.2-7
TS 4.2-8
TS 4.2-10
TS B4.2-3
TS B4.2-4
TS B4.2-5
TS B4.2-6
TS B4.2-7
Table TS 4.2-2, page 1 of 1
Figure TS 4.2-1

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. 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 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 ~~with~~in 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.

6. F* and EF* Tubesheet Crevice Region Plugging Criteria

The following criteria are to be used for disposition or repair of steam generator tubes experiencing degradation in the tubesheet crevice region.

a. Tubes with indications of degradation within the roll expanded region below the midpoint of the tubesheet may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.12" (plus an allowance for NDE uncertainty). This criteria is called the F* criteria and applies to the factory roll expansion, or to additional roll expansions formed as an extension of the original roll. Any degradation existing below the F* (plus an allowance for NDE uncertainty) is acceptable for continued service.

b. Indications of degradation not repairable by 4.2.b.6.a may be repaired using the EF* criteria. The EF* region is located a minimum of 4" below the top of the tubesheet, and is formed by an additional roll expansion of the tube in the originally unexpanded length. Tubes with indications of degradation within the EF* region may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.44" (plus an allowance for NDE uncertainty). Any degradation existing below EF* (including uncertainty) is acceptable for continued service.

7. 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. This report shall include the tubes for which the F* or EF* criteria were applied.

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.

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 ΔD determination. Application of the pressure boundary for HEJ sleeved tubes provides allowance for leakage in a faulted loop during a postulated steam line break (SLB) event. A SLB leakage of 0.025 gpm is assumed for each applicable indication.~~ Steam line break leakage from all sources must be calculated to be ~~$\leq$ 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.

[(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.

~~(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 [illegible]2-2
STEAM GENERATOR NON-REPAIRED TUBE INSPECTION

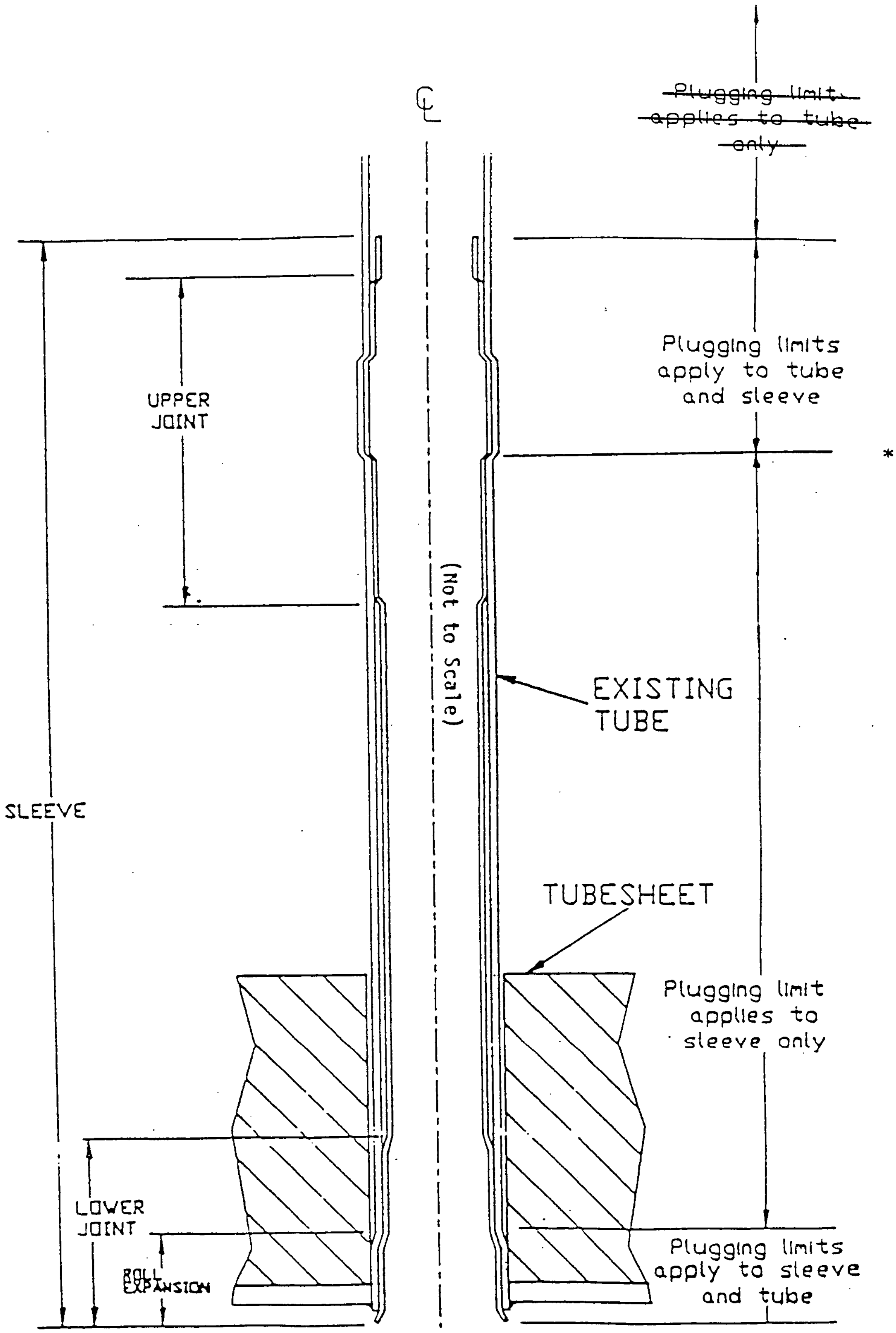
<table>
<tr><th colspan="3">1ST SAMPLE INSPECTION</th><th colspan="2">2ND SAMPLE INSPECTION</th><th colspan="2">3RD SAMPLE INSPECTION</th></tr>
<tr><th>Sample Size</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th><th>Result</th><th>Action Required</th></tr>
<tr><td rowspan="9">A minimum of S Tubes per S.G.</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="5">C-2</td><td rowspan="5">Plug or repair defective tubes and inspect additional 2S tubes in this S.G. (2)</td><td>C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-2</td><td rowspan="3">Plug or repair defective tubes and inspect additional 4S tubes in this S.G. (2)</td><td>C-1</td><td>None</td></tr>
<tr><td>C-2</td><td>Plug or repair defective tubes</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td></tr>
<tr><td>C-3</td><td>Perform action for C-3 result of first sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td rowspan="3">C-3</td><td rowspan="3">Inspect all tubes in this S.G., (2) plug or repair defective tubes and inspect 2S tubes in the other S.G. (2)

Prompt notification of the Commission. (1)</td><td>The other S.G.'s are C-1</td><td>None</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Some S.G.'s C-2 but no additional S.G. are C-3</td><td>Perform action for C-2 result of second sample</td><td>N/A</td><td>N/A</td></tr>
<tr><td>Additional S.G. is C-3</td><td>Inspect all tubes in each S.G. and plug or repair defective tubes. Prompt notification of the Commission. (1) (2)</td><td>N/A</td><td>N/A</td></tr>
</table>

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

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

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



* Refer to TS ~~4.2.b.4.b~~

Application of Plugging Limit for a Westinghouse Mechanical Sleeve

Amendment No. ~~128~~
~~09/25/96~~

Figure TS 4.2-1

ATTACHMENT 3

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

May 14, 1998

<u>Proposed Amendment 155</u>
Affected TS Pages:

TS 4.2-2
TS 4.2-4
TS 4.2-5
TS 4.2-6
TS 4.2-7
TS 4.2-8
TS 4.2-10
TS B4.2-3
TS B4.2-4
TS B4.2-5
TS B4.2-6
TS B4.2-7
TS B4.2-8
Table TS 4.2-2, page 1 of 1
Figure TS 4.2-1

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 by tube repair methods described in TS 4.2.b.4.a 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 TS 4.2.b.2.a and TS 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 TS 4.2.b.2.a through TS 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 TS 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 TS 4.2.b.2.a through TS 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 TS 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. In addition to the sample required in TS 4.2.b.2.a through TS 4.2.b.2.c, all Westinghouse mechanical hybrid expansion joint sleeves allowed to remain in service by the length criterion of TS 4.2.b.4.c shall be inspected during each in-service inspection in the upper joint region as depicted in Figure TS 4.2-1.

g. 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.

h. 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 TS 4.2-2. For repaired tubes, actions shall be taken as described in Table TS 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 TS 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 TS 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 TS 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. Figure TS 4.2-1 depicts a Westinghouse HEJ sleeve.

c. For disposition of parent tube indications in the upper joint of Westinghouse HEJ sleeved tubes, as depicted in Figure TS 4.2-1, the following requirements will apply:

1. HEJ sleeved tubes shall be inspected with a non-destructive examination technique capable of locating the bottom of the hardroll upper transition. HEJ sleeved tubes with circumferential parent tube indications located $\geq$ 0.92 inch (plus an allowance for NDE uncertainty) below the bottom of the hardroll upper transition, as measured on the inside of the sleeve, may remain in service.

2. HEJ sleeved tubes with circumferential parent tube indications located < 0.92 inch (plus an allowance for NDE uncertainty) from the bottom of the hardroll upper transition, as measured on the inside of the sleeve, shall be plugged or repaired prior to returning the steam generator to service.

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

d. 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.

e. 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 ≤ 2.0 volts will be allowed to remain in service.

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

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

6. F* and EF* Tubesheet Crevice Region Plugging Criteria

The following criteria are to be used for disposition or repair of steam generator tubes experiencing degradation in the tubesheet crevice region.

a. Tubes with indications of degradation within the roll expanded region below the midpoint of the tubesheet may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.12" (plus an allowance for NDE uncertainty). This criteria is called the F* criteria and applies to the factory roll expansion, or to additional roll expansions formed as an extension of the original roll. Any degradation existing below the F* (plus an allowance for NDE uncertainty) is acceptable for continued service.

b. Indications of degradation not repairable by TS 4.2.b.6.a may be repaired using the EF* criteria. The EF* region is located a minimum of 4" below the top of the tubesheet, and is formed by an additional roll expansion of the tube in the originally unexpanded length. Tubes with indications of degradation within the EF* region may remain in service provided the distance from the bottom of the uppermost roll transition to the tip of the crack is greater than 1.44" (plus an allowance for NDE uncertainty). Any degradation existing below EF* (including uncertainty) is acceptable for continued service.

7. 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. This report shall include the tubes for which the F* or EF* criteria were applied.

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.

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-15050.[5] The pressure boundary described in the WCAP will allow PTIs located in the upper joint to remain in service if there is a minimum non-degraded (i.e., no detectable degradation in the parent tube) hardroll length of 0.92 inch (plus an allowance for NDE uncertainty) as measured from the bottom of the hardroll upper transition. The minimum hardroll engagement length is derived from structural and leakage testing. During field application, the PTI is located in reference to the bottom of the hardroll upper transition to ensure the minimum length of non-degraded hardroll exits. The inspection is performed using eddy current techniques capable of profiling and flaw detection as described in "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications."[6] The NDE uncertainty for this criterion is a function of the eddy current probe and technique used. The uncertainty has been calculated to be 0.023 inch. However, for field application, an eddy current uncertainty of 0.03 inch will be applied to the minimum hardroll engagement length of 0.92 inch.

Leakage testing performed for the HEJ pressure boundary showed that leak rates for normal operating and steam line break (SLB) are comparable. However, statistical analysis shows that for a 99 percent confidence level the ratio of leak rate at SLB to normal operating is 9.3.[5] To bound SLB leak rate, the assumption is made that SLB leak rate is one order of magnitude greater than normal operating leak rate. The normal operating primary-to-secondary leakage limit is 0.104 gpm (150 gpd per TS 3.1.d.2). Therefore, the maximum primary-to-secondary leak rate during a SLB is assumed to be approximately 1 gpm (9.3 x 0.104 gpm). The 1 gpm will be the assigned leakage encompassing the HEJs left in service using the length criterion described in the paragraph above. Steam line break leakage in the faulted loop from all sources must be calculated to be less than or equal to the maximum allowable leakage described in the Basis for TS 3.4.d. Maintenance of the maximum allowable leak rate limit ensures off-site doses will remain within a small fraction of the 10 CFR Part 100 guidelines and ensures control room doses will not exceed GDC-19 during a SLB.

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[5] WCAP-15050, "HEJ Sleeved Tube Length Based Degradation Acceptance Criterion," May 1998.

[6] "NDE Technique to Determine Length Measurements in HEJ Sleeved Tubes with Parent Tube Indications," Attachment 5 to Letter to Document Control Desk from C.R. Steinhardt dated May 14, 1998.

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[(7)] and WCAP-14685, Revision 2, Addendum 1.[(8)]

Topical CEN-629-P[(9)] 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[(10)] 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.

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

[(8)]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).

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

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

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

The hydraulic equivalency ratios for the application of normal operating, upset, and accident condition bounding analyses have been evaluated. Design, installation, testing, and inspection of steam generator tube sleeves requires substantially more engineering than plugging, as the tube remains in service. Because of this, the NRC has defined steam generator tube repair to be an Unreviewed Safety Question as described in 10 CFR 50.59(a)(2). As such, other tube repair methods will be submitted under 10 CFR 50.90; and in accordance with 10 CFR 50.91 and 92, the Commission will review the method, issue a significant hazards determination, and amend the facility license accordingly. A 90-day time frame for NRC review and approval is expected.

Technical Specification 4.2.b.5

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

The voltage-based repair limits of TS 4.2.b.5 implement the guidance in Generic Letter 95-05 and are applicable only to Westinghouse-designed steam generators with outside diameter stress corrosion cracking (ODSCC) located at the tube-to-tube support plate intersections. The voltage-based repair limits are not applicable to other forms of tube degradation nor are they applicable to ODSCC that occurs at other locations within the steam generators. Additionally, the repair criteria apply only to indications where the degradation mechanism is predominantly axial ODSCC with no indications extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of TS 4.2.b.5 requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this surveillance).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation, at the 95 percent prediction interval curve reduced to account for the lower 95/95 percent tolerance bound for tubing material properties at 650°F (i.e., the 95 percent LTL curve). The voltage structural limit must be adjusted downward to account for potential flaw growth during an operating interval and to account for NDE uncertainty. The upper voltage repair limit, V_{URL}, is determined from the structural voltage limit by applying the following equation:

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

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

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

Technical Specification 4.2.b.6

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.[11] 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.

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[11] 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).

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.

TABLE [illegible]2-2

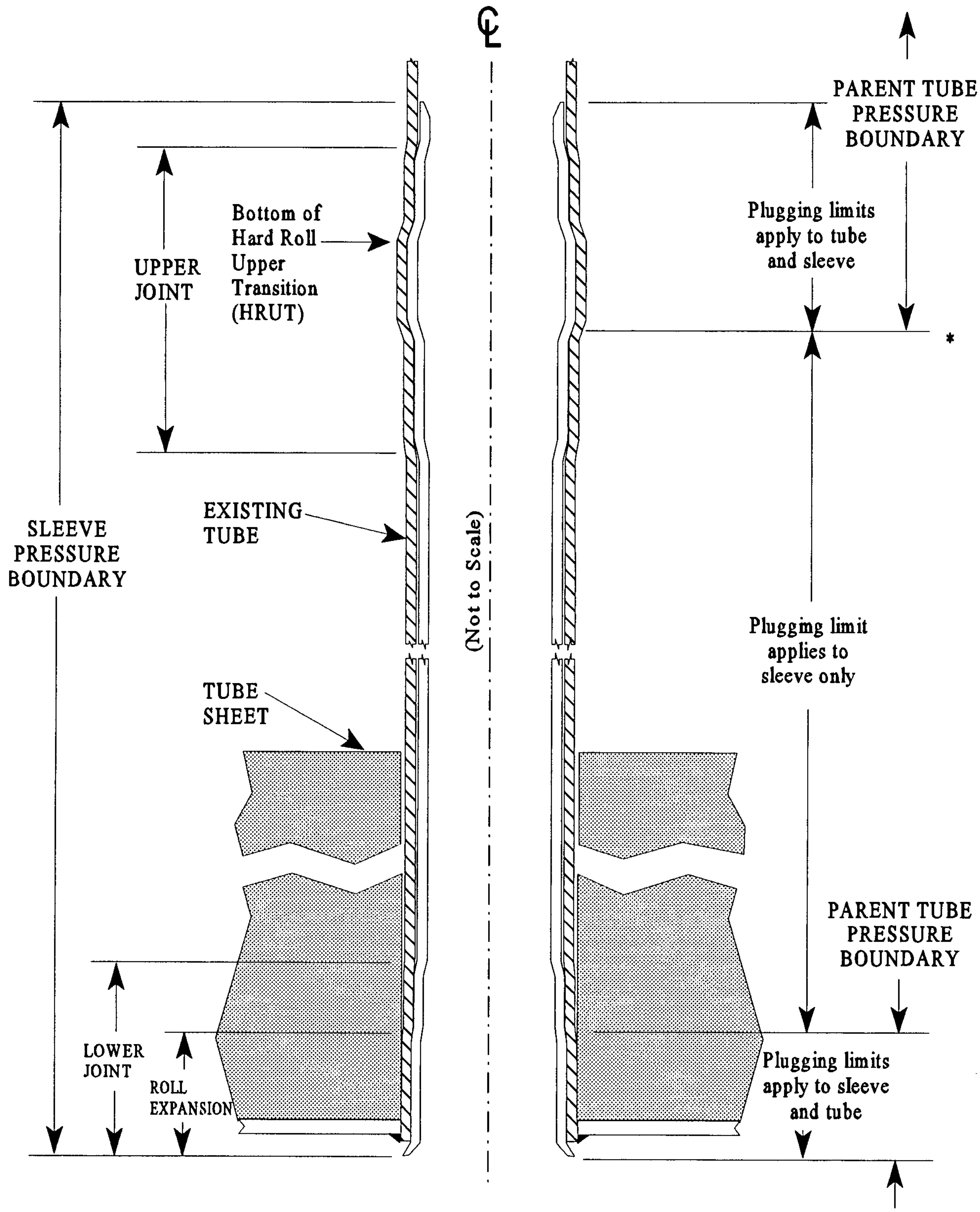
STEAM GENERATOR NON-REPAIRED TUBE INSPECTION

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

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

Notes:

1. Refer to Specification 4.2.b.7.c
2. As allowed by TS 4.2.b.2.g, the second and third sample inspections during each inservice inspection may be less than the full length of each tube by concentrating the inspection on those areas of the tube sheet array and on those portions of the tubes where tubes with imperfections were previously found.



* Refer to TS 4.2.b.4.c

Application of Plugging Limit for a Westinghouse Mechanical Sleeve

Figure TS 4.2-1

ATTACHMENT 4

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

May 14, 1998

WCAP-15050, "HEJ Sleeved Tube Length Based Degradation Acceptance Criterion," Dated May 1998

ATTACHMENT 5

Letter from C. R. Steinhardt (WPSC)

To

Document Control Desk (NRC)

Dated

May 14, 1998

NDE Technique To Determine Length Measurements in HEJ Sleeved Tubes With Parent Tube Indications

NDE Technique To Determine Leugth Measuremeuts in HEJ Sleeved Tubes With Parent Tube Iudications

Technique Description

The proposed Technical Specification (TS) amendment request demonstrates that tubes with parent tube indications (PTIs) located 0.92 inches or farther below the bottom of the hardroll upper transition (HRUT) have sufficient structural and leakage integrity, and do not compromise the safety of the steam generator (SG) tube bundle (not including NDE uncertainty). The reference location for determination of the parent tube pressure boundary, i.e., bottom of the HRUT, was selected based on the ability to measure the elevation of the PTIs using existing eddy current probe technology.

For the upcoming refueling outage, all of the upper hybrid expansion joints (HEJ) will be inspected for the detection of PTIs and determination of the PTI location relative to the HRUT. For determining the relative PTI elevation, two eddy current coils, a +Point coil and a bobbin coil, on one probe head, will be used.

The combination +Point/bobbin probe uses the bobbin coil to profile the internal (ID) surface of the sleeve and locate the bottom of the HRUT, and the +Point coil to detect and locate the PTI. The distance between the two coils will be fixed during the probe manufacturing such that at the Technical Specification (TS) defined distance, including NDE uncertainty, the bobbin coil locates the bottom of the HRUT concurrent with the +Point detection of the PTI. Figure 1 depicts the coil spacing in relation to the HEJ geometry. Fixing the coil spacing at the TS defined distance, including NDE uncertainty, during manufacture results in a "Go-No-Go" type of examination. If the PTI is detected after the bottom of the HRUT is detected, as the probe is pushed through the upper HEJ, then the parent tube pressure boundary definition is not satisfied and the tube will be plugged or repaired. If the PTI is detected before, or concurrent with, the detection of the bottom of the HRUT, then the distance is greater than the parent tube pressure boundary definition and the tube is acceptable for continued operation.

By using the +Point/bobbin combination probe the process of defining the relative distance between the PTI and lower tangent point of the HRUT is not dependent on setting scales and referencing a distant landmark. Therefore, the uncertainties associated with setting scales are eliminated. Slewing of data is also not required for implementation of this technique, therefore uncertainties associated with probe translation speeds are eliminated. The surface riding ability of the +Point is enhanced by employing a gimbaled delivery assembly. Monitoring of probe wear will follow normal industry practices associated with probe recalibration requirements and frequency. Normal field practice for this inspection will involve an HEJ standard with a selected

EDM notch. The initial amplitude measurement of the EDM notch will be compared to amplitude measurements obtained during subsequent measurements made at each calibration group change, at each probe change, or after 4 hours of service for a given probe. Probe changes will be required whenever the amplitude measurements differ by more than 15%, and all PTIs reported since the last successful probe wear check will be re-examined with a new, acceptable probe. Therefore, the only variables affecting the precision with which the PTIs can be located relative to the bottom of the HRUT are the spacing between the +Point and bobbin coils, PTI detection and location using the +Point coil, and locating the lower tangent point of the HRUT using the bobbin coil.

Distance Between Coils

The distance between the bobbin coil and the +Point coil will be fixed during probe manufacture such that at the TS defined distance, including NDE uncertainty, the bobbin coil locates the bottom of the HRUT concurrent with the +Point detection of the PTI. Coil spacing will initially be determined by drawing an adjustable combination probe (Figure 2) through a standard containing an approximate 70% OD circumferential EDM notch. The notch will be located 0.95 inches below the bottom of a fabricated ID reference (0.92 inches plus 0.03 inches for NDE uncertainty, which is discussed later). The spacing between the EDM notch and the fabricated ID reference can be controlled to within 0.005 inches in accordance with Zetec manufacturing tolerances. The coil spacing will be adjusted until the EDM notch signal response from the +Point coil and the fabricated ID reference response from the bobbin coil occur simultaneously. Once the coil spacing is determined, the probes will be manufactured in accordance with the Zetec quality assurance program. All of the combination probes manufactured for the Kewaunee inspection will be verified against the standard described above to ensure proper coil spacing prior to leaving the factory.

PTI Detection and Location Determination

The +Point portion of the combination probe is used for PTI detection and location determination. The +Point to be utilized is specifically designed for sleeve examinations and has been previously qualified in accordance with Appendix H of the EPRI PWR Steam Generator Examination Guidelines for the detection of PTIs in Westinghouse HEJ sleeves.

The PTIs found in the Kewaunee HEJs to date have been predominantly (>99%) circumferential in nature. The flaw location of interest for these indications is defined as the center of the +Point coil's field, or the middle scan line, as opposed to the peak signal response. A phenomena known as rolldown, exhibited in some of the Kewaunee HEJ sleeved tubes, may result in non-conservatism in reporting PTI locations using the peak signal response.

The HEJ tubes are installed using a tapered roller assembly. In order to remove the rolling tool from the installed sleeve, the direction of rolling is reversed to release the rollers from contact with the ID surface of the sleeve. If the rollers do not immediately retract, additional rolling in the downward direction occurs, resulting in an elongation of the hardroll lower transition, which is referred to as rolldown. The rolldown phenomena is depicted in Figure 3 and a non-rolldown condition is depicted in Figure 4.

For sleeves with rolldown, the middle scan line from the +Point coil's field accurately corresponds to the peak amplitude of the PTI. For sleeves with no rolldown, the peak amplitude of the PTI is not at the center of the +Point coil's field. This condition results from the geometry difference between the +Point coil's gimbaled shoe and the non-rolldown hardroll lower transition (HRLT) geometry, as illustrated in Figure 5. As a result, liftoff will occur as the +Point coil rides upwards from the HRLT into the hardroll. If the PTI is located in the upper half of the transition, the signal amplitude decreases as liftoff occurs, and the signal peak amplitude occurs prior to the center scan line response. This liftoff will result in a nonconservative reporting of the PTI location. For this reason, peak amplitude signal response will not be employed for PTI location determination. Rather, the center of the +Point coil's field, or the middle scan line, will be employed for PTI location determination. This method for PTI location determination is independent of rolldown and non-rolldown geometries.

Location of the center of the +Point coil's field will be determined from the +Point coil's strip chart and lissajous response, where the analyst will locate the beginning and end of the signal response and note the associated data points, as shown in Figure 6. The left hand graphic illustrates the location of the beginning of the PTI, with the associated data point 99550. The right hand graphic illustrates the location of the end of the PTI, with the associated data point 96710. The data point corresponding to center of the +Point coil's signal response is then calculated, as shown on the bottom graphic in Figure 6, as data point 98130.

Use of this method is dependent on an accurate determination of the center of the +Point coil's signal response. In order to quantify the analyst variability in selecting this point, a study has been conducted in which ten Zetec eddy current analysts determined flaw locations from ten HEJ samples used for the structural and leak rate qualification tests. The Zetec analysts are Level IIA and Level III production analysts and are Qualified Data Analysts in accordance with Appendix G of the EPRI PWR Steam Generator Examination Guidelines. Each analyst reported the flaw locations as a data point corresponding to the center line of the +Point signal response, using the technique described above. Table 1 reports the results. Using the sample rate of 1280 samples/second and probe translation speed of 0.15 inch/second used in collection of the data, the number of samples (i.e., data points) per inch is calculated to be 8533.33 samples/inch. From Table 1 the deviation between analysts in reporting flaw locations for the ten samples analyzed ranged from 75 to 157 data points which, using the number of samples per inch, corresponds to 0.008 inches to 0.018 inches in axial elevation (it is interesting to note that one revolution of the probe (at 500 rpm) equates to 0.018 inches in axial distance).

HRUT Lower Tangent Point Determination

The lower tangent point of the HRUT will be determined using the bobbin coil and Eddynet Bobbin Profile software. The bobbin coil gives an accurate profile of the ID surface of the sleeve and the software gives a graphic interface and measurement capability. Since the bobbin coils are contained in the same probe body as the +Point coil, the bobbin coils will be rotating along with the +Point coil. Sinusoidal signal response from the bobbin coils due to rotation of the probe have not been observed at Kewaunee (implementation of the ΔD criteria during the 1996 refueling outage resulted in the use of a similar combination probe (containing two bobbin coils spaced equidistant from the +Point coils) for over 1500 HEJ sleeved tubes in the Kewaunee steam generators). If large amplitude sinusoidal bobbin coil signal responses are reported in future inspections, the probe will be replaced and the affected tubes reexamined. Typical bobbin coil signal response is depicted in Figures 3 and 4.

When selecting the lower tangent point of the HRUT, the eddy current field spread is not considered. Eddy current will show the tangent point prior to the coil reaching the tangent point because the look ahead of the eddy current field "sees" the tangent point prior to the coil passing over it. As a result, the determination of the HRUT tangent point is conservative.

To eliminate analyst uncertainty in selecting the HRUT tangent point, a computer algorithm within the Eddynet Bobbin Profile software will be employed which will automatically select the HRUT tangent point. Use of this algorithm will ensure consistent and repeatable selection of the HRUT tangent. This computer alogorithm will be employed in determining the coil spacing during probe manufacture.

The methodology in computer selection of the HRUT tangent is as follows. For each HEJ examined, the analyst will select a point near the center of the hard roll flat. The algorithm will determine the average diameter over a specified range (± 300 data points, which corresponds to approximately 0.25 inches above and below the selected point). The last point which crosses this average diameter before moving up the HRUT is assigned the lower tangent point.

Bobbin coil profiling will be done at 600 kHz in the absolute mode. As part of the qualification for the combination probe used for the ΔD pressure boundary criteria, it was shown that 600 kHz frequency was unaffected by conductive deposits potentially existing on the tube OD in the area of the upper HEJ.

Overall NDE Uncertainty

Overall NDE uncertainty in utilizing the combination probe includes 0.005 inches for manufacturing tolerances in fabricating the specimen used to set coil spacing, and 0.018 inches for analyst variability in locating the center of the flaw response using the +Point coil. Therefore, the total NDE uncertainty to be added to the structural limit of 0.92 inches, is 0.023 inches. For conservatism, this will be rounded to 0.03 inches. The uncertainty value of 0.03 inches will be added to the structural limit of 0.92 inches, 0.95 inches total, and will be the value used to set coil spacing during probe manufacture.

Go/No-Go Examination

Once the spacing between the bobbin coil and the +Point coil is established to coincide with the Technical Specification (TS) defined distance, including NDE uncertainty, examination of HEJ tubes becomes a Go-No-Go examination. That is, if the PTI is detected before, or concurrent with the detection of the bottom of the HRUT as the probe is pushed through the upper HEJ, then the distance is greater than the pressure boundary definition and the tube is acceptable for continued operation. An illustration of an acceptable condition is shown in Figure 7. The left strip chart shows the +Point response while the right strip chart shows a portion of the sleeve ID profile. As can be seen from Figure 6, the center of the +Point coil field occurs prior to the detection of the bottom of the HRUT (i.e., below the cursor position), which indicates that the PTI is located outside the parent tube pressure boundary (i.e., ≥0.95 inches from the HRUT, including NDE uncertainty) and, therefore, acceptable for continued service.

Alternatively, if the PTI is detected after the bottom of the HRUT is detected as the probe is pushed through the upper HEJ, then the pressure boundary definition is not satisfied and the tube will be plugged or repaired. An illustration of an acceptable condition is shown in Figure 7. As can be seen from Figure 8, the center of the +Point coil field occurs after the detection of the bottom of the HRUT (i.e., above the cursor position), which indicates that the PTI is located within the parent tube pressure boundary. This condition will require plugging or additional repair.

During the eddy current examination, quantification of the distance from the HRUT (i.e., 0.92", 0.98", 1.05", etc.) will not be performed. This technique simply indicates whether a PTI is within the TS defined parent tube pressure boundary or not. For PTIs located outside the parent tube pressure boundary (i.e. ≥0.92 inches from the HRUT, not including NDE uncertainty), a three letter code will be assigned to track these PTIs for future inspections as proposed by TS 4.2.b.2.f.

Analyst Training/Qualification and Field Analysis Protocol

It has been the practice at Kewaunee since 1985 to have two independent analysis teams complete the analysis of all plant eddy current data. The two teams continue to be from different inservice inspection organizations in order to maintain independence. For the analysis of HEJ sleeve tubes utilizing the +Point/bobbin combination probe, this practice will be maintained.

During the 1998 refueling outage Kewaunee will inspect all of the upper HEJ joints using the +Point/bobbin combination probe, using independent analysis teams, for the detection and location of PTIs relative to the HRUT. Analysis results from the two teams will be compared, and any discrepancies between the primary and secondary analysts will be resolved by the Level III shift lead analysts from both the primary and secondary analysis teams. At a minimum, discrepancy conditions will include the following:

1. One analysis team reports a PTI and the other does not.

2. One analysis team reports a PTI within the parent tube pressure boundary (i.e., ≥0.95 inches, including NDE uncertainty); other analysis team reports the same PTI outside the parent tube pressure boundary.

3. Discrepancies in tube location.

The analysts using the +Point/bobbin combination probe will be qualified as part of the Kewaunee site specific performance demonstration program. Detailed procedures and testing governing the analysis of combination probe data will be prepared by Kewaunee and Zetec prior to field implementation to ensure the analysts are familiar with the analysis process and can consistently detect and locate the PTI.

Table 1
Analyst Variability in Flaw Location Determination

Tube	Analyst 1	Analyst 2	Analyst 3	Analyst 4	Analyst 5	Analyst 6	Analyst 7	Analyst 8	Analyst 9	Analyst 10	Average	Low	High	Dev
K-98-25	105405	105487	105440	105405	105473	105433	105435	105443	105368	105436	105436	105368	105487	119
K-98-35	95908	95979	95977	95832	95827	95904	95920	95981	95875	95831	95906	95827	95981	154
K-98-9	96924	96919	96994	96928	96993	96997	96973	96918	96997	96921	96951	96918	96997	79
K-98-11	105309	105310	105306	105307	105283	105272	105271	105310	105240	105235	105295	105235	105310	75
K-98-21	105123	105052	105087	105048	105092	105084	105093	105050	105052	105123	105086	105048	105123	75
K-98-16	98136	98055	98209	98066	98163	98210	98113	98053	98136	98135	98136	98053	98210	157
K-98-7	102462	102460	102531	102468	102567	102535	102503	102458	102460	102536	102486	102458	102567	109
K-98-22	100837	100837	100909	100803	100885	100838	100840	100838	100803	100838	100838	100803	100909	106
K-98-14	101338	101264	101337	101224	101231	101340	101272	101264	101264	101263	101264	101224	101340	116
K-98-30	96655	96657	96582	96581	96538	96656	96648	96649	96655	96654	96652	96538	96657	119

Figure 1

Coil Spacing in Relation to HEJ Geometry

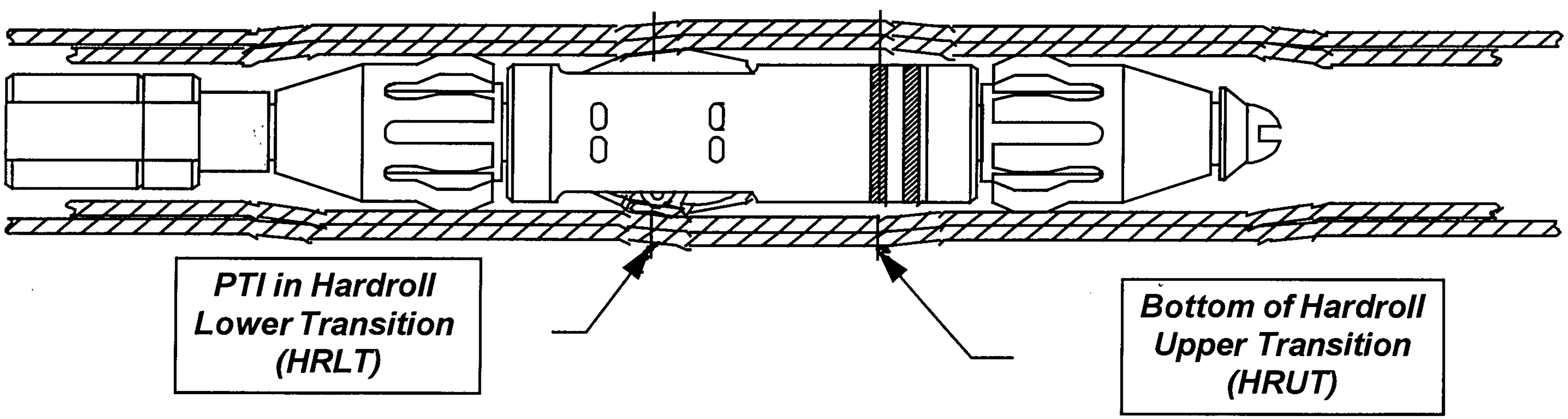
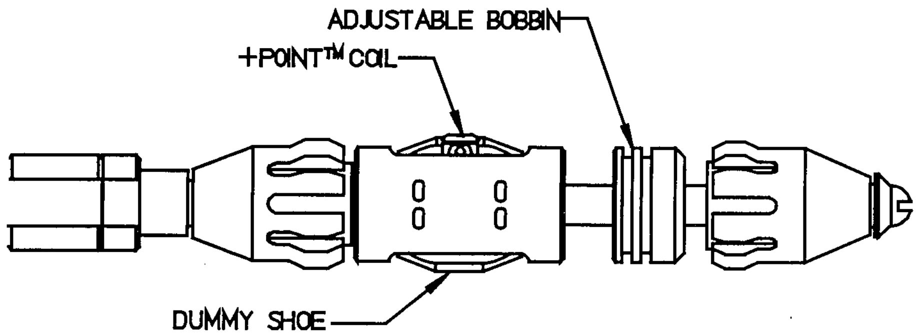
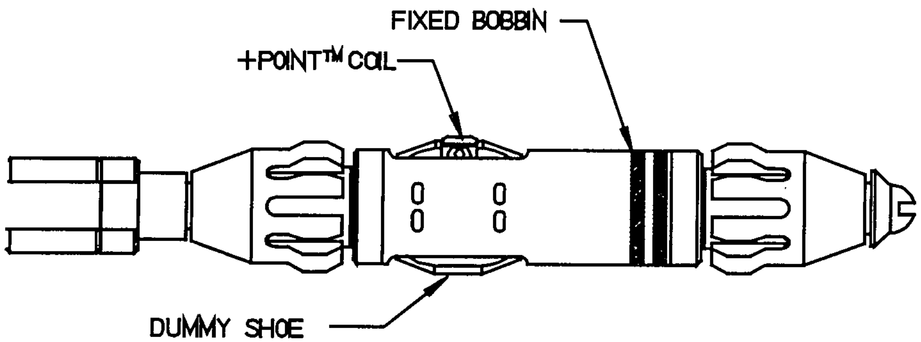


Figure 2



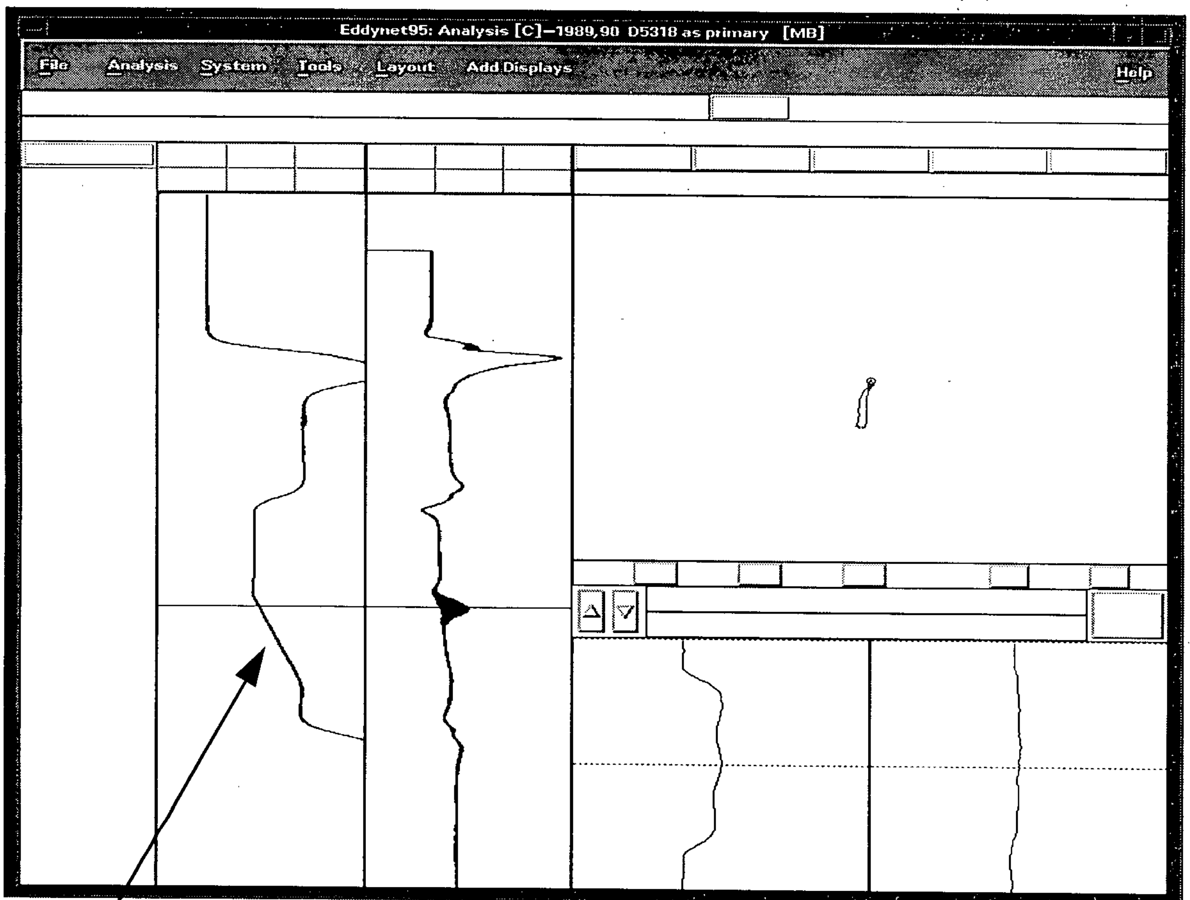
Prototype Combination Probe



Production Combination Probe

Figure 3

Rolldown Geometry

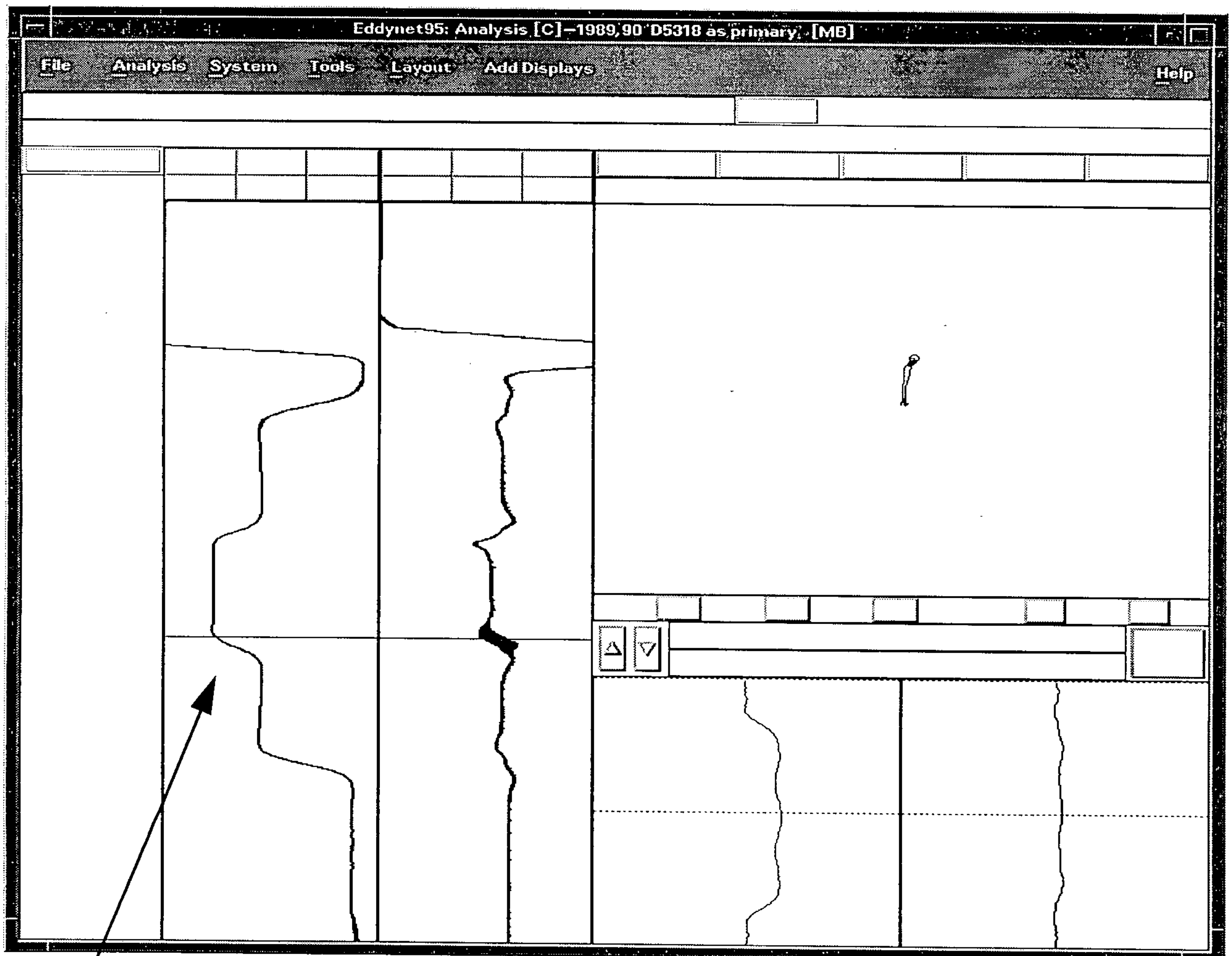


Bobbin Profile +Point Response

Elongation of HRLT referred to as rolldown

Figure 4

Non Rolldown Geometry

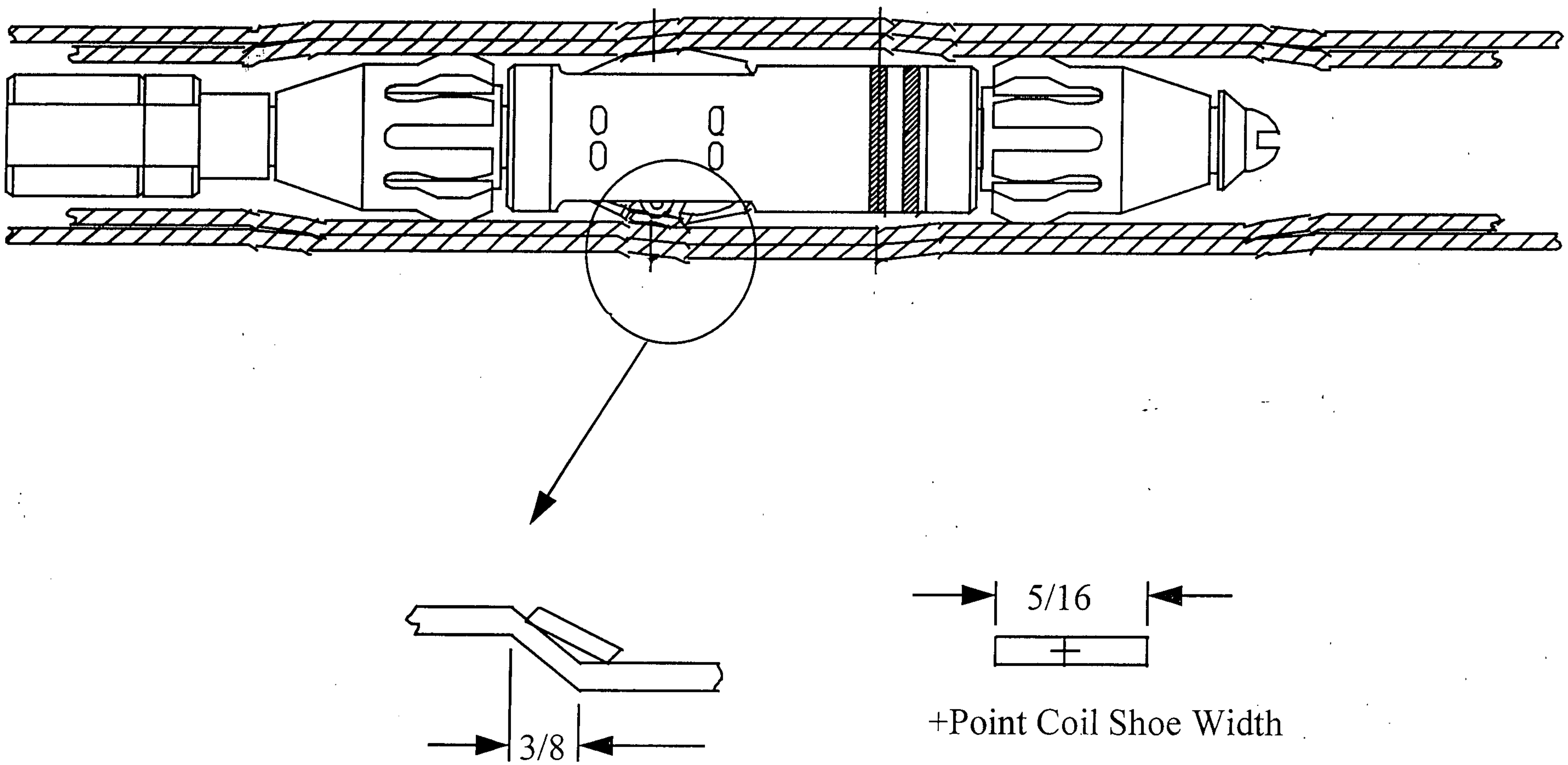


Bobbin Profile +Point Response

Normal HRLT (non rolldown condition)

Figure 5

Probe Lift Off In Non-Rolldown Geometry



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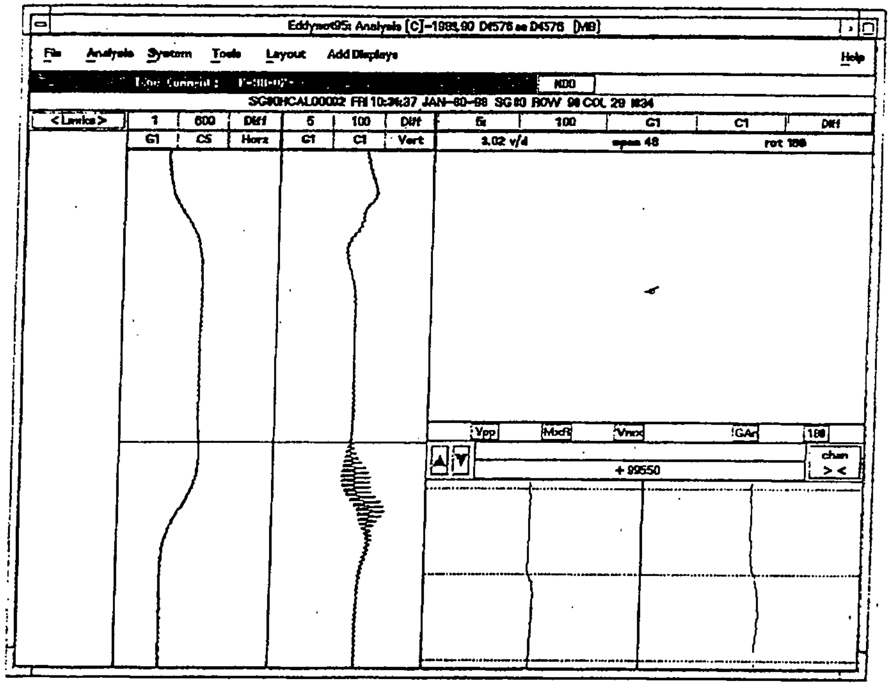
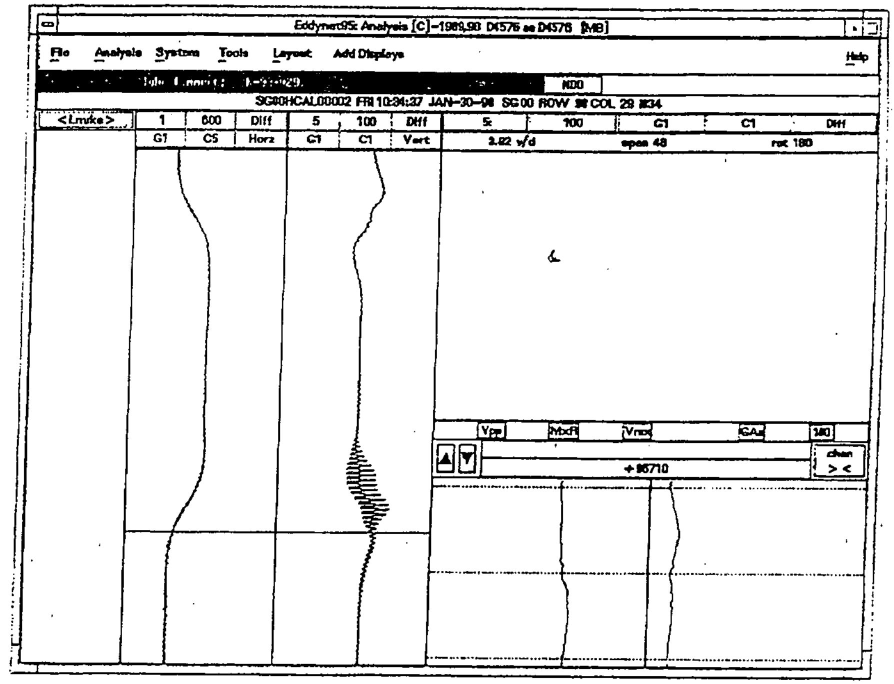
Selection of Flaw Centerline





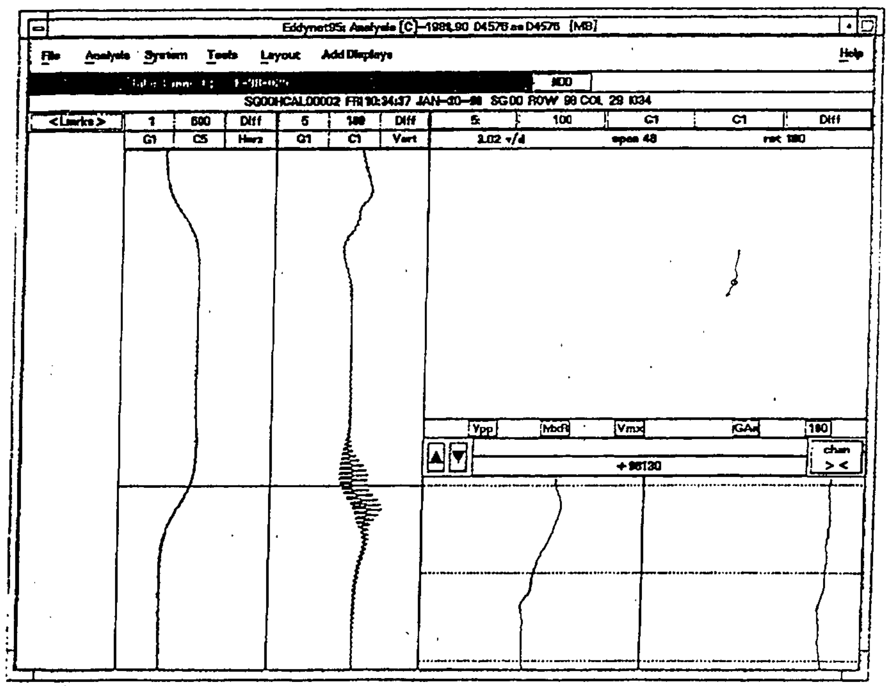
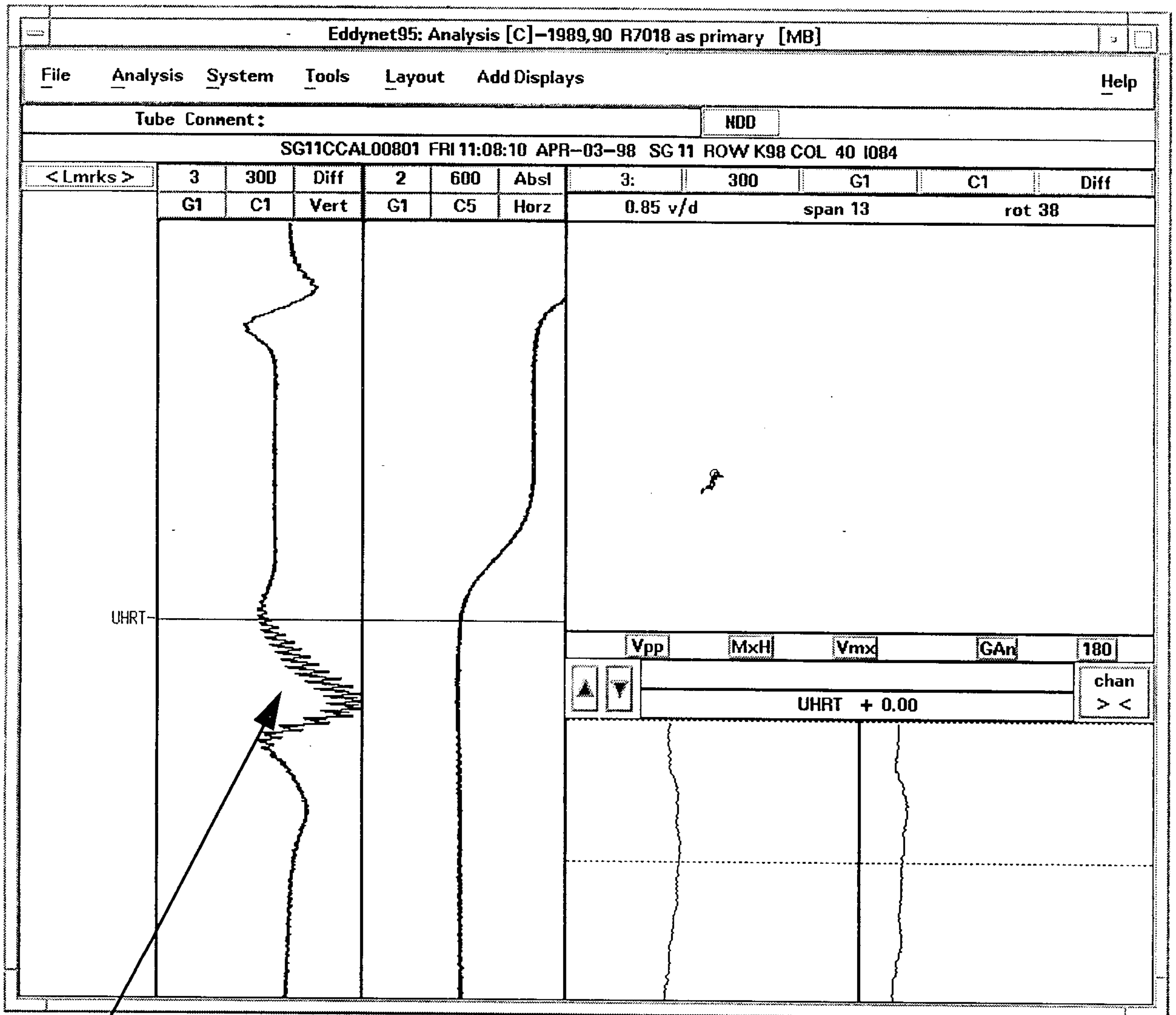


Figure 7

Acceptable Condition

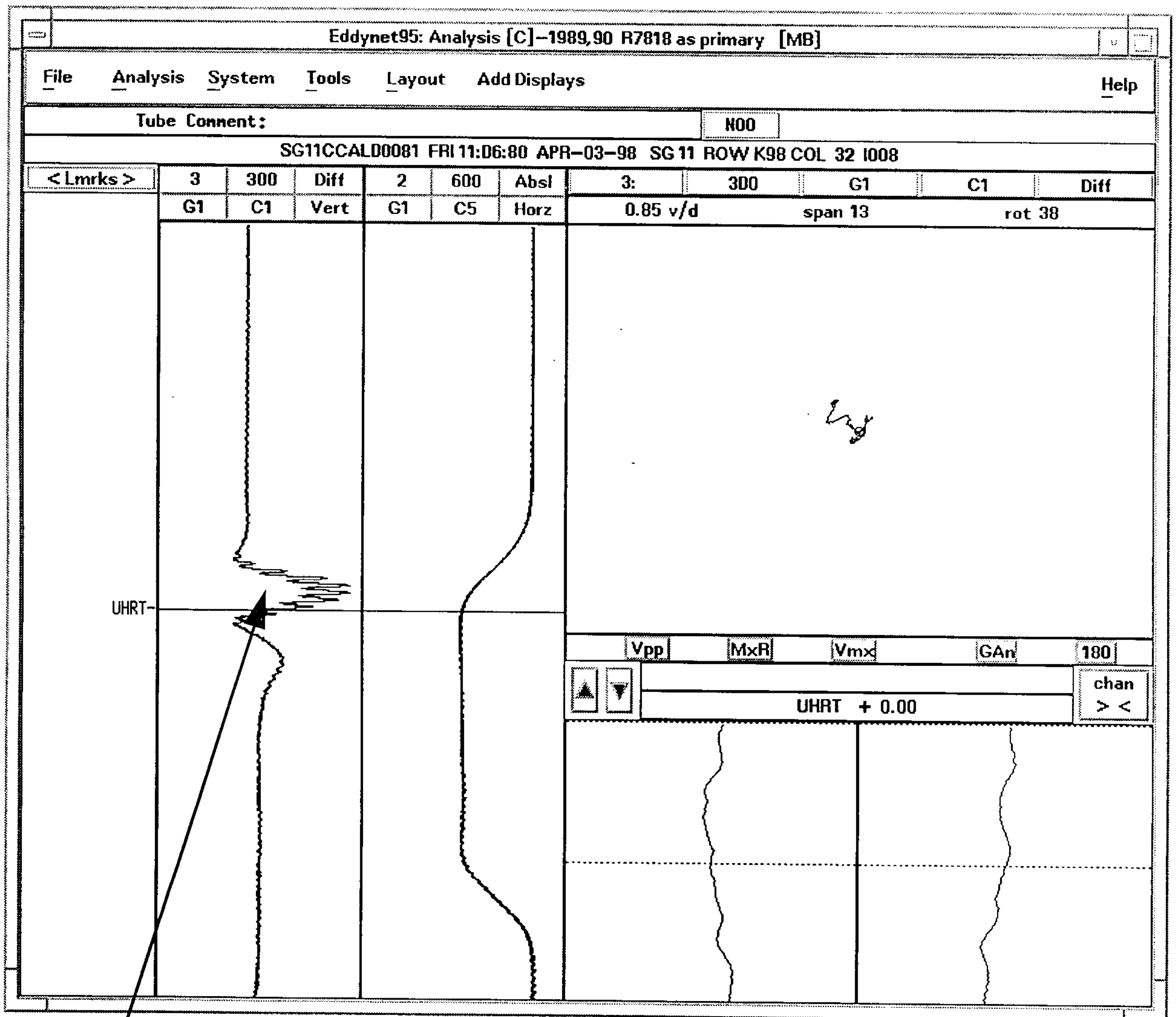


+Point Response Bobbin Profile

Center of +Point coil field occurs prior to HRUT tangent point as probe is pushed through the HEJ

Figure 8

Unacceptable Condition



+Point Response Bobbin Profile

Center of +Point coil field occurs after HRUT tangent point as probe is pushed through HEJ