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APR 10 2000

TVA-WBN-TS-99-14 10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

Gentlemen:

In the Matter of	)	Docket No. 50-390
Tennessee Valley Authority	)	

WATTS BAR NUCLEAR PLANT (WBN) - UNIT 1 - TECHNICAL SPECIFICATION (TS) CHANGE NO. WBN-TS-99-014 - STEAM GENERATOR ALTERNATE REPAIR CRITERIA FOR AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)

In accordance with the provisions of 10 CFR 50.4 and 50.90, TVA is submitting a request for an amendment to WBN's license NPF-90 to change the TSs for Unit 1. The proposed change WBN-TS-99-014 provides an alternate 1.0 Volt repair criteria for axial ODSCC.

NRC issued Generic Letter 95-05, "Voltage-Based Criteria for Westinghouse Steam Generator Tubes Affected by Outside Diameter Stress Corrosion Cracking," on August 3, 1995. This letter provided guidance to nuclear plants with Westinghouse steam generators that wished to request a license amendment to the technical specifications to implement this alternate steam generator tube repair criteria specifically for ODSCC at the tube-to-tube support plate intersection. NRC has approved the repair method for several plants with Westinghouse steam generators, e.g., Comanche Peak Nuclear Plant, letter dated September 22, 1999, Diablo Canyon Nuclear Plant, letter dated March 12, 1998, Sequoyah Nuclear Plant, letter dated April 9, 1997, etc.

TVA has determined that there are no significant hazards considerations associated with the proposed change and that

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the change is exempt from environmental review pursuant to the provisions of 10 CFR 51.22(c)(9). The WBN Plant Operations Review Committee and the WBN Nuclear Safety Review Board have reviewed this proposed change and determined that operation of WBN Unit 1 in accordance with the proposed change will not endanger the health and safety of the public. Additionally, in accordance with 10 CFR 50.91(b)(1), TVA is sending a copy of this letter and enclosures to the Tennessee State Department of Public Health.

Enclosure 1 to this letter provides the description and evaluation of the proposed change. This includes TVA's determination that the proposed change does not involve a significant hazards consideration, and is exempt from environmental review. Enclosure 2 contains copies of the appropriate TS pages from Unit 1 marked-up to show the proposed change. Enclosure 3 forwards the revised TS pages for Unit 1 which incorporate the proposed change.

Enclosure 4 identifies a commitment to revise the Final Safety Analysis Report to include a reference to this letter for implementing the alternate repair criteria for ODSCC once approved by NRC.

TVA requests approval for the use of the voltage based alternate repair criteria before the upcoming September 2000 Cycle 3 outage. The implementation of this alternate repair criteria could prevent unnecessary plugging of steam generator tubes with indications of ODSCC at tube support plate intersections due to the uncertainty related to depth estimation of ODSCC at tube support plates. TVA is prepared to meet with the Staff as necessary to facilitate the Staff's review. If you have any questions concerning this change, please contact P. L. Pace at (423) 365-1824.

Sincerely,

W. R. Lagergren

for R. T. Purcell

Enclosures
cc: See page 3

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Subscribed and sworn to before me
on this 10th day of April 2000.

E. Jeannette Long
Notary Public

My Commission Expires June 27, 2001

cc (Enclosures):

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ENCLOSURE 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
DOCKET NO. 50-390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-99-14
DESCRIPTION AND EVALUATION OF THE PROPOSED CHANGE

I. DESCRIPTION OF CHANGE

TVA proposes to revise the Watts Bar Nuclear Plant (WBN) Unit 1 Technical Specification (TS) to incorporate new requirements associated with steam generator (SG) tube inspection and repair. The new requirements establish an alternate voltage based SG tube repair criteria at tube support plate (TSP) and Flow Distribution Baffle (FDB) plate intersections. This change is consistent with NRC Generic Letter (GL) 95-05 "Voltage-Based Repair Criteria for Westinghouse Steam Generator Tubes Affected By Outside Diameter Stress Corrosion Cracking."

The proposed changes affect the following sections:

1. TS Section 3.4.13, "RCS Operational Leakage" and the associated TS Bases are being revised to provide the Limiting Condition for Operation (LCO) for the allowable leakage.

2. TS Section 5.7.2.12, "Steam Generator (SG) Tube Surveillance Program," is being revised to incorporate the new requirements for an alternate voltage based steam generator tube inspection and repair criteria.

3. TS Section 5.9.9, "SG Tube Inspection Report," is being revised to include the new NRC notification and reporting requirements for the implementation of the voltage based repair criteria.

The specific changes to the TS and Bases are noted in the marked up copies of the applicable TS pages provided in Enclosure 2. By separate letter, TVA is also submitting a request for an alternate repair criteria F* (pronounced F-Star) in the tubesheet region of the steam generator that affects some of the same technical specification pages. If changes are needed to the enclosed technical specification pages because of prior approval of the F* criteria, this effort will be coordinated through the NRC WBN Project Manager.

II. REASON FOR CHANGE

TVA is proposing to change WBN Unit 1 TS to reduce the need for repairing or plugging SG tubes having indications that exceed the current TS depth-based plugging limit. TVA proposes to add alternate voltage based tube repair criteria (ARC) at TSP intersections and FDB plate intersections that are voltage based and maintain structural and leakage integrity of tubes with indications of Outside Diameter Stress Corrosion Cracking (ODSCC) within the confines of the TSP and FDB regions.

The proposed change would preserve the reactor coolant flow margin and reduce the radiation exposure incurred in the process of plugging or repairing the SG tubes (approximately 0.060 man-Rem per tube of exposure would be saved for a plugging operation). Other benefits of not plugging TSP indications that meet the ARC would be a reduction in man-hours and potential impact to critical path time during refueling outages.

TVA requests approval for use of the voltage based ARC per GL 95-05 criteria for the upcoming September 2000 inspection. Eddy current signals were noted during the last inspection that may be precursors to the occurrence and detection of ODSCC at TSP intersections. As the ODSCC mechanism at WBN Unit 1 may be assumed to be in a formative stage, the implementation of the ARC could prevent unnecessary plugging of steam generator tubes with indications of ODSCC at TSP intersections due to the uncertainty related to depth estimation of ODSCC at TSPs.

III. SAFETY ANALYSIS

WBN Unit 1 is a Westinghouse 4-loop pressurized water reactor plant which utilizes Model D3 SGs with drilled hole carbon steel tube support plates and ¾-inch diameter mill annealed Alloy 600 tubing. The design number of tubes in a SG is 4674. These SGs incorporate a FDB plate located approximately 8 inches above the top of the tube sheet, similar to Model D4 steam generators in which the voltage based plugging criteria have been applied. The tube holes located in the FDB design include an increased nominal tube-to-plate diametrical gap ranging from approximately 0.115 inches to 0.150 inches, compared to 0.023 inches nominal gap at the TSPs. Based on this increased tube to plate gap at the FDB, an upper voltage repair limit based on providing tube structural integrity against a pressure equivalent of three times the normal primary-to-secondary tube differential pressure is provided. The design features of the WBN Unit 1 steam generators are consistent with the scope of applicability of GL 95-05.

Summary of Voltage Based Repair Criteria

The following items outline the specific requirements and actions associated with the implementation of the voltage based repair criteria at WBN Unit 1:

- All tubes shall be inspected using the bobbin coil. The inspection shall include hot leg TSP intersections and cold leg intersections down to the lowest TSP for which ODSCC has been identified.

- Bobbin coil flaw indications greater than 1.0 volt shall be repaired and inspected by a rotating pancake coil probe (or equivalent) to evaluate the presence of detectable ODSCC and to confirm that the dominant corrosion mechanism occurring is axially oriented ODSCC.

- Eddy current analysis guidelines shall be compatible with and satisfy GL 95-05 requirements.

- TS operational leak rate limit (Limiting Condition for Operation (LCO) 3.4.13.d and e) shall be reduced to 600 gpd total primary to secondary leakage through all SGs and 150 gpd primary to secondary leakage through any one SG.

- Axial ODSCC indications less than or equal to the lower voltage repair limit of 1.0 volt, as measured by bobbin coil, may remain in service without further inspection or analysis.

- Axial ODSCC bobbin coil indications greater than 1.0 volt and less than or equal to the upper voltage repair limit (V_{URL}) can remain in service if rotating pancake coil (RPC) inspection indicates no detectable degradation.

- Axial ODSCC bobbin coil indications greater than 1.0 volt and less than or equal to the upper voltage repair limit (V_{URL}) must be repaired if RPC inspection indicates detectable degradation.

- Axial ODSCC bobbin coil indications exceeding the upper voltage repair limit must be repaired.

- Postulated faulted condition primary-to-secondary leakage through indications to which the criteria is applied shall be calculated. Postulated leakage in the limiting steam generator shall be less than the bounding faulted condition leakage necessary to ensure that offsite doses remain a small fraction of the 10 CFR Part 100 reactor site criteria and that control room doses remain within 10 CFR 50 Appendix A, General Design Criteria (GDC) 19 limits.

- Projected tube burst probability at a pressure differential equal to the limiting faulted condition pressure differential shall be calculated and compared to the reporting value of 1.0×10^{-2} in the limiting steam generator.

- TVA is requesting the use of a voltage dependent probability of prior cycle detection (POPCD), the methodology of which was originally submitted to the NRC for review by the Nuclear Energy Institute (NEI).

Acknowledgment of GL 95-05 Performance Requirements

An item by item declaration of the individual items of GL 95-05 which require utility action or present a specific analysis methodology is provided in Table 1.

Application

The repair criteria will be applied to axially oriented ODSCC indications at tube-to-TSP and FDB intersections (within the plate thickness) of the steam generator tube bundle (Section 1.a of Attachment 1 to GL 95-05). Rotating probe examination of TSP intersections at WBN Unit 1 has indicated that the degradation morphology currently, is generally below detectable thresholds for the +Point probe and only signals interpreted as possible precursors to degradation are present, e.g., deposits in the tube/TSP crevice.

TVA will follow the Industry Tube Pull Program presented to the NRC in NP-7480-L, Addenda 3 (May 1999), *"Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1999 Database Update,"* EPRI, Palo Alto, CA. Tubes pulled from other plants using both 7/8-inch and ¾-inch outside diameter (OD) tubing for indications of ODSCC at TSPs have been shown to have crack morphology consistent with the EPRI database used for the supporting voltage correlations. Any other type of tube degradation or any other location in the tube bundle shall continue to be evaluated in accordance with existing WBN Unit 1 TSs. The observation of circumferential cracks, or primary water stress corrosion cracking associated with TSP indications, or ODSCC beyond the TSP thickness will be reported to the NRC prior to return to power as inserted in TS Section 5.9.9.

The voltage based repair criteria will not be applied at the following tube-to-TSP intersections of the steam generator tube bundle per Section 1.b of Attachment 1 of GL 95-05:

1. At locations where tubes with degradation could substantially deform or collapse during postulated loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SSE) loading (Section 1.b.1 of Attachment 1 of GL 95-05.) Bounding analysis results indicate that 466 intersections out of approximately 100,588 total intersections are to be excluded from application of

the criteria based on a potential to experience permanent deformation of >0.030-inch during the combined LOCA + SSE event. The tube collapse determination methodology is identical to that employed in evaluations of other Model Westinghouse SGs. Table 2 provides a summary of the number of intersections per TSP which are excluded due to combined LOCA + SSE loadings. (See also Attachment 2 of this enclosure) The top TSP contains the largest number of excluded intersections, 256, which represent 2.7 percent of the total hot and cold leg intersections at that TSP. Excluded tubes are clustered around wedge locations. It should be noted that a large number of the affected tubes contain multiple excluded intersections, both at multiple plate elevations and at both hot and cold leg intersections. Hence, the total number of tubes affected is less than the number of affected intersection locations. Bounding LOCA rarefaction wave loadings and WBN Unit 1 seismic input data were used to develop the list of tubes in Table 2. The applied loadings are compared to data from a TSP crush test program to ultimately define the list of excluded intersections. This analysis is conservative in that leak-before-break of the WBN Unit 1 loop piping has been approved by the NRC Staff, and use of small break LOCA loadings is, therefore, justifiable, although not credited in this submittal. The large break LOCA loadings used conservatively bound the small break locations. The intersections in Table 2 represent an in-leakage potential due to opening of the flaw from the potential deformation. Deformation potential does not imply that the tube will completely collapse, thereby resulting in a total loss of flow area during a LOCA + SSE event.

The tube exclusion analysis has been shown to be quite conservative. The NRC has permitted the use of small break combined event loadings for ARC evaluation for another plant with D4 steam generators (NRC Letter from Ramin R. Assa, Project Manager, NRR, to Commonwealth Edison Co., "Safety Evaluation Regarding Leak-Before-Break Analysis - Byron Station, Units 1 and 2, and Braidwood Station, Units 1 and 2", dated October 25, 1996). Should a more detailed tube exclusion analysis be required at a future date, a subsequent submittal can be made which would include the results of a detailed LOCA + SSE tube exclusion analysis using WBN Unit 1 specific small break LOCA + SSE combined loadings and a more refined steam generator finite element model.

2. At tube-to-TSP intersections with dent signals exceeding 5.0 (bobbin) volts (Section 1.b.2 of Attachment 1 of GL 95-05), and any indications confirmed by RPC will be repaired.

3. At tube-to-TSP intersections at which there are mixed residuals of sufficient magnitude to mask a 1.0 volt bobbin volt ODSCC indication (Section 1.b.3 of Attachment 1 of GL 95-05), and any indications confirmed by RPC will be repaired.

4. At tube-to-TSP intersections where copper deposits interfere with bobbin volt signals, and any indications confirmed by RPC (or +Point) will be repaired (Section 1.b.4 of Attachment 1 of GL 95-05). There are no known occurrences of copper deposits at tube/TSP intersections in any Westinghouse SGs for which the ARC has been applied. There are currently no plans to perform chemical cleaning of the secondary side of the WBN Unit 1 SGs.

Tube Integrity Evaluation

There are three principal engineering analyses that shall be performed during each implementation of the 1.0 volt ARC at WBN Unit 1:

a. Prediction of SG bobbin voltage population distribution.
b. Calculation of SG tube leakage during a postulated main steam line break (MSLB).
c. Calculation of SG tube burst probability during a postulated MSLB.

The latest approved EPRI database (3/4-inch diameter tubing) utilizing NRC approved data exclusion criteria, NP-7480-L, Addendum 1 (November 1996), Addendum 2 (April 1998), and Addendum 3 (May 1999), "*Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 [and 1998 & 1999 respectively] Database Update,*" EPRI, Palo Alto, CA, will be applied in the voltage correlations (burst, probability of leakage, MSLB leak rate) used for the leak rate, burst probability and upper voltage repair limit calculations.

The NRC approved industry protocol for updating the database will be followed by TVA. The correlations of NP-7480-L, Addendum 3 (May 1999), "*Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1999, Database Update,*" EPRI, Palo Alto, CA are slightly conservative relative to those of Addendum 2 and are expected to be used if NRC approval of Addendum 3 is obtained prior to the next outage.

The methodology to be applied by TVA at WBN Unit 1 for the performance of these analyses, including correlations which relate bobbin voltage amplitudes, free span burst pressure, probability of leakage and associated leak rates is documented in Westinghouse WCAP-14277, Revision 1, "*SLB Leak*

Rate and Tube Burst Probability Analysis Methods for ODSCC at TSP Intersections," December 1996 and is consistent with the methodology of Attachment 1, Section 2 of GL 95-05. Any future NRC approved revisions to Westinghouse WCAP-14277 or equivalent methodology reports may be considered for implementation. In addition, the upper voltage repair limit used to repair bobbin indications independent of RPC confirmation is determined at each outage based on the guidance of Section 2.a.2 of GL 95-05.

MSLB Tube Leak Rate

The calculated maximum allowable tube leak rate for WBN Unit 1 SGs during a postulated MSLB event shall not exceed 10.0 gpm in the faulted loop (calculated at standard temperature and pressure conditions). The 10.0 gpm leak rate is calculated using as an input the TS 3.4.16, "RCS Specific Activity," limit for RCS dose equivalent Iodine (I-131) concentration of 1.0 μCi/gm, and is consistent with Section 2.b.4 of GL 95-05 and other NRC approved MSLB maximum allowable leak rate calculations. The primary coolant noble gas activity concentrations used are also consistent with the TS limits. The 10.0 gpm leak rate value in the faulted loop (with leakage in the intact loops equal to the TS normal operation leakage limit of 150 gpd) will not result in either control room dose exceeding the GDC-19 limit or the off-site dose exceeding 10 percent of the 10 CFR 100 guidelines, and therefore is consistent with the WBN Unit 1 licensing basis. The establishment of the 10.0 gpm leak rate value is controlled by the zero to 2 hour offsite doses at the site boundary for the accident initiated iodine spike case, not control room dose. If it is determined during the operating cycle that this leakage limit might be exceeded, the reporting requirements of GL 95-05, Section 6.a.1 will be followed. The calculated leak rate limit and maximum allowable leak rate values specified for WBN Unit 1 are specified as room temperature values, therefore, these values are compared using a consistent set of reference conditions.

Consistent with the guidance of GL 95-05, Section 2.c, the WBN Unit 1 MSLB leak rate analysis performed prior to returning the SGs to service may be performed based on the projected next end of cycle (EOC) voltage distribution or the actual measured distribution at a given outage. The method to be used for the first outage when ODSCC indication growth rates are available will be based on the indications found during that outage. As noted in GL 95-05, it may not always be practical to complete these calculations prior to returning the SGs to service. Under these circumstances, it is acceptable to use the actual measured bobbin voltage distribution instead of the projected EOC voltage distribution to determine whether the reporting criteria is being satisfied. Using the EOC 3 voltage distribution, if any, the projected EOC 4 MSLB leak rate will be calculated using a probability of detection (POD) of 0.6.

The offsite dose calculations were performed by TVA consistent with GL 95-05 recommendations and other analysis methods/models previously approved by the NRC staff related to the GL 95-05 plugging criteria. A summary of the MSLB dose analysis results and inputs are provided in Attachment 1 to this Enclosure. The calculated radiological consequences of the exclusion area boundary and the low population zone are larger than previously reported for the postulated steamline break event due to the increased leakage and more conservative iodine spiking factors. However, the calculated radiological consequences remain in compliance with the guidelines in the Standard Review Plan, Chapter 15, 10 CFR 50, Appendix A, GDC-19 and 10 CFR 100. Therefore, it is concluded that the proposed changes do not result in a significant increase in the radiological consequences of an accident previously analyzed. The analysis of MSLB effects upon the reactor core, fuel, and DNBR potential described in the Final Safety Analysis Report (FSAR) indicates no fuel failures are predicted as a result of this event.

Pressurizer Power Operated Relief Valve Availability

WBN Unit 1 TS Section 3.4.11, "Pressurizer Power Operated Relief Valves (PORVs)," requires that both PORVs and their associated block valves shall be OPERABLE during Modes 1, 2, and 3. Action statements are included in the TS which limit plant operation with one PORV and/or associated block valve inoperable. Compliance with TS Section 3.4.11 meets the requirements of NRC Generic Letter 95-05 and provide the maximum pressure for the MSLB event. With the PORV set point of 2335 psig and a 3 percent uncertainty on the opening pressure, the MSLB pressure differential for SG tubing is 2405 psig. This differential pressure is used for the calculation of the degradation structural limit and for calculations of tube MSLB tube burst probability per SG.

Probability of Detection

GL 95-05, Section 2.b.1 requires that the frequency distribution by voltage of bobbin indications actually found during an inspection should be scaled upward by a factor of one over the POD to specify indications considered to be present in the SG tubes for the next operating cycle. A constant POD value of 0.6 is also specified for use in evaluating the potential effect of missed indications. The POD value of 0.6 is considered exceptionally conservative by the industry, especially for larger voltage indications. GL 95-05 also allows the use of an alternative POD value or a distribution function may be used, subject to NRC approval. Because there are neither historical nor growth data distributions for indications in WBN Unit 1 SG tubes, the

POD value of 0.6 will be directly applied to the distribution of detected indications for the first cycle of implementation.

The use of a constant POD results in mathematically including an additional percentage of an indication in the distribution of indications for performing leak rate and burst probability analyses. This leads to large voltage indications being included in the beginning of cycle (BOC) distributions even if repaired; large voltage indications contribute significantly to the projected leak rates and tube burst probabilities. The ARC data indicate that the actual POD approaches unity above about 2.5 to 3.5 volts. The net effect is that large indications with a POD of or near unity are caused to dominate the results of the evaluations of the next operating cycle because of the use of a lower than realistic POD.

An alternative treatment of the POD, known as the probability of prior cycle detection (POPCD), has been developed by the industry and is documented in NP-7480-L, Addenda 1, 2, and 3. If that methodology is approved by the staff, it may be implemented in evaluations performed at future plant outages.

For ARC applications, the important indications are those that could significantly contribute to EOC leak or burst probability. These significant indications may be expected to be detected by the bobbin coil examination and confirmed by subsequent RPC inspection. Thus, the population of interest for APC POD assessments are the current inspection RPC confirmed indications that were detected or not detected at the prior inspection. This POPCD approach to defining a POD function accounts for undetected indications, indications below the detection threshold and new indications. Consequently, there is no need to modify the associated POD values or otherwise account for new indications. Using the information provided in NP-7480-L, Addendum 1, 2, and 3, it is concluded that a large ARC field experience database is available and has been applied to develop a voltage dependent POD. The recommended POD, based on the lower 95 percent confidence limit of the data, is lower than the NRC recommended POD value of 0.6 below about 0.5 volts, increases to 0.9 at 1.2 volts, and approaches unity at 2.5 to 3.5 volts. Flaw population growth increases the need to apply a voltage dependent POD to reduce the number of high voltage "phantom" indications included in the leak and burst analysis as a result of applying a constant POD of 0.6.

Application of Unconfirmed Indications in BOC Distribution

GL 95-05, Section 2.b.1 provides an option for applying a fraction of unconfirmed indications in the BOC voltage distribution. NRC approval for applying a fractional representation of the RPC no detectable defect (NDDs) in the MSLB leak rate projection has been approved at Beaver Valley Unit 1. Currently, a sufficient database for RPC NDDs does not exist for WBN Unit 1. Once sufficient data has been collected to include a fractional representation of the RPC NDDs into the MSLB leak rate projection, TVA may submit an update for NRC approval. This data will compare the RPC NDDs left in service at the end of the previous and current cycles. Inspection data from other plants indicates that typically about 50 percent of the RPC NDDs progress to detectable RPC indications from one cycle to the next, compared to the GL 95-05 guidance which includes all RPC NDDs in the MSLB leak rate projection.

Voltage Growth Due to Defect Progression

GL 95-05 Section 2.b.2(2) provides guidance on the determination of a defect growth curve to be used in the evaluation of the SG indication population. There are three possibilities considered:

1. When there has been no prior TSP ODSCC detected,
2. When there are growth data from only one prior cycle of operation, and
3. When there are data from two or more prior cycles of operation.

Because there are no reported indications from previous inspections, a bounding probability distribution function of growth rates will be used for the initial cycle of application of the ODSCC ARC. The bounding curve will be based on the experiences of similarly designed and operated plants.

For subsequent inspections, TVA will continue to abide by the specified guidance in GL 95-05. For example, if growth rates become available from more than one cycle of operation, the more conservative growth rate of the previous two cycles shall be used for the projection of bobbin voltage distribution during the next operating cycle. Both BOC and corresponding EOC bobbin indications at a TSP intersection are necessary to specify a growth data point. The following points are noted relative to the establishment of a growth curve for subsequent inspections at WBN Unit 1:

- Growth data from the previous two cycles may be combined if necessary to obtain at least 200 data points in the distribution, otherwise industry data will be used.
- If > 200 indications are unavailable on a per SG basis, the more limiting of SG specific or all SG combined growth will be used.

- Negative growth rates will not be used in growth rate distributions used to make voltage projections although those rates shall be used in establishing average growth for determining the upper voltage repair limit of Sections 2.a.2 and 2.a.3 of GL 95-05

Establishment of Upper Voltage Repair Limits (V_{URL}) for TSP and FDB Intersections

The voltage structural limit is the voltage from the regression analysis of the burst pressure on the bobbin voltage, at the 95 percent lower prediction interval curve reduced to account for lower tolerance limit material properties at 650°F. The upper voltage repair limit must be adjusted for flaw growth during an operating interval and to account for non-destructive examination (NDE) uncertainty. The upper voltage repair limit, V_{URL}, is determined from the following equation:

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

where V_{SL} is the structural limit, V_G represents the growth allowance, and V_{NDE} represents the allowance for potential sources of error, including analyst variability. The structural limit voltage is taken from the latest NRC approved database (currently contained in NP-7480-L, Addenda 1, 2, and 3). The proximity of the TSP prevents tube burst during normal operating conditions, so the structural limit voltage at TSP intersections is calculated using a differential pressure of 1.43 times the bounding MSLB pressure differential of 2405 psig, or 3367 psig. The increased tube to FDB gap may not provide sufficient constraint to prevent burst at locations within the FDB, hence the FDB voltage structural limit is established using a pressure loading of 3 times the normal operating differential pressure across the SG tubes. The current maximum SG tube ΔP_{NOP} for WBN Unit 1 is on the order of 1300 psig, thus the differential pressure used to establish the voltage structural limit for the FDB locations is 3900 psig. In summary, V_{SL} for the TSP intersection locations is 6.03 volts while V_{SL} for the FDB intersection locations is 3.81 volts. Using the above equation, with values for V_{NDE} of 20 percent of V_{URL} and V_G of 30 percent of V_{URL} (minimum value allowable per GL 95-05), the measurement upper voltage repair limits become 3.71 and 2.34 volts respectively.

Inspection Criteria

All steam generator tubes will be inspected with the bobbin coil during each normally scheduled refueling outage at WBN Unit 1. The inspection includes hot leg side tube to TSP intersections and cold leg side tube to TSP intersections down to the lowest cold leg side TSP with identified ODSCC. Data acquisition and analysis will be performed to provide consistent methodology as that described GL 95-05 and NDE guidelines utilized for the most recent field applications of the criteria, as updated by the clarifications listed below, which include use of the updated probe wear guidelines and new probe acceptability guidelines. The supplementary guidance of Section 3 of GL 95-05 will be applied with the clarifications noted below. Any indication with bobbin voltage exceeding 1.0 volts shall be inspected with a RPC or equivalent, and shall be repaired if the bobbin indication is confirmed as a flaw by RPC. Any indication will be plugged or repaired regardless of any RPC inspection results, if the bobbin voltage exceeds the upper voltage repair limit as obtained per Section 2.a.2 of GL 95-05. For WBN Unit 1, specific upper voltage repair limits are separately developed for both TSP and FDB intersections.

<u>New Probe Variability Criteria</u>

In NEI letter from Alex Marion to Brian Sheron of NRC (NRR) dated January 23, 1996, NEI provided a methodology for meeting the new probe variability criteria in Section 3.c.2 of GL 95-05. The NRC staff requested additional information via NRC letter from Brian Sheron (NRR) to Alex Marion of NEI dated February 9, 1996. Initial NRC concurrence of this methodology was provided in a letter from Mr. Brian Sheron to Alex Marion of NEI, dated March 18, 1996. Additional industry information was provided by NEI in NEI letter from Alex Marion to Brian Sheron of NRC (NRR) dated October 15, 1996, which dealt with test data for probes manufactured by Westinghouse and Zetec. Briefly summarized, this methodology requires, in part, that the primary frequency and mix frequency voltage response of a new probe be compared to the nominal response determined by the vendor to ensure that the new probe is within ± 10 percent of the nominal response for both the primary and mix channels. In a letter from Brian Sheron to Dave Modeen of NEI, the NRC determined that NEI has provided sufficient information in response to the NRC staff request in the February 9, 1996 letter. TVA will follow the guidance provided in the NRC letter dated March 18, 1996 as supplemented by the test data contained in the NEI letter dated October 15, 1996.

<u>Probe Wear Criteria</u>

In NEI letter from Alex Marion to Brian Sheron of NRC (NRR) dated January 23, 1996, NEI provided an alternative to the probe wear criteria in GL 95-05. NRC concurrence of this methodology was provided via NRC letter from Brian Sheron (NRR) to Alex Marion of NEI dated February 9, 1996. Briefly summarized, this alternative to the GL 95-05 probe wear criteria requires that when a probe does not pass the probe wear check (15 percent), tube locations inspected with the worn probe having detected indications with amplitudes greater than or equal to 75 percent of the repair voltage limit, i.e., 1.0 volt for ¾-inch OD tubes, will be re-inspected with a new acceptable probe. TVA will follow the guidance provided in NRC letter from Brian Sheron (NRR) to Alex Marion of NEI dated February 9, 1996.

Alternatives to the Rotating Pancake Coil

GL 95-05 (Background, Page 3 of 7) permits use of alternates to the rotating pancake coil for RPC inspections. Currently, the +Point probe and Cecco probes are considered by the industry as acceptable alternatives to RPC. TVA will utilize the +Point probe for TSP indication confirmation at EOC 3 and +Point or equivalent probe at later inspections.

Tube Removal and Examination/Testing

The WBN Unit 1 program for tube removal and examination will follow the Industry Tube Pull Program presented to the NRC in NP-7480-L, Addendum 3 (May 1999). Currently, no tubes have been removed from WBN Unit 1. The basic feature of the recommendation is that the decision to remove a tube section should be based primarily on the usefulness of the expected data in expanding the database for the leak rate correlation. This is based on the observation that there have been no instances where the actual morphology did not match the expected morphology of the degradation, i.e., ODSCC at TSP intersections. Thus, confirmation of morphology may be delayed by one operating cycle at the utility's discretion if there are no indications for which a tube pull is required to expand the leak rate database. This means that tube section removal would be delayed for one operating cycle if no indications ≥ 3.0 volts are found during the inspection of the SGs at WBN Unit 1. Table 8-3 of NP-7480-L, Addenda 1, 2, and 3 delineates a specific schedule for obtaining needed leak rate data while avoiding redundancy relative to the existing data.

Operational Leakage

The TS operational leakage limit, LCO 3.4.13, will be changed from 1 gpm total for all SGs, 500 gpd maximum in any SG, as currently defined, to 600 gallons per day total through all SGs and 150 gpd maximum through any one SG. SG tubes with known leaks will be repaired prior to returning the SGs to service, consistent with GL 95-05.

Reporting Requirements

WBN Unit 1 will comply with the reporting requirements of Section 6 of GL 95-05.

IV. NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

TVA has concluded that operation of Watts Bar Nuclear Plant (WBN) Unit 1 in accordance with the proposed change to the technical specifications does not involve a significant hazards consideration. TVA's conclusion is based on its evaluation, in accordance with 10 CFR 50.91(a)(1), of the three standards set forth in 10 CFR 50.92(c).

A. *The proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.*

Tube burst criteria are inherently satisfied during normal operating conditions due to the proximity of the tube support plate. Test data indicates that tube burst cannot occur within the tube support plate (TSP), even for tubes which have 100 percent through-wall electric discharge machining (EDM) notches, 0.75 inches long, provided that the TSP is adjacent to the notched area. Since tube to tube support plate proximity precludes tube burst during normal operating conditions, use of the criteria must retain tube integrity characteristics which maintain a margin of safety of 1.43 times the bounding faulted condition [main steam line break(MSLB)] differential pressure of 2405 psig. As previously stated, the Regulatory Guide (RG) 1.121 criterion requiring maintenance of a safety factor of 1.43 times the MSLB pressure differential on tube burst is satisfied by 3/4-inch diameter tubing with bobbin coil indications with signal amplitudes less than V_{SL} = 6.03 volts, regardless of the indicated depth measurement. At the flow distribution baffle (FDB), a safety factor of 3 against the normal operating condition ΔP is applied; here a voltage of V_{SL} = 3.81 volts satisfies the burst capability recommendation.

The upper voltage repair limit (V_{URL}) will be determined prior to each outage using the most recently approved NRC database to determine the tube structural limit (V_{SL}). The structural limit is reduced by allowances for nondestructive examination (NDE) uncertainty (V_{NDE}) and growth (V_G) to establish V_{URL}. As an example, the NDE uncertainty component of 20 percent and a voltage growth allowance of 30 percent per full power year can be utilized to establish a V_{URL} of 3.71 volts for TSP indications, and 2.34 volts for the FDB indications. The 20 percent NDE uncertainty represents a square-root-sum-of-the-squares (SRSS) combination of probe wear uncertainty and analyst variability. The flaw growth allowance should be an average growth rate or 30 percent per effective full power year, whichever is larger. The 30 percent growth allowance used to determine V_{URL} is conservative for the current conditions at WBN Unit 1. The most current NRC approved database, contained in EPRI NP-7480-L, Addendum 2, was used to establish the V_{URL} values for the FDB and TSP intersections. Once approved by NRC, the industry protocol for updating the database will be followed by TVA, ensuring that the most current database is utilized for future applications of the criteria.

Relative to the expected leakage during accident condition loadings, it has been previously established that a postulated MSLB outside of containment but upstream of the main steam isolation valves (MSIV) represents the most limiting radiological condition relative to the alternate voltage based repair criteria. In support of implementation of the revised repair limit, it will be determined whether the distribution of cracking indications at the tube support plate intersections during future cycles are projected to be such that primary to secondary leakage would result in site boundary doses within a fraction of the 10 CFR 100 guidelines or control room doses within the 10 CFR 50, Appendix A, General Design Criteria (GDC)-19 limit. A separate calculation has determined this allowable MSLB leakage limit to be 10 gallons per minute (gpm) in the faulted loop assuming a reactor coolant system (RCS) dose equivalent iodine concentration of 1.0 μCi/gm. The establishment of the 10 gpm leak rate value is controlled by the 0 to 2 hour offsite doses at the site boundary for the accident initiated iodine spike case, not control room dose.

The methods for calculating the radiological dose consequences for this MSLB are consistent with FSAR Chapter 15 and therefore, the WBN licensing basis. TVA bases the calculated thyroid dose consequences on conversion factors from the International Commission on Radiation Protection (ICRP) Publication 2.

In summary, the calculated radiological consequences of the exclusion area boundary and the low population zone are larger than previously reported for the postulated steamline break event due to the increased leakage and more conservative iodine spiking factors. However, the calculated radiological consequences remain in compliance with the guidelines in the Standard Review Plan, Chapter 15, 10 CFR 50, Appendix A, GDC-19 and 10 CFR 100 reported for the postulated steamline break event. Therefore, it is concluded that the proposed changes do not result in a significant increase in the radiological consequences of an accident previously analyzed.

Consistent with the guidance of Section 2.c of Generic Letter (GL) 95-05, the WBN Unit 1 MSLB leak rate analysis performed prior to returning the SGs to service may be performed based on the projected next end-of-cycle (EOC) voltage distribution or the actual measured distribution at a given outage. The method to be used for the first outage when outside diameter stress corrosion cracking (ODSCC) indication growth rates are available will be based on the indications found during that outage. As noted in GL 95-05, it may not always be practical to complete EOC calculations prior to returning the SGs to service. Under these circumstances, it is acceptable to use the actual measured bobbin voltage distribution instead of the projected EOC voltage distribution to determine whether the reporting criteria is being satisfied.

Therefore, as implementation of the 1.0 volt voltage-based repair criteria at WBN Unit 1 does not adversely affect steam generator tube integrity and implementation is shown to result in acceptable radiological dose consequences, the proposed TS change does not result in a significant increase in the probability or consequences of an accident previously evaluated within the WBN Final Safety Analysis Report (FSAR).

B. **The proposed amendment does not *create the possibility of a new or different kind of accident from previously analyzed.***

Implementation of the proposed steam generator tube alternate voltage based repair criteria (1.0 volts) does not introduce any significant changes to the plant design basis. Neither a single or multiple tube rupture event would be expected in a steam generator in which the repair limit has been applied (during all plant conditions).

The bobbin probe voltage-based tube repair criteria of 1.0 volt is supplemented by: enhanced eddy current inspection guidelines to provide consistency in voltage normalization, a 100 percent eddy current inspection sample size at the tube support plate elevations, and rotating pancake coil (RPC) inspection requirements for the larger indications left in service to characterize the principal degradation as ODSCC.

TVA will implement a maximum normal operating condition primary to secondary leakage rate limit of 600 gallons per day (gpd) total primary to secondary leakage and 150 gpd primary to secondary leakage per steam generator to help preclude the potential for excessive leakage during all plant conditions. The 150 gpd leakage limit is more restrictive than the current TS operating leakage limit (of 500 gpd) and is intended to provide additional margin to accommodate a stress corrosion crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Leakage trending capability consistent with EPRI Report TR-104788, "Primary-to-Secondary Leak Guidelines" has been implemented at WBN Unit 1.

As steam generator tube integrity upon implementation of the 1.0 volt repair limit continues to be maintained through in-service inspection and primary to secondary leakage monitoring, the possibility of a new or different kind of accident from any accident previously evaluated is not created.

C. ***<u>The proposed amendment does not involve a significant reduction in a margin of safety.</u>***

The use of the voltage-based bobbin probe tube support plate elevation repair criteria at WBN Unit 1 maintains steam generator tube integrity commensurate with the criteria of Regulatory Guide (RG) 1.121. RG 1.121 describes a method acceptable to the NRC staff for meeting GDCs 14, 15, 31, and 32 by reducing the probability or the consequences of steam generator tube rupture. This is accomplished by determining the limiting conditions of degradation of steam generator tubing, as established by in-service inspection, for which tubes with unacceptable cracking should be removed from service. Upon implementation of the proposed criteria, even under the worst case conditions, the occurrence of ODSCC at the tube support plate elevations is not expected to lead to a steam generator tube rupture event during normal or faulted plant conditions. The EOC distribution of crack indications at the tube support plate elevations is confirmed to result in acceptable primary to secondary leakage during all plant conditions and that radiological consequences are not adversely impacted.

As a preventative measure, a total of 214 tubes are excluded from the application of the ODSCC criteria because of the combined effects of loss-of-coolant-accident (LOCA) + safe shutdown earthquake (SSE) on the steam generator component (as required by GDC 2). It was determined that tube deformation or through-wall cracks could occur in these tubes.

As noted previously, implementation of the tube support plate intersection voltage-based repair criteria will decrease the number of tubes which must be repaired. The installation of steam generator tube plugs reduces the RCS flow margin. Thus, implementation of the 1.0 volt repair limit will maintain the margin of flow that would otherwise be reduced in the event of increased tube plugging.

CONCLUSION

Based on the above, it is concluded that using the bobbin coil probe 1.0 volt steam generator tube repair limit for tube support plate intersection ODSCC for repairing or removing tubes from service at WBN Unit 1 is acceptable, and accordingly, a determination that no significant hazards consideration is involved is justified.

V. ENVIRONMENTAL IMPACT CONSIDERATION

The proposed change does not involve a significant hazards consideration, a significant change in the types of or significant increase in the amounts of any effluents that may be released offsite, or a significant increase in indiv idual or cumulative occupational radiation exposure. Therefore, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), an environmental assessment of the proposed change is not required.

Table 1
Acknowledgment of Individual GL 95-05 Performance Criteria

GL 95-05 Item	GL Methodology Concurrence	Comments
1.b	Followed	The exclusion criteria listed will be followed. Enclosure 1, Attachment 2 lists individual intersections excluded due to permanent deformation potential from a combined LOCA + SSE event. V_{URL} for FDB intersections is defined in this Enclosure
2.a.1	Modified from original version, approved by NRC	The latest NRC approved database at the time of the inspection will be utilized (NP-7480-L, Addenda 1 (November 1996), 2 (April 1998), and 3 (May 1999), "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 [and 1998 & 1999 respectively] Database Update," EPRI, Palo Alto, CA). EPRI NP-7480-L, Addendum 2, is the latest NRC approved database. Approval is requested to use the database of Addendum 3.
2.a.2	Followed	See response to section 2.a.1.
2.a.3	Followed	See response to section 2.a.1.
2.b.1	Request for NRC approval	Distribution of bobbin indications included in the MSLB leak rate projection will be based on the voltage dependent probability of detection, or POPCD, as described in NP-7480-L, Addenda 1 (November 1996), 2 (April 1998), and 3 (May 1999), "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 [and 1998 & 1999 respectively] Database Update," EPRI, Palo Alto, CA.
2.b.4	Modified	TVA used the assumption identified in Enclosure 1 Attachment 1 TVA's codes use ICRP-2 conversion factors which are more conservative (by about a factor of 1.3) than the ICRP-30 conversion factors. TVA is not reducing the TS allowable RCS specific activity at this time.
3.b	Followed	TVA will utilize the +Point or future equivalent probe for confirmation of bobbin indications. Rotating pancake coil (RPC) for the purposes of the technical specification (TS) change, also includes the use of comparable or improved nondestructive examination techniques.
3.b.1	Followed	+Point (or future equivalent) will be used for inspection of bobbin voltages > 1.0 volt.
3.b.2	Followed	+Point (or future equivalent) will be used for inspection where copper could influence bobbin signal, possibly masking a 1.0 volt indication. Any indication found at such intersections with RPC should cause the tube to be repaired.
3.b.3	Followed	+Point (or future equivalent) will be used for inspection of all dents > 5 volts, possibly masking a 1.0 volt indication.. Any indications found at such intersections with RPC should cause the tube to be repaired. If circumferential cracking or primary water stress corrosion cracking indications are detected, it may be necessary to expand the RPC sampling plan to include dents less than 5.0 volts.
3.b.4	Followed	+Point (or future equivalent) will be used for inspection of large mixed residuals, possibly masking a 1.0 volt indication. TVA will inspect all intersections with large mixed residuals utilizing a RPC probe. Any indications found at such intersections with RPC should cause the tube to be repaired.
3.c.1	Followed	TVA will use a bobbin coil calibrated against a reference standard used in the laboratory as part of the development of the voltage based approach, through the use of a transfer standard or the latest industry method approved by NRC.
3.c.2	Modified, accepted by NRC	The probe variability limits defined in the NRC letter dated March 18, 1996, as supplemented by test data contained in NEI letter dated October 15, 1996, will be implemented.

Table 1

Acknowledgment of Individual GL 95-05 Performance Criteria

GL 95-05 Item	GL Methodology Concurrence	Comments
3.c.3	Modified, accepted by NRC	Limits on re-inspection of tubes due to out of specification probe wear will be followed according to the NRC letter dated February 9, 1996.
3.c.4	Followed	Data analysts will be trained and qualified in the use of the analysis guidelines and procedures specific for application of the criteria
3.c.5	Followed	Data analysts will use quantitative noise criteria guidelines in the evaluation of the data. However, it is expected that these criteria will be evolving over the inspection and as a result, are subject to change. Data failing to meet these criteria should be rejected, and the tube will be re-inspected.
3.c.6	Followed	TVA data analysts will review the mixed residuals on the standard itself and take action as required in the TVA analysis guidelines.
3.c.7	Followed	TVA will use 0.610 inch diameter bobbin probes. Prior to TVA's use of a different bobbin probe size, TVA will demonstrate (on a plant specific or generic basis) that probes and procedures will provide (on a statistically significant basis) equivalent voltage response and detection capability when compared to the 0.610 inch diameter bobbin probe.
3.c.8	Followed	Data analysts will be trained on the potential for PWSCC to occur at TSP intersections and sensitized to identifying indications attributable to PWSCC.
4.	Followed	TVA will follow the industry guidance in NP-7480-L, Addenda 3 (May 1999), "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1999 Database Update," EPRI, Palo Alto, CA..
5.a	Followed	Operational leakage LCO 3.4.13 will be reduced to 600gpd total in all SGs and 150 gpd in any one SG
5.b	Followed	TVA leakage monitoring techniques are consistent with EPRI Report TR-104788, "Primary-to-Secondary Leak Guidelines", and are adequate to meet GL 95-05 recommendations.
5.c	Followed	Known leaking tubes will be repaired.
6.	Followed	Reporting requirements will be followed.

Table 2

Number of TSP Intersections with $\Delta D > 0.030$ inch, subsequently excluded from application of the voltage based plugging criteria.

Watts Bar Unit 1

Note: Model D3 SGs nominally have 4674 tubes, therefore, the maximum number of tube support holes at a plate elevation is 9348 (4674 in each of the hot leg and cold leg portions of the plates).

Westinghouse TSP Designation	Watts Bar 1 TSP Designation	Number of Affected Intersections	Number of Intersections in TSP	% Affected Intersections
A (FDB)	H01	0	4674	0 %
B (FDB)	C01	0	< 4674 [1.]	0 %
C	H02	0	4674	0 %
D	C02	0	4674	0 %
E	C03	6	4354 [2.]	0.14 %
F	C04	6	4354	0.14 %
G	H03	6	4674	0.13 %
H	C05	6	4354	0.14 %
J	C06	20	4354	0.46 %
K	C07	20	4354	0.46 %
L	H04	20	4674	0.43 %
M	C08	20	4354	0.46 %
N	C09	20	4354	0.46 %
P	C10	20	4674	0.43 %
Q	H05, C11	18	9348 [3.]	0.19 %
R	H06, C12	18	9348	0.19 %
S	H07, C13	30	9348	0.32 %
T	H08, C14	256	9348	2.74 %

1. The FDB has a small crescent shaped center cutout region in the cold leg side of the baffle, therefore, the number of intersections is less than twice the tube count.
2. Number of tube holes in plates D, E, F, H, J, K, M, and N have less than 4674 intersections to allow the flow to turn in the pre-heater section.
3. Plates Q, R, S and T are full plates, represented by a 360° circumference. All other plates are half-plates.

ENCLOSURE 1
ATTACHMENT 1

WATTS BAR NUCLEAR PLANT UNIT 1
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

It has been previously established that a postulated main steam line break (MSLB) outside of containment but upstream of the main steam isolation valves (MSIV) represents the most limiting radiological condition relative to the alternate voltage based repair criteria for axial ODSCC.

A calculations has been performed by TVA to determine the maximum permissible steam generator primary to secondary leak rate during a steam line break for WBN Unit 1. The calculation determined that 10 gallons per minute (at standard temperature and pressure) primary to secondary leakage (assuming RCS dose equivalent iodine concentration of 1.0 μCi/gm) in the faulted steam generator would result in site boundary doses within 10 CFR 100 guidelines and control room doses within the 10 CFR 50, Appendix A, General Design Criteria (GDC)-19 limit. The establishment of the 10 gpm value is controlled by the 0 to 2 hour offsite doses at the site boundary, not the control room dose.

The calculation used TVA computer codes STP, FENCDOSE, and COROD. The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which determines 30-day and 2-hour low population zone (LPZ) offsite dose).

Assumptions/Inputs for one year prior to the accident

1. The primary side to secondary side leakage of 150 gpd per steam generator has remained constant for one year in order to maximize the radionuclide inventory in the secondary side.

2. The primary side activity consists of expected radionuclide activity levels for reactor coolant with 0.12 percent failed fuel multiplied by a factor of 8 to bring the levels up the 1 percent failed fuel design limit. Converting the iodine concentrations to I-131 dose equivalence results in >1 μCi/gm (1.01E+00 μCi/gm), which is the Technical Specification maximum limit for reactor coolant system (RCS).

3. The average blowdown of 100 gpm is used for the pre-accident one year period.

Assumptions/Inputs during and post accident with 10 gpm (at standard temperature and pressure) primary to secondary leakage used to calculate doses.

1. It is assumed that the primary-to-secondary leakage mass release to the environment is 10 gpm (at standard temperature and pressure).

2. Pre-Accident Iodine Spike of 60 μCi/gm I-131 equivalent in the RCS plus an accident initiated iodine spike which increases the iodine release rate to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium into the reactor coolant from the fuel rods.

3. Steam released to the atmosphere
    1. total from the non-defective steam generators (0 - 2 hr), 480,000 lb
    2. total from the non-defective steam generators (2 hr - 8 hr), 871,000 lb
    3. total from the faulted steam generator (0 - 30 min), 152,509 lb (150,000 lb SG sec inventory + 2509 lb primary-to-secondary leakage)
    4. total from the faulted steam generator (30 min - 8 hr), 28,600 lb (primary-to-secondary leakage)

4. Iodine partition coefficients for steaming of SG water
    1. non-defective steam generators (initial inventory), 0.01
    2. non-defective steam generators primary-to-secondary leakage, 1.0
    3. faulted steam generator (initial inventory and primary-to-secondary leakage), 1.0

5. No Condenser Vacuum release during the accident

6. After 8 hours following the accident, no steam and activity are released to the environment.

7. No noble gas is dissolved or contained in the steam generator water, i.e. all noble gas leakage to the secondary system is continuously released with steam from the steam generators.

8. The 0-2 and 2-8 hour accident atmospheric dilution factors given in WBN Unit 1 FSAR Appendix 15A and the 0-8 hour breathing rate of 3.47×10^{-4} m^3/sec are applicable. Doses are based on the dose models presented in WBN Unit 1 FSAR Appendix 15A.

Main Steam Line Break Dose Calculation Results

10 gpm Primary-to-Secondary leakage	Control Room Operator (rem)	SRP Guidance for GDC 19 Limits(rem)	30-Day LPZ (rem)	2-Hour EAB (Site boundary) (rem)	SRP Guidance for 10 CFR 100 Limits (rem)
Gamma:	0.06789	5	0.05911	0.1474	2.5
Beta:	0.7662	30	0.0377	0.0781	30
Inhalation:	10.13	30	8.618	14.129	30

ENCLOSURE 1
ATTACHMENT 2

EXCERPTS FROM
WESTINGHOUSE WAT-D-10709
JANUARY 12, 2000
APPLICATION FOR IMPLEMENTATION OF VOLTAGE BASED REPAIR CRITERIA WESTINGHOUSE STEAM GENERATOR TUBES AFFECTED BY ODSCC AT TSPs

WAT-D-10709, ATTACHMENT 2

TUBES EXCLUDED FROM ARC FOR WATTS BAR UNIT 1
DUE TO TUBE DEFORMATION UNDER
COMBINED LOCA + SSE LOADS

WAT-D-10709 EXCERPT, ATTACHMENT 2

TUBES EXCLUDED FROM ARC FOR WATTS BAR UNIT 1 DUE TO TUBE DEFORMATION UNDER COMBINED LOCA + SSE LOADS

1.0 Introduction

The Tennessee Valley Authority contracted with Westinghouse to develop an alternate repair criteria (ARC) in the event they discover axial ODSCC at tube support plate locations for Watts Bar Unit 1. The application of the ARC will allow the utility to leave in service some tubes that may have through wall crack indications at TSP locations. One concern with the ARC, however, is the potential for in-leakage following a postulated large LOCA+SSE[1] loading.

When the tube bundle is subject to lateral (in-plane) loads, the overall TSP load is transferred to the steam generator shell through wedge groups located at discrete locations around the plate circumference. The number of wedge groups varies in number, size, and orientation among the various plates. For the combined SSE + LOCA loading condition, the potential exists for yielding of the tube support plate in the vicinity of the wedge groups, accompanied by deformation of tubes and a subsequent postulated in-leakage. In-leakage may occur if axial cracks are present and propagate through wall as tube deformation occurs. This deformation may also lead to opening of pre-existing tight through-wall cracks, resulting in primary to secondary leakage during the SSE + LOCA event, with consequent in-leakage following the event. In-leakage is a potential concern, as a small amount of leakage may cause an unacceptable increase in the core peak clad temperature. Thus, any tubes that are defined to be potentially susceptible to significant deformation under SSE + LOCA loads are excluded from consideration under alternate voltage based repair criteria.

2.0 Tube Bundle Geometry

The overall TSP load is transferred to the steam generator shell through wedge groups located at

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1 LOCA is an abbreviation for Loss Of Coolant Accident, and SSE is an abbreviation for Safe Shutdown Earthquake.

discrete locations around the plate circumference. The geometry of the tube bundle region for the Watts Bar (Model D3) steam generators is shown in Figure 1. Referring to Figure 1, for the Model D3 steam generators, there is a flow distribution baffle at the bottom of the bundle, Plates H01/C01, seven tube support plates, Plates H02 through H08/C11 through C14, and nine flow baffles, baffles C02 through C10.

3.0 Applied Loads

3.1 Seismic Loads

TSP loads from a plant specific seismic analysis for the Watts Bar Plant were used in the analysis. Loads were not provided for several of the flow baffles (C03, C04, C06, C07, C09, and C10), thus these baffles were not included in the seismic model due to the added complexity of representing them in the model, and also because it is conservative in terms of tube stresses to exclude them from the model. With them in the model, it is anticipated that contact impact loads for the lower cold leg plates would be distributed among the various plates and baffles resulting in reduced loads. However, it is difficult to estimate these loads, due to the flexible nature of the partition plate that forms one of the support members for these plates. For calculating the number of deformed tubes, it is conservatively assumed that the loads for the baffles included in the model also apply to the baffles in the immediate vicinity of the modeled baffle. In other words the loads for baffle C05 are assumed to apply to C03 and C04 also, see Figure 1, and the loads for baffle C08 are also assumed to apply to baffles C06, C07, C09, and C10.

3.2 LOCA Loads

For a LOCA event, the tubes are subject to both a rarefaction pressure wave that travels through the tube bundle, and to loads resulting from shaking of the overall steam generator. The loads resulting from the LOCA rarefaction wave are taken from an analysis of a large break LOCA event for a plant with Model D steam generators. Watts Bar Unit 1 is qualified for leak-before-break for the primary piping, and thus small break LOCA becomes limiting for the steam generators. However, small break LOCA loads are not available, and the large break LOCA loads form a bounding set of loads. LOCA shaking loads are also taken from a plant specific analysis for the Watts Bar units; LOCA shaking loads for the cold leg baffles were estimated in the same manner as for the seismic analysis.

3.3 Combined Loads

In calculating the combined TSP load, the LOCA rarefaction and LOCA shaking loads are combined directly, while the LOCA and SSE loads are combined using the square root of the sum of the squares.

4.0 Acceptance Criteria

In estimating the number of deformed tubes, the results of TSP crush tests for Model D steam generators are used. The deformation criteria for establishing a tube as being susceptible to in-leakage is defined to be 0.030 inch. In reporting the crush test results, tube deformations are reported for various deformation magnitudes. This is the smallest deformation reported. Although test data are not available for leak rate as a function of tube deformation, it is judged that deformation levels less than this magnitude will not result in significant leakage.

5.0 Plate Deformation Characteristics

The deformation characteristics of the TSP are estimated using crush test results for Model D plates. The deformation characteristic of interest for this analysis is the change in tube diameter as a function of load and the number of tubes affected. The tube deformation data, however, is provided only at the conclusion of the test. A record is provided, however, of the force/deflection characteristics of the plate, and it is logical to assume that the change in tube diameter and number of affected tubes are proportional to the plate deflection for a given load level.

The force/deflection results demonstrate that the behavior of the plate and tubes is not elastic. In order to make use of the data, an approximation was made between the elastic analyses that determine the plate loads, and the inelastic crush test results. The approximation was based on the area under the force/-deflection curve for the crush test versus the area corresponding to the elastic plate response. Calculations were performed to relate the area under the force/deflection curve to an equivalent elastic force.

Using the approach discussed above, where the tube diameter change is assumed to be proportional to the deflection of the wedge for a given load level, calculations were performed to determine the number of tubes with this deflection level as a function of load.

It should be noted the Model D crush tests used a typical tube support plate with both tube and flow holes. For the cold leg flow baffles, Plates C03 through C10, there are no flow holes. Thus, these plates are significantly stronger than the crush test plates. However, in the absence of test data for these plates, the data for the plates with flow holes was conservatively used to predict the number of affected tubes for these plates as well. Flow holes are also not present for the flow distribution baffle, Plates H01/C01, and baffle Plate C02. The tube holes in these plates are larger than the nominal plate tube holes. However, the net metal area adjacent to a tube hole is still smallest for the nominal TSP (because of the flow holes), and the use of the crush test data for the nominal TSP geometry is conservative for these plates as well.

6.0 Estimate of Wedge Group Loads

The TSP loads are reacted by wedge groups located at their periphery. The number of wedge groups varies in number, size, and orientation among the various plates. Two different wedge group sizes were employed in the fabrication of the Model D3 steam generators. The wedge group size is important, because it affects the local distribution of load into the neighboring tubes. The TSP crush tests' results that were used to approximate the number of affected tubes used a wedge width which was matched or was conservative relative to the actual steam generator wedge widths.

In reacting the load among the various wedge groups, a cosine distribution is assumed among the wedges that are loaded. Typically, only half of the wedge groups are loaded at any given time. In determining the distribution of load for seismic and LOCA loads, the directionality of the load is considered. LOCA rarefaction loads are uni-directional, in that they only act in the plane of the U-bend. Seismic and LOCA shaking loads on the other hand are random, and can act in any direction.

7.0 Estimated Number of Deformed Tubes

Applying the wedge load factors to the plate loads gives corresponding loads at each of the wedge locations. Then, using the approximation for the number of deformed tubes as a function of load, the number of deformed tubes at each wedge position was calculated. Note that the maximum plate load for Plates H01/C01 is less than the load necessary to cause any permanent tube deformation. Thus, there will be no permanently deformed tubes for these plates. An overall summary of the number of affected tubes is provided in Table 1.

8.0 Tube Maps/Summary Tables

Watts Bar Unit 1 is a four-loop plants. As such, there are two loops with "left-hand" steam generators and two loops with "right-hand" steam generators. These designations refer to the orientation of the nozzles and manways on the channel head. For the purpose of this analysis, "left-hand" units are defined to be those loops where the primary fluid flows from the reactor to the steam generator to the pump and back to the reactor vessel in a counter-clockwise direction when viewed from above. Conversely, for the "right-hand" units, the flow is in the clockwise direction. The left- versus right-hand designation affects the location of the nozzles and manways, and the manner in which the columns are numbered for tube identification purposes. Reference configurations used in identifying wedge locations are shown in Figures 2 and 3 for the left-hand and right-hand units, respectively. As shown in the figures, for left-hand units, the nozzle and tube column 1 are located at 0°, while for right-hand units they are located at 180°. Note the quadrant designations on these plots, as they are used in the figures defining the locations of affected tubes.

Maps showing the location of the tubes excluded from the ARC are provided in Figures 4 through 15. The maps provide row and column designations relative to the left-hand units. Column numbers for the right-hand units are shown in brackets. Identification of the potentially susceptible tubes is based on the crush test results for Model D steam generators. In performing the crush tests, wedge/tube configurations identical to those for Watts Bar Unit 1 steam generators were not tested. As such, it is not possible to identify exactly the tubes that might be limiting at each wedge group. Thus, due to the uncertainties involved, there are more tubes identified at each wedge group as being limiting than identified in Table 1.

Tabular summaries of the tubes excluded from the ARC are summarized in Tables 2-6 for the left-hand units (Steam Generators 2 and 4), and in Tables 7-11 for the right-hand units (Steam Generators 1 and 3).

9.0 Tubes Adjacent to the Partition Plate

With the presence of the preheater, there is also the potential for deformation of the plates (and tubes) adjacent to the points of load transfer across the partition plate. The load transfer varies for the different pre-heater plates. In general, however, the load transfer is either across "tabs" that span cutouts in the plates adjacent to the partition plate, or across wedges than span between the partition plate and the edge of the preheater plates. In both cases, the tabs/wedges located more towards plate center will support the largest percentage of the load.

Recognizing that the plate loads that are obtained from the dynamic seismic and LOCA shaking analyses represent the load of the full bundle at any given plate location, the load that is transferred across the partition plate is one-half of that total. The largest plate load in the preheater region, for Plates H05/C08, was conservatively assumed to be reacted by only two of the wedges/tabs to transfer the load across the partition plate. The plate crush tests demonstrated that no tubes would be predicted to experience a ΔD greater than 0.030" for that load. Thus, there are no tubes to be excluded from the ARC located along the partition plate.

Table 1
Overall Summary of Number of Tubes with ΔD > 0.030 inch
Watts Bar Unit 1

Plate	Wedge Position[(1)] (deg)	Number fo Deformed Tubes	Total For the Plate
H01 / C01	No Tubes Affected		
H02 / C02	No Tubes Affected		
H03 / C03, C04, C05[(2)]	-56.	2	6
	-10.	1	
	45.	0	
	78.	0	
H04 / C06, C07, C08, C09, C10[(3)]	-56.	4	20
	-10.	1	
	10.	1	
	60.	4	
H05 / C11[(4)]	0.	3	18
	60.	3	
H06 / C12[(4)]	0.	3	18
	60.	3	
H07 / C13[(4)]	0.	3	30
	60.	6	
H08 / C14[(4)]	0.	6	256
	60.	61	

(1) Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator.

(2) Total For the Plate = 2 x (# @ 45 + # @ 78 + # @ -56 + # @ -10)

(3) Total For the Plate = 2 x (# @ 10 + # @ 60 + # @ -56 + # @ -10)

(4) Total For the Plate = 2 x # @ 0 + 4 x # @ 60

Table 2
Summary of Tubes Excluded from ARC
Hot Leg - TSP H03
Cold Leg - TSP C03, C04, C05

Steam Generators 2 and 4 (Left-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
No Tubes Affected			190°	6	113 - 114
				7	112 - 113
				8	112 - 113
				9	112 - 113
				10	112 - 113
				11	112 - 113
				12	112 - 113
			226°	42	93 - 94
				43	91 - 93
				44	90 - 92
				45	89 - 90
				46	89
			304°	42	21 - 22
				43	22 - 24
				44	23 - 25
				45	25 - 26
				46	26
			350°	6	1 - 2
				7	2 - 3
				8	2 - 3
				9	2 - 3
				10	2 - 3
				11	2 - 3
				12	2 - 3

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 3
Summary of Tubes Excluded from ARC
Hot Leg - TSP H04
Cold Leg - TSP C06, C07, C08, C09, C10

Steam Generators 2 and 4 (Left-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
10°	6	1 - 2	190°	6	113 - 114
	7	2 - 3		7	112 - 113
	8	2 - 3		8	112 - 113
	9	2 - 3		9	112 - 113
	10	2 - 3		10	112 - 113
	11	2 - 3		11	112 - 113
	12	2 - 3		12	112 - 113
60°	46	26 - 29	226°	43	90 - 91
	47	28 - 31		44	88 - 92
	48	29 - 32		45	87 - 90
	49	31 - 32		46	86 - 89
120°	46	86 - 89		47	86 - 87
	47	84 - 87		48	86
	48	83 - 86	304°	43	24 - 25
	49	83 - 84		44	23 - 27
170°	6	113 - 114		45	25 - 28
	7	112 - 113		46	26 - 29
	8	112 - 113		47	28 - 29
	9	112 - 113		48	29
	10	112 - 113	350°	6	1 - 2
	11	112 - 113		7	2 - 3
	12	112 - 113		8	2 - 3
				9	2 - 3
				10	2 - 3
				11	2 - 3
				12	2 - 3

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 4
Summary of Tubes Excluded from ARC
Hot Leg - TSP H05, H06
Cold Leg - TSP C11, C12

Steam Generators 2 and 4 (Left-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	1 - 3	180°	6	112 - 114
	2	1 - 3		7	112 - 114
	3	2		8	113
60°	45	27 - 28	240°	45	87 - 88
	46	26 - 30		46	85 - 89
	47	28 - 31		47	84 - 87
	48	29 - 32		48	83 - 86
	49	31 - 32		49	83 - 84
120°	45	87 - 88	300°	45	27 - 28
	46	85 - 89		46	26 - 30
	47	84 - 87		47	28 - 31
	48	83 - 86		48	29 - 32
	49	83 - 84		49	31 - 32
180°	1	112 - 114	350°	6	1 - 3
	2	112 - 114		7	1 - 3
	3	113		8	2

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 5
Summary of Tubes Excluded from ARC
Hot Leg - TSP H07
Cold Leg - TSP C13

Steam Generators 2 and 4 (Left-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	1 - 3	180°	1	112 - 114
	2	1 - 3		2	112 - 114
	3	2		3	113
60°	45	27 - 29	240°	45	86 - 88
	46	26 - 30		46	85 - 89
	47	28 - 32		47	83 - 87
	48	29 - 32		48	83 - 86
	49	31 - 32		49	83 - 84
120°	45	86 - 88	300°	45	27 - 29
	46	85 - 89		46	26 - 30
	47	83 - 87		47	28 - 32
	48	83 - 86		48	29 - 32
	49	83 - 84		49	31 - 32
180°	1	112 - 114	350°	1	1 - 3
	2	112 - 114		2	1 - 3
	3	113		3	2

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 6
Summary of Tubes Excluded from ARC
Hot Leg - TSP H08
Cold Leg - TSP C14

Steam Generators 2 and 4 (Left-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	1 - 3	180°	1	112 - 114
	2	1 - 3		2	112 - 114
	3	2		3	113
60°	41	27 - 32	240°	41	83 - 88
	42	26 - 33		42	82 - 89
	43	26 - 34		43	81 - 89
	44	26 - 35		44	80 - 89
	45	26 - 35		45	80 - 89
	46	26 - 36		46	79 - 89
	47	28 - 36		47	79 - 87
	48	29 - 36		48	79 - 86
	49	31 - 35		49	80 - 84
120°	41	83 - 88	300°	41	27 - 32
	42	82 - 89		42	26 - 33
	43	81 - 89		43	26 - 34
	44	80 - 89		44	26 - 35
	45	80 - 89		45	26 - 35
	46	79 - 89		46	26 - 36
	47	79 - 87		47	28 - 36
	48	79 - 86		48	29 - 36
	49	80 - 84		49	31 - 35
180°	1	112 - 114	350°	1	1 - 3
	2	112 - 114		2	1 - 3
	3	113		3	2

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 7
Summary of Tubes Excluded from ARC
Hot Leg - TSP H03
Cold Leg - TSP C03, C04, C05

Steam Generators 1 and 3 (Right-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
No Tubes Affected			190°	6	1 - 2
				7	2 - 3
				8	2 - 3
				9	2 - 3
				10	2 - 3
				11	2 - 3
				12	2 - 3
			226°	42	21 - 22
				43	22 - 24
				44	23 - 25
				45	25 - 26
				46	26
			304°	42	93 - 94
				43	91 - 93
				44	90 - 92
				45	89 - 90
				46	89
			350°	6	113 - 114
				7	112 - 113
				8	112 - 113
				9	112 - 113
				10	112 - 113
				11	112 - 113
				12	112 - 113

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 8
Summary of Tubes Excluded from ARC
Hot Leg - TSP H04
Cold Leg - TSP C06, C07, C08, C09, C10

Steam Generators 1 and 3 (Right-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
10°	6	113 - 114	190°	6	1 - 2
	7	112 - 113		7	2 - 3
	8	112 - 113		8	2 - 3
	9	112 - 113		9	2 - 3
	10	112 - 113		10	2 - 3
	11	112 - 113		11	2 - 3
	12	112 - 113		12	2 - 3
60°	46	86 - 89	226°	43	24 - 25
	47	84 - 87		44	23 - 27
	48	83 - 86		45	25 - 28
	49	83 - 84		46	26 - 29
120°	46	26 - 29		47	28 - 29
	47	28 - 31		48	29
	48	29 - 32	304°	43	90 - 91
	49	31 - 32		44	88 - 92
170°	6	1 - 2		45	87 - 90
	7	2 - 3		46	86 - 89
	8	2 - 3		47	86 - 87
	9	2 - 3		48	86
	10	2 - 3	350°	6	113 - 114
	11	2 - 3		7	112 - 113
	12	2 - 3		8	112 - 113
				9	112 - 113
				10	112 - 113
				11	112 - 113
				12	112 - 113

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 9
Summary of Tubes Excluded from ARC
Hot Leg - TSP H05, H06
Cold Leg - TSP C11, C12

Steam Generators 1 and 3 (Right-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	112 - 114	180°	6	1 - 3
	2	112 - 114		7	1 - 3
	3	113		8	2
60°	45	87 - 88	240°	45	27 - 28
	46	85 - 89		46	26 - 30
	47	84 - 87		47	28 - 31
	48	83 - 86		48	29 - 32
	49	83 - 84		49	31 - 32
120°	45	27 - 28	300°	45	87 - 88
	46	26 - 30		46	85 - 89
	47	28 - 31		47	84 - 87
	48	29 - 32		48	83 - 86
	49	31 - 32		49	83 - 84
180°	1	1 - 3	350°	6	112 - 114
	2	1 - 3		7	112 - 114
	3	2		8	113

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 10
Summary of Tubes Excluded from ARC
Hot Leg - TSP H07
Cold Leg - TSP C13

Steam Generators 1 and 3 (Right-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	112 - 114	180°	1	1 - 3
	2	112 - 114		2	1 - 3
	3	113		3	2
60°	45	86 - 88	240°	45	27 - 29
	46	85 - 89		46	26 - 30
	47	83 - 87		47	28 - 32
	48	83 - 86		48	29 - 32
	49	83 - 84		49	31 - 32
120°	45	27 - 29	300°	45	86 - 88
	46	26 - 30		46	85 - 89
	47	28 - 32		47	83 - 87
	48	29 - 32		48	83 - 86
	49	31 - 32		49	83 - 84
180°	1	1 - 3	350°	1	112 - 114
	2	1 - 3		2	112 - 114
	3	2		3	113

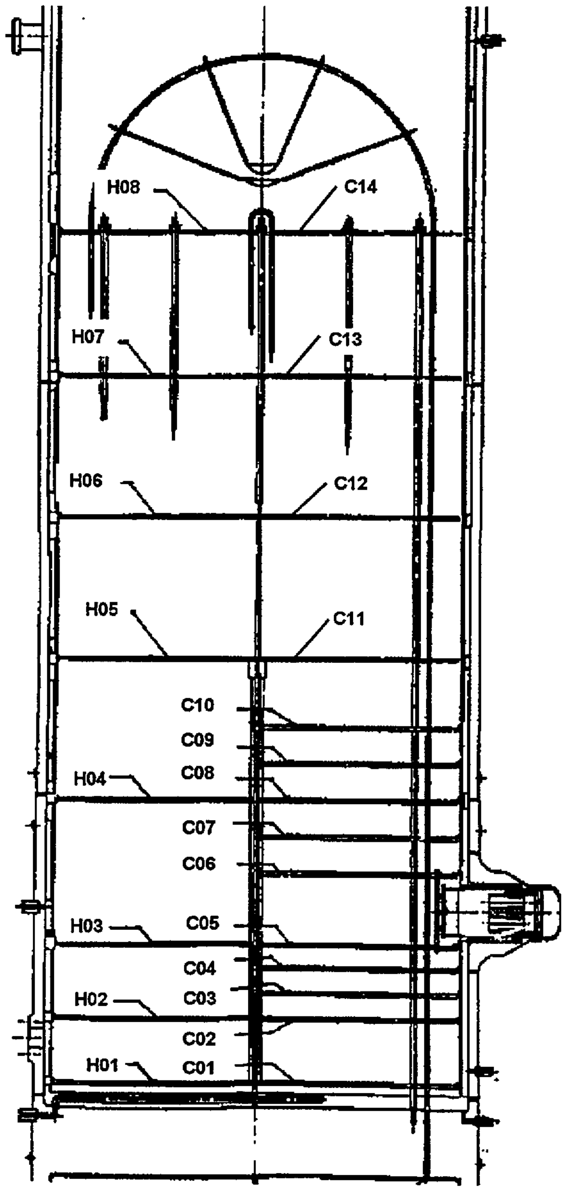
* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator

Table 11
Summary of Tubes Excluded from ARC
Hot Leg - TSP H08
Cold Leg - TSP C14

Steam Generators 1 and 3 (Right-Hand Units)

Hot Leg			Cold Leg		
Wedge Location*	Row	Column	Wedge Location*	Row	Column
0°	1	112 - 114	180°	1	1 - 3
	2	112 - 114		2	1 - 3
	3	113		3	2
60°	41	83 - 88	240°	41	27 - 32
	42	82 - 89		42	26 - 33
	43	81 - 89		43	26 - 34
	44	80 - 89		44	26 - 35
	45	80 - 89		45	26 - 35
	46	79 - 89		46	26 - 36
	47	79 - 87		47	28 - 36
	48	79 - 86		48	29 - 36
	49	80 - 84		49	31 - 35
120°	41	27 - 32	300°	41	83 - 88
	42	26 - 33		42	82 - 89
	43	26 - 34		43	81 - 89
	44	26 - 35		44	80 - 89
	45	26 - 35		45	80 - 89
	46	26 - 36		46	79 - 89
	47	28 - 36		47	79 - 87
	48	29 - 36		48	79 - 86
	49	31 - 35		49	80 - 84
180°	1	1 - 3	350°	1	112 - 114
	2	1 - 3		2	112 - 114
	3	2		3	113

* Angles are measured counter-clockwise with 0° aligned with the three o'clock position looking down on the steam generator



Note: Preheater Modification Not Shown

Figure 1
Tube Bundle Geometry
Watts Bar Unit 1

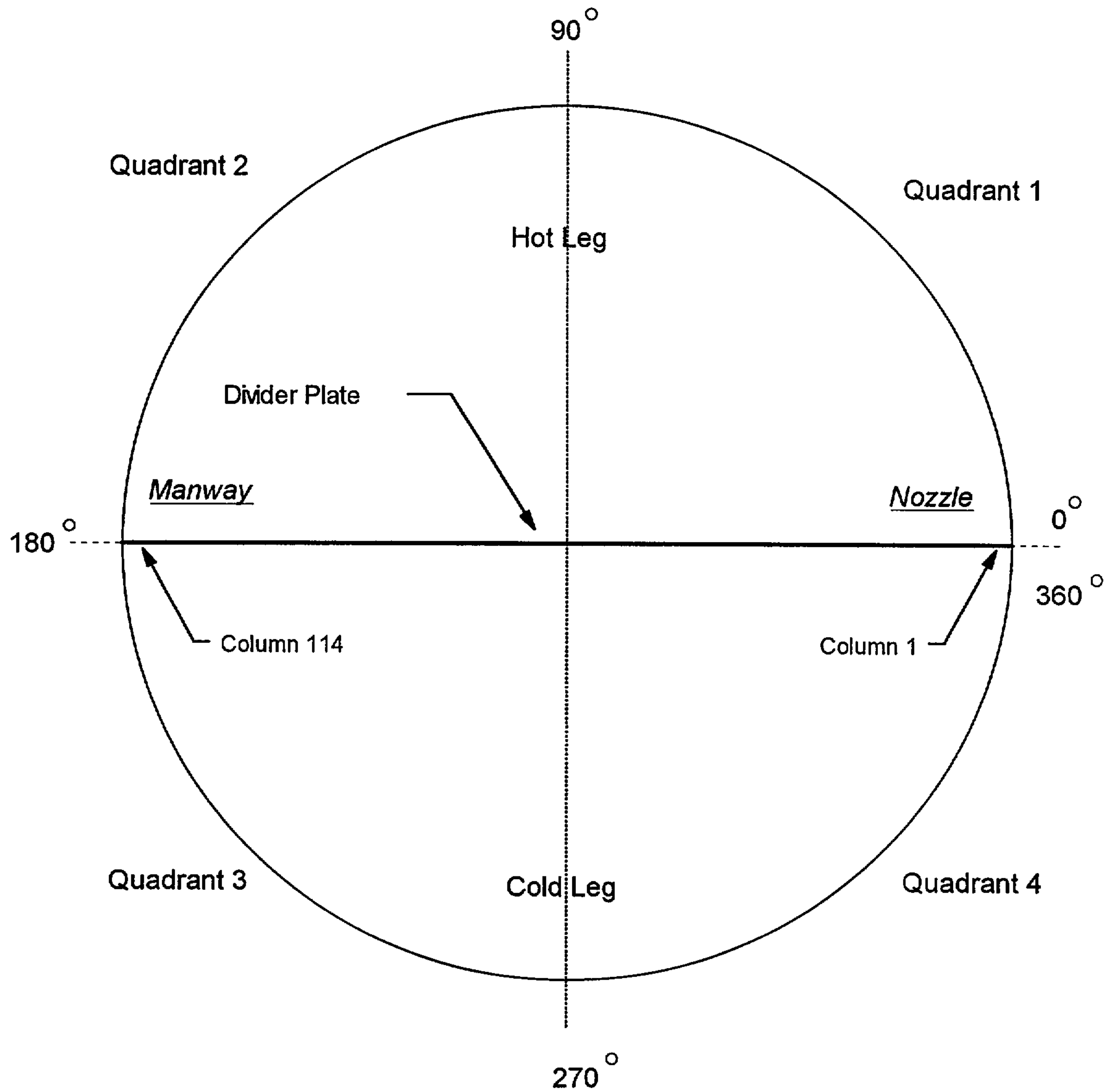


Figure 2
Reference Configuration
Looking Down on Steam Generator
Left-Hand Unit

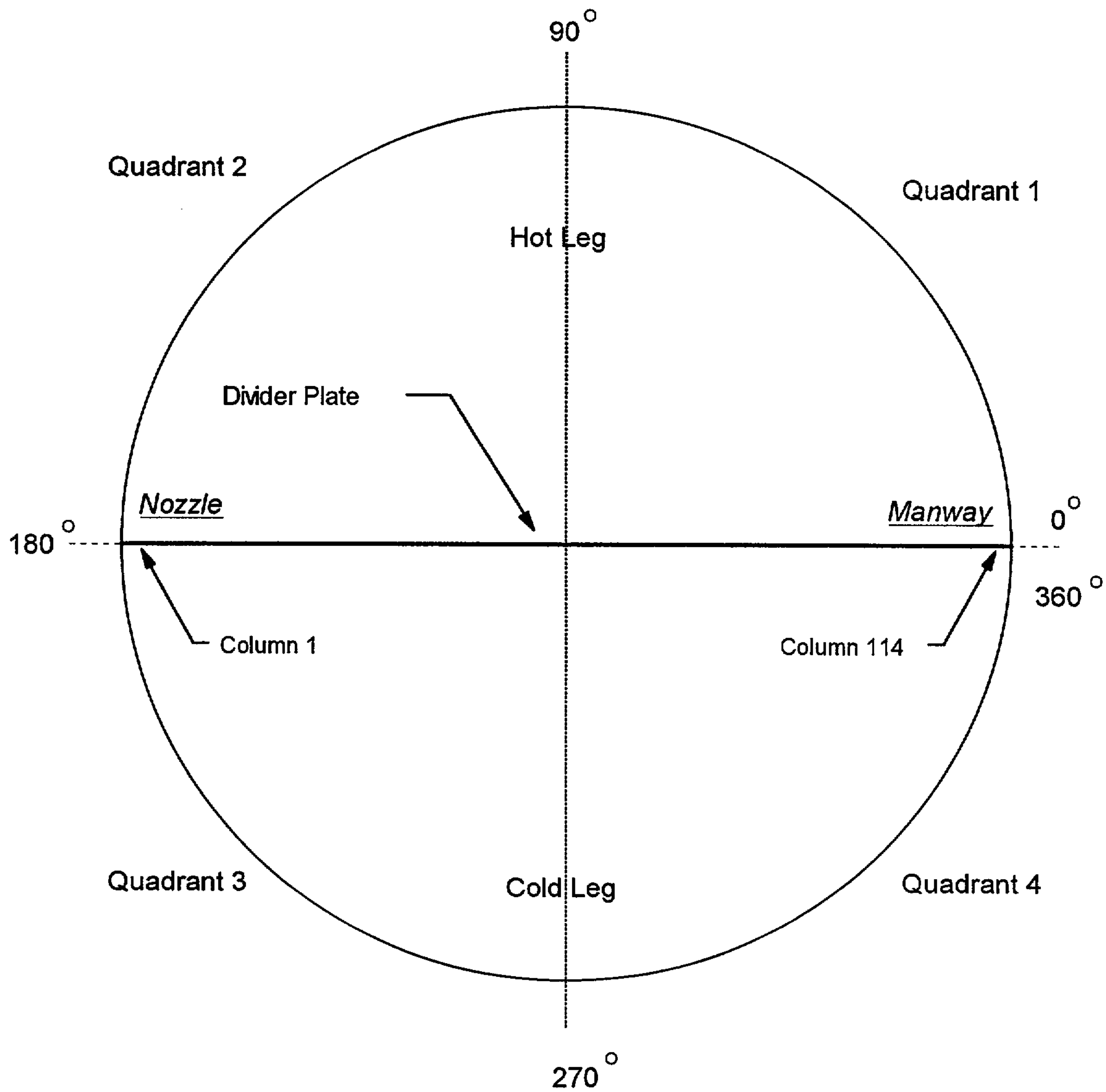


Figure 3
Reference Configuration
Looking Down on Steam Generator
Right-Hand Unit

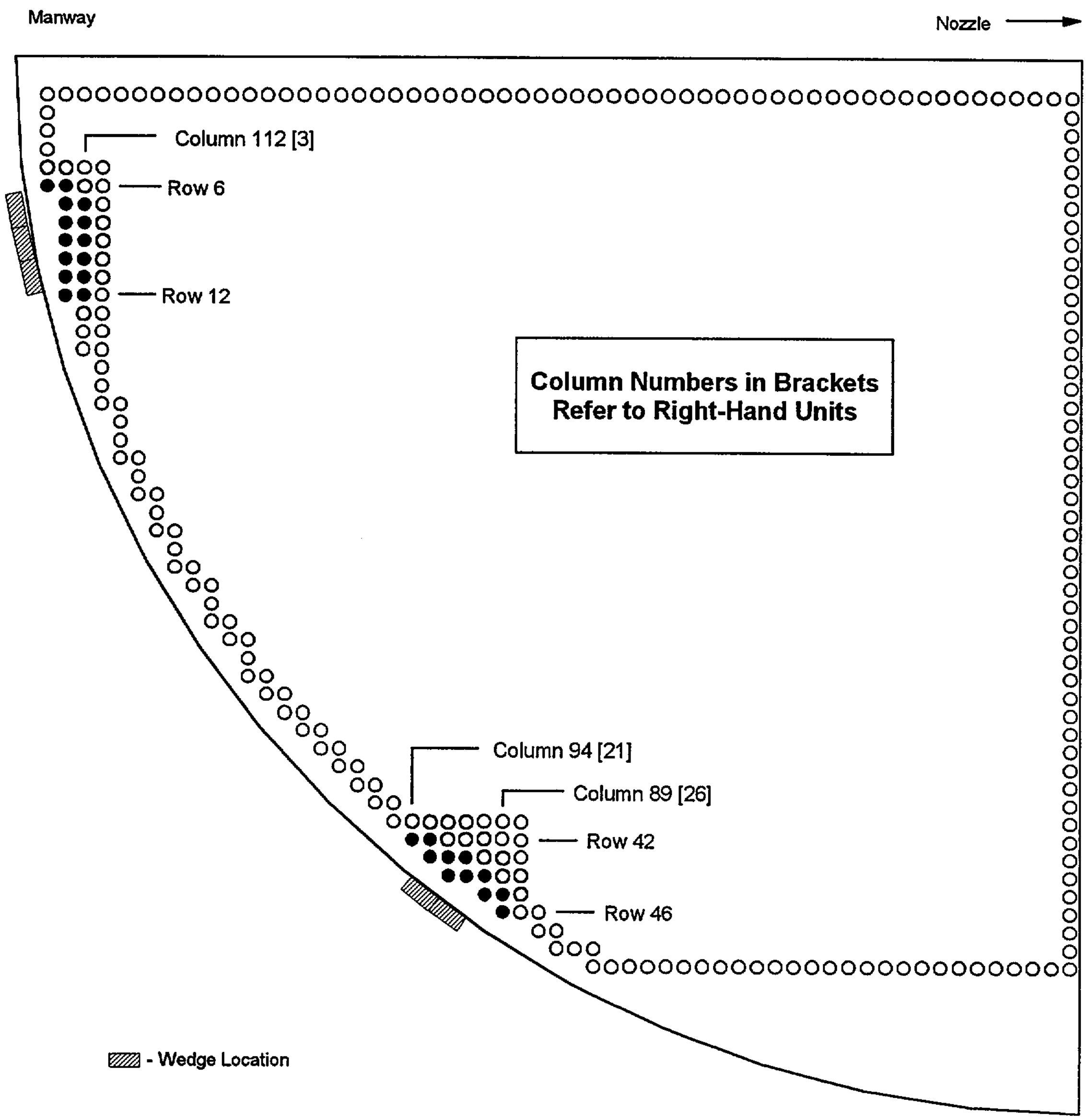


Figure 4
Tubes Excluded from ARC
TSP C03, C04, C05 - Quadrant 3
Left-Hand Unit

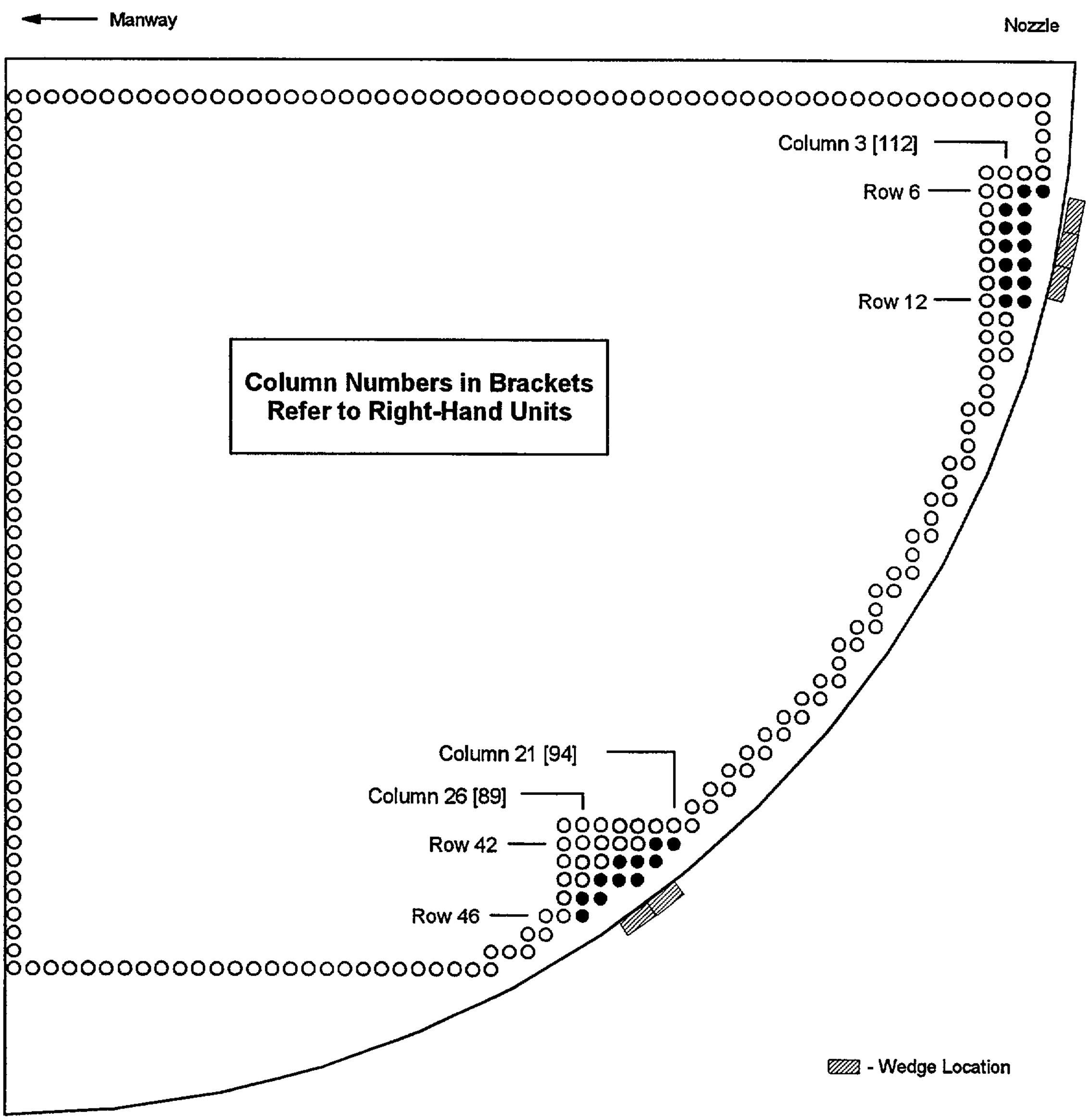


Figure 5
Tubes Excluded from ARC
TSP C03, C04, C05 - Quadrant 4
Left-Hand Unit

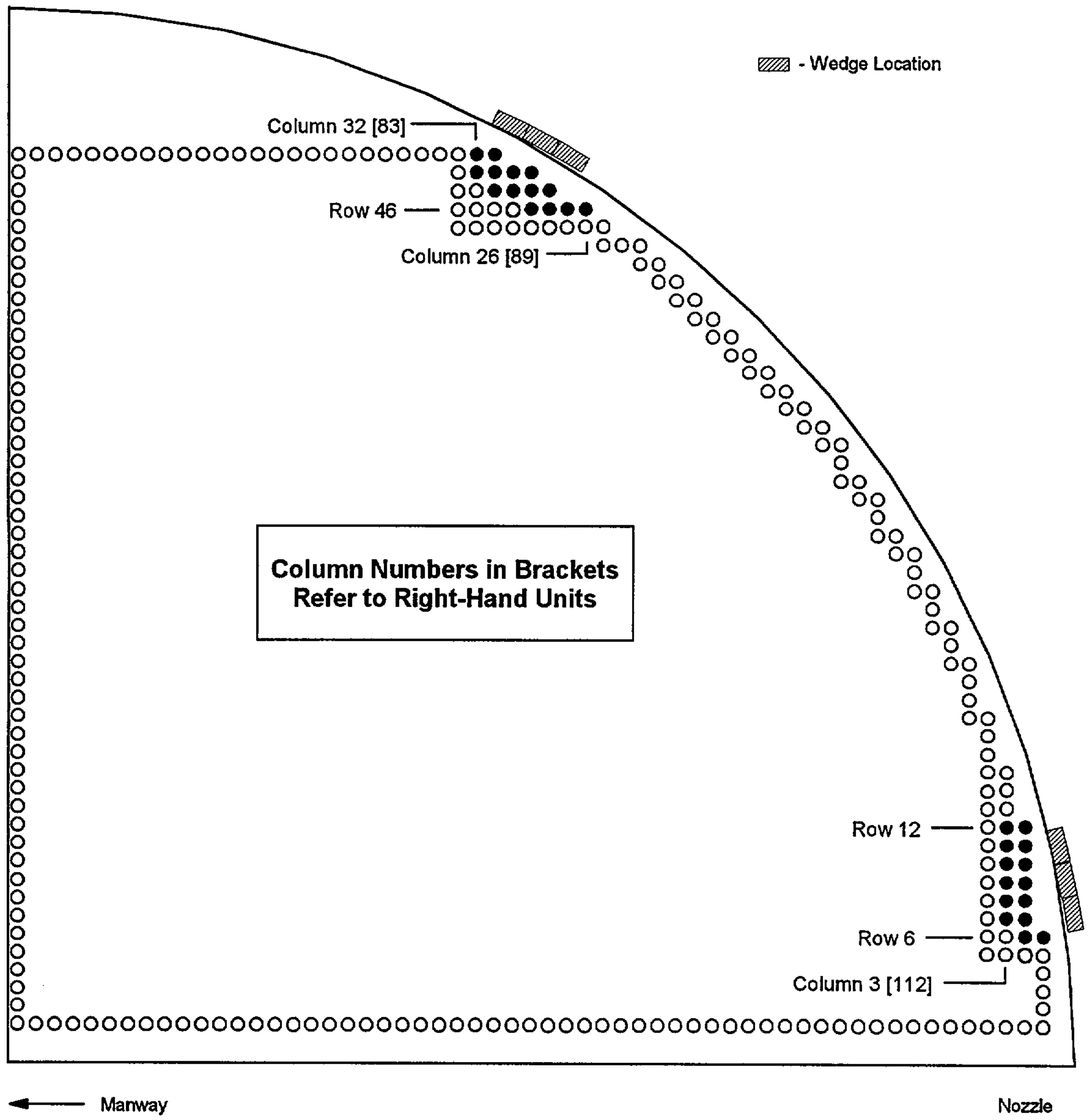


Figure 6
Tubes Excluded from ARC
TSP H04 - Quadrant 1
Left-Hand Unit

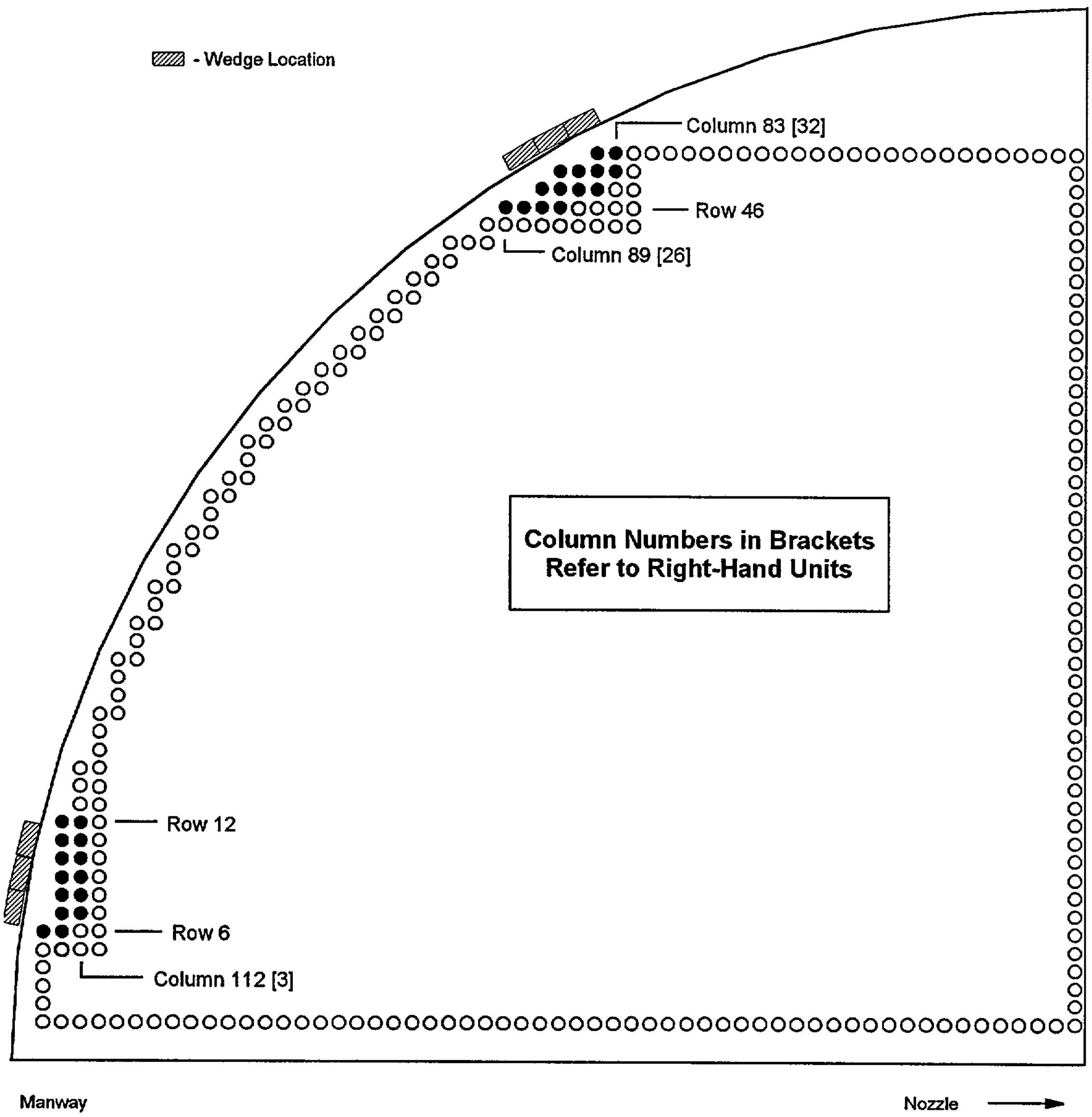


Figure 7
Tubes Excluded from ARC
TSP H04 - Quadrant 2
Left-Hand Unit

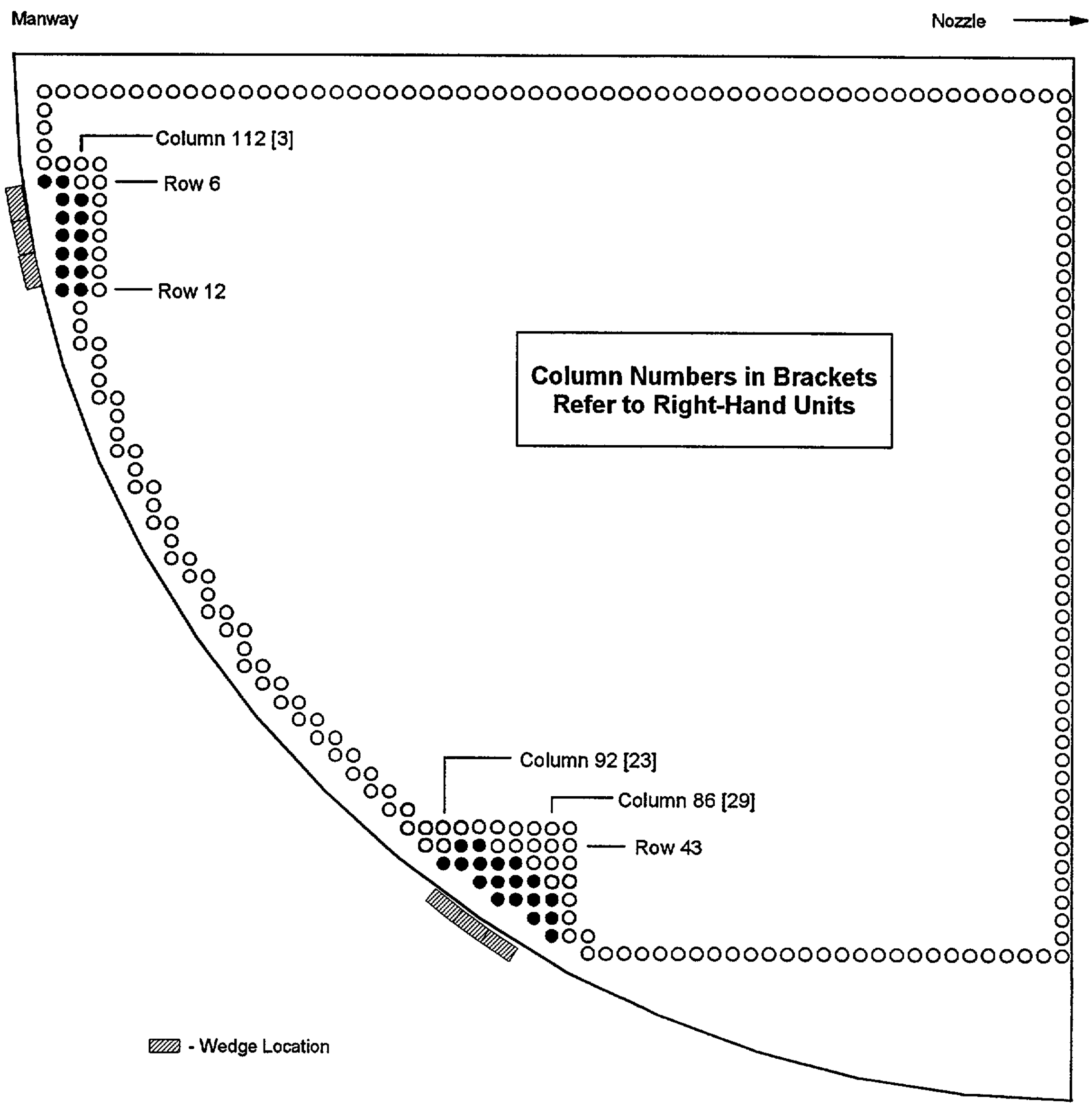


Figure 8
Tubes Excluded from ARC
TSP C06, C07, C08, C09, C10 - Quadrant 3
Left-Hand Unit

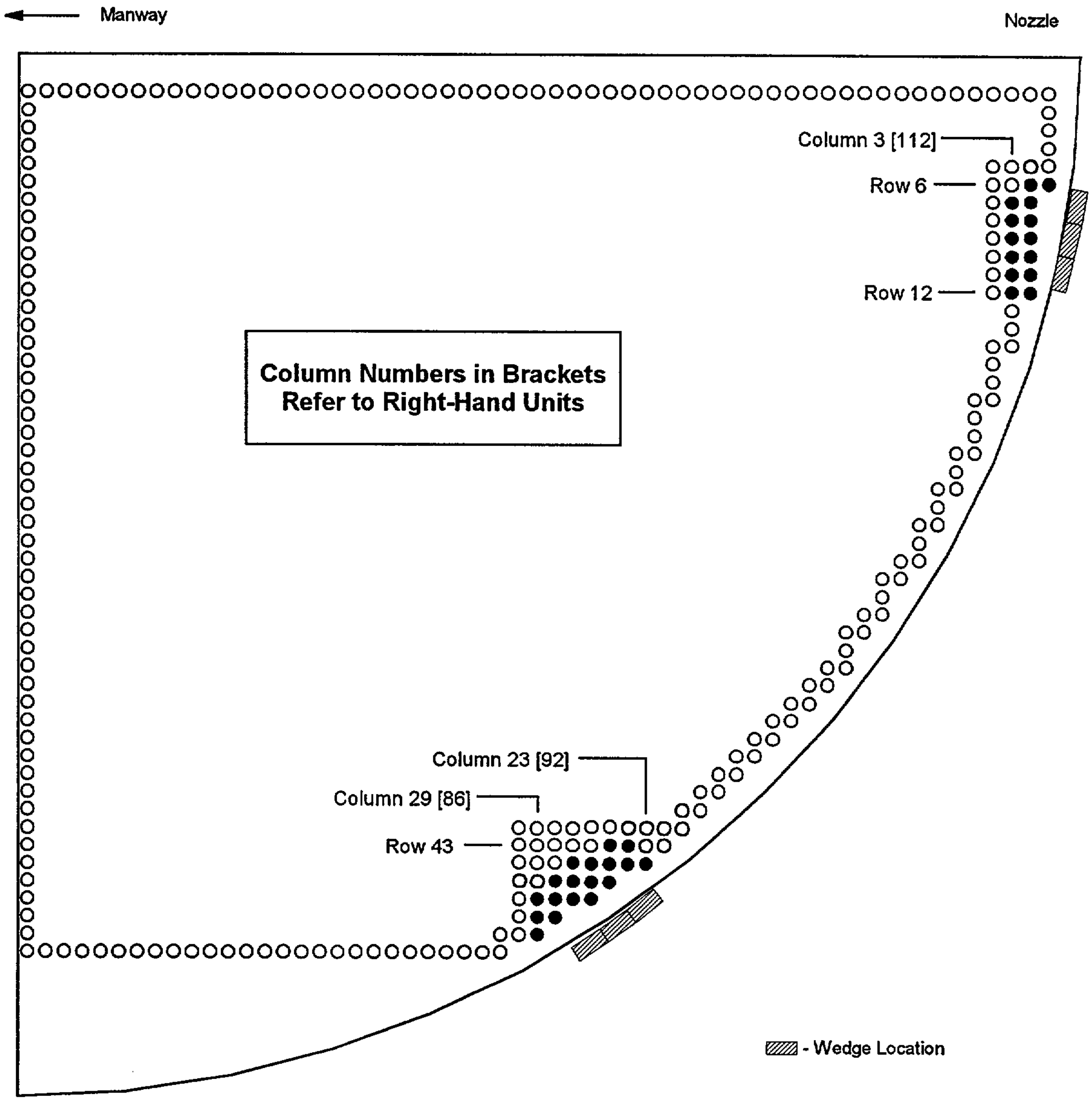


Figure 9
Tubes Excluded from ARC
TSP C06, C07, C08, C09, C10 - Quadrant 4
Left-Hand Unit

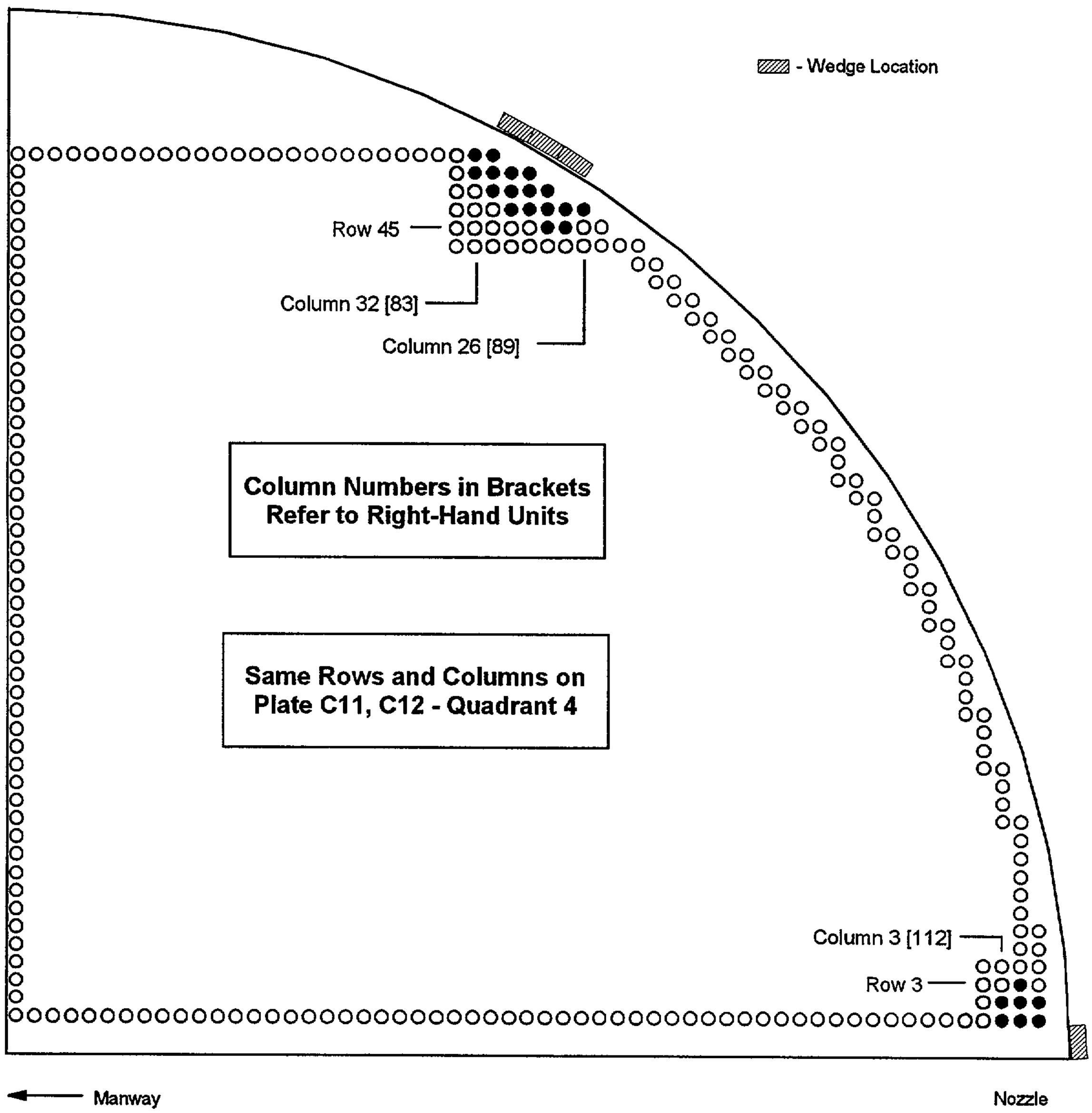


Figure 10
Tubes Excluded from ARC
TSP H05, H06 - Quadrant 1
Left-Hand Unit

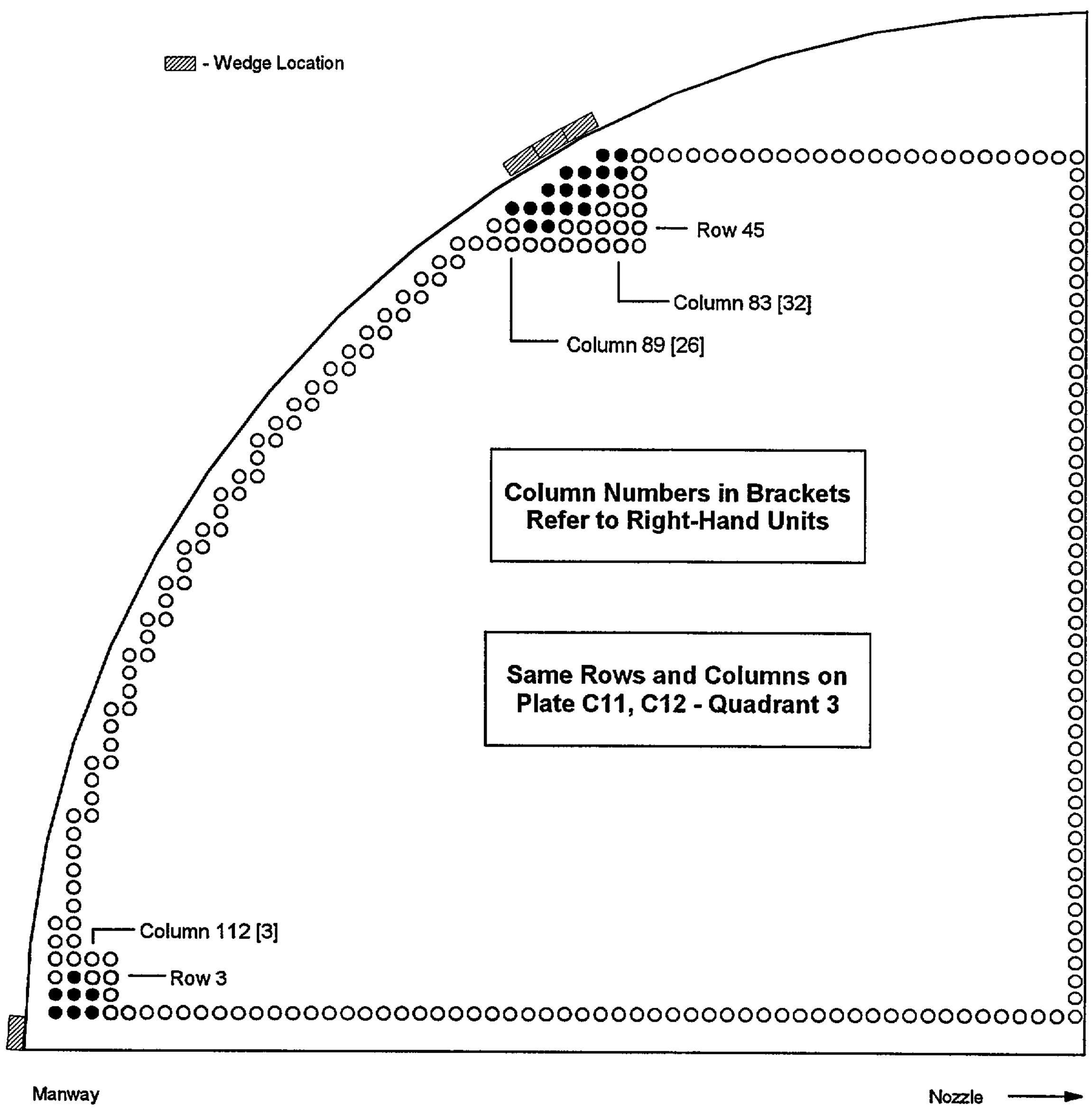


Figure 11
Tubes Excluded from ARC
TSP H05, H06 - Quadrant 2
Left-Hand Unit

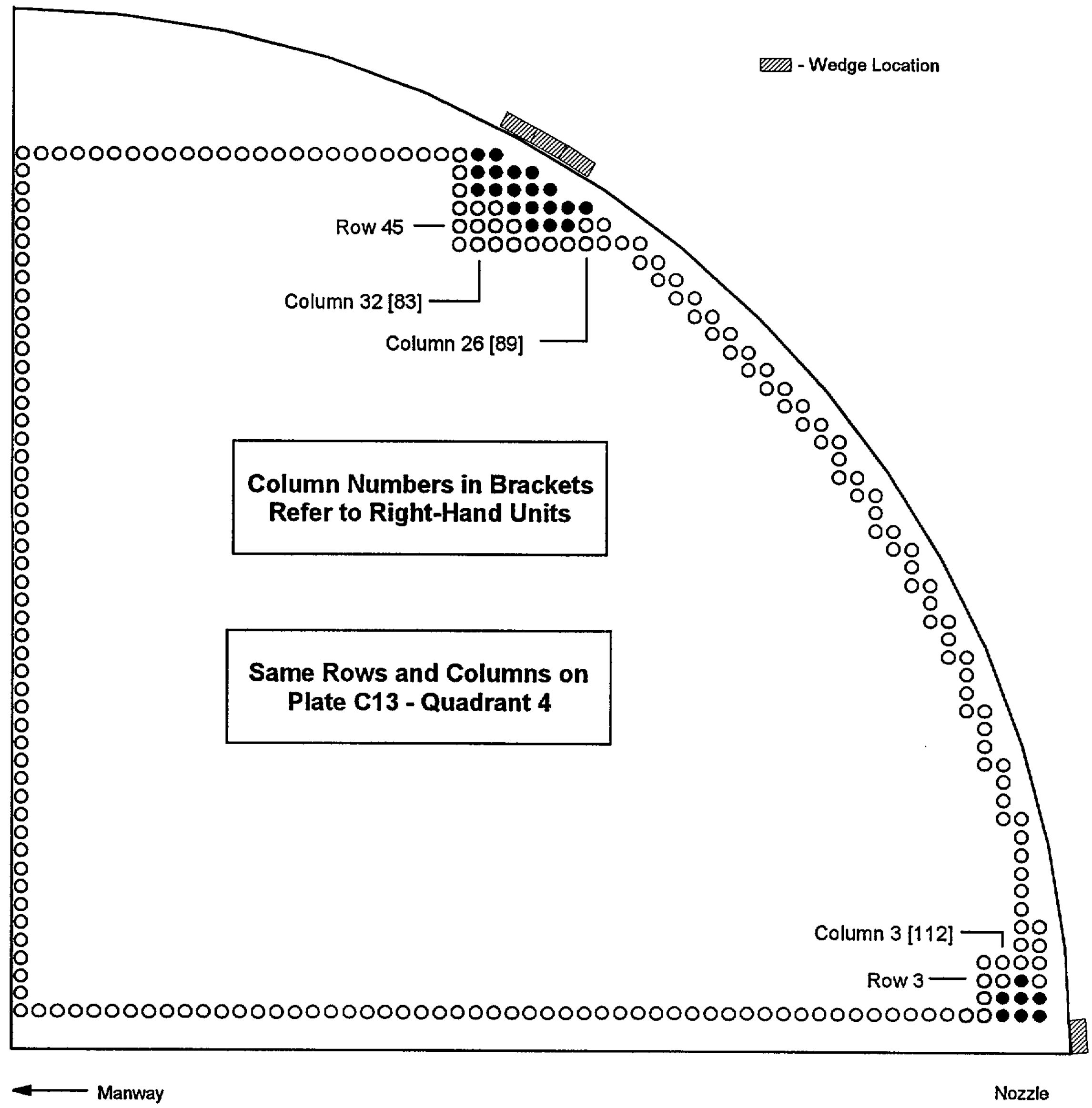


Figure 12
Tubes Excluded from ARC
TSP H07 - Quadrant 1
Left-Hand Unit

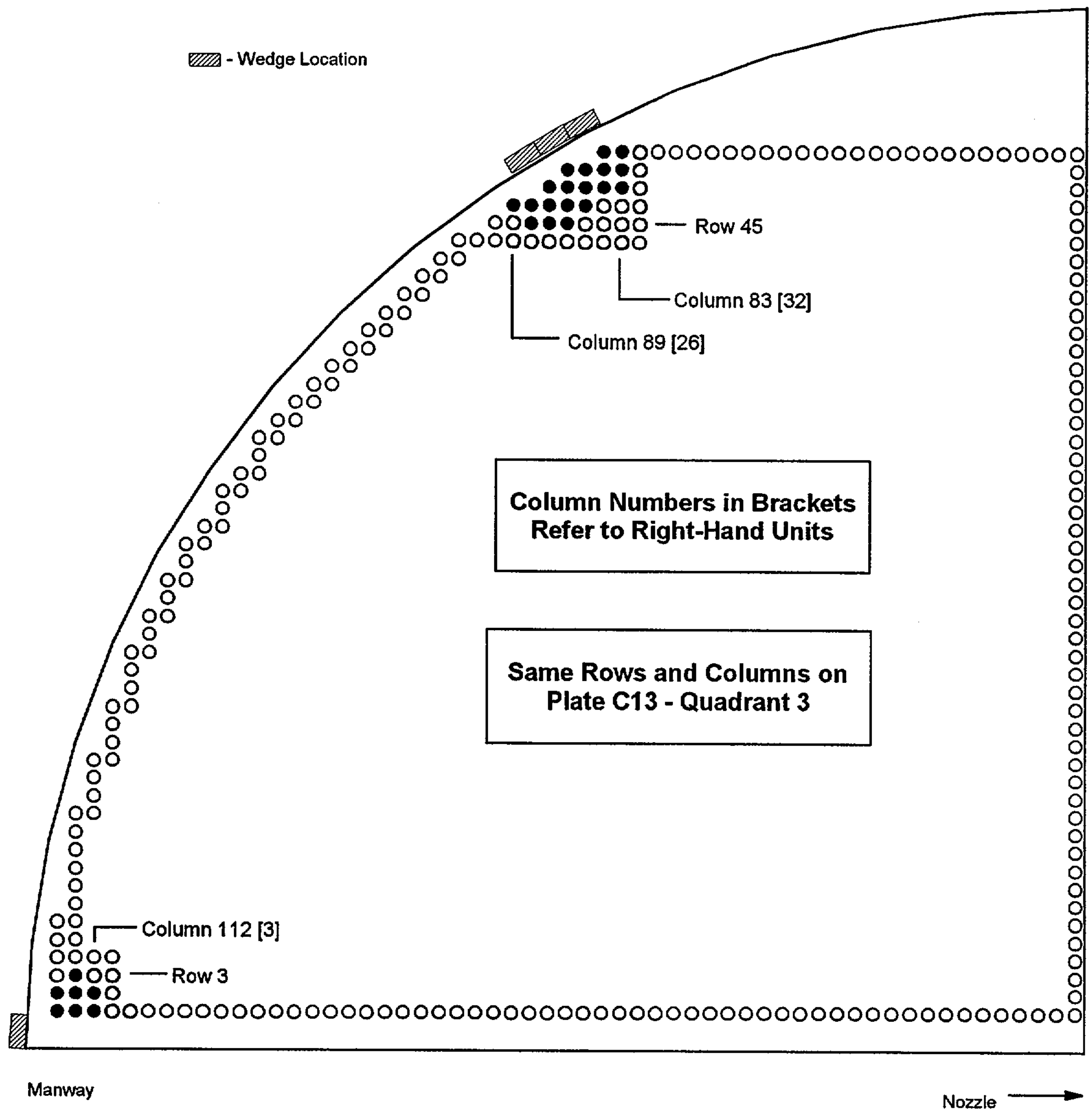


Figure 12
Tubes Excluded from ARC
TSP H07 - Quadrant 2
Left-Hand Unit

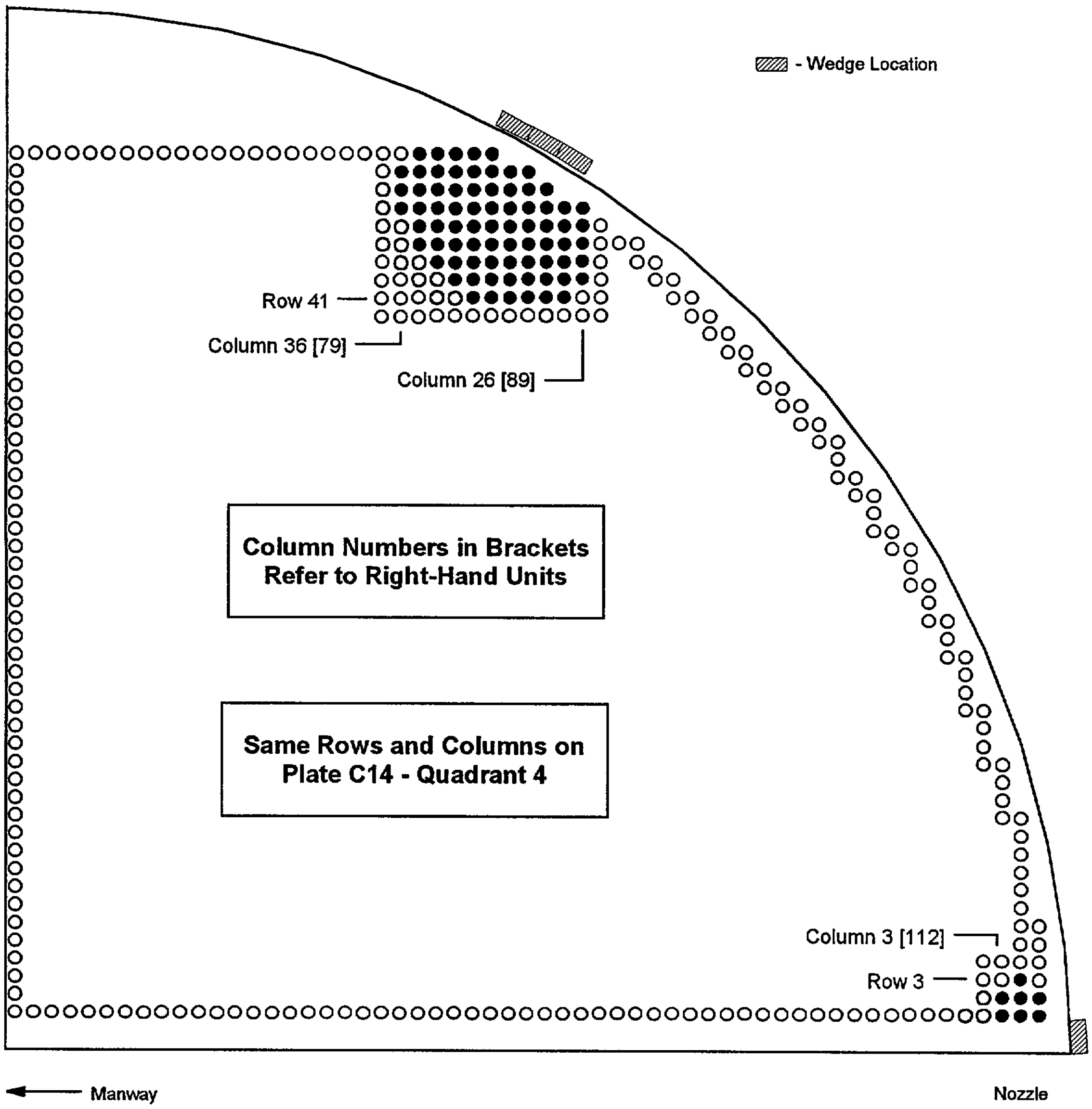


Figure 14
Tubes Excluded from ARC
TSP H08 - Quadrant 1
Left-Hand Unit

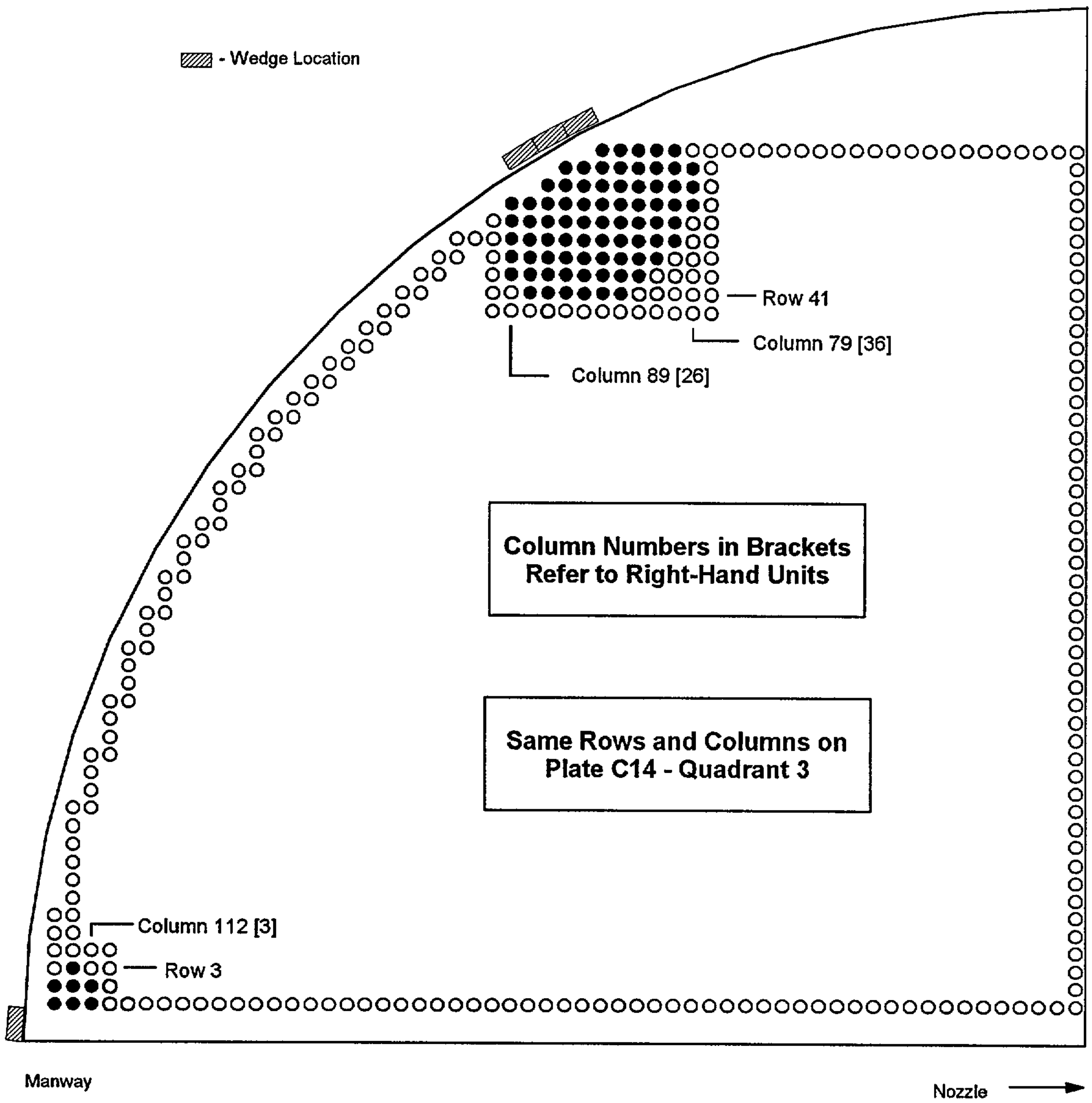


Figure 15
Tubes Excluded from ARC
TSP H08 - Quadrant 2
Left-Hand Unit

ENCLOSURE 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
DOCKET NO.50-390

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-99-14
MARKED PAGES

I. **.AFFECTED PAGE LIST**

Technical Specifications

3.4-30
5.0-15
5.0-16
5.0-17
5.0-18
5.0-19

Technical Specification Bases

B 3.4-74
B 3.4-76

II. MARKED PAGES

See Attached.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

600 gallons per day

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 1 gpm total primary to secondary LEAKAGE through all steam generators (SGs);

150

e. 500 gallons per day primary to secondary LEAKAGE through any one SG; and

f. Four SGs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Steam Generator Tube Surveillance Program not met.	B.1 Determine steam generator tube integrity is acceptable for continued operation	4 hours

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program

Each steam generator (SG) shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the provisions for inservice inspection of ASME Code Class 1, 2, and 3 components which shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a. See Specification 5.7.2.11 for applicable inspection Frequencies.

a. <u>SG Sample Selection and Inspection</u> - Each SG shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of SG tubes specified in Table 5.7.2.12-1.

b. <u>SG Tube Sample Selection and Inspection</u> - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.7.2.12-1. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.7.2.12.e and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.7.2.12.f. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

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g

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from the critical areas;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

b) Tubes in those areas where experience has indicated potential problems, and

DELETE

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

c) A tube inspection (pursuant to Specification 5.7.2.12.~~f~~) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection and

g

ADD

d) Indications left in service as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outage.

c. Examination Results - The results of each sample inspection shall be classified into one of the following three categories:

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 (greater than 10%) further wall penetrations to be included in the above percentage calculations.

d. Supplemental Sampling Requirements - The tubes selected as the second and third samples (if required by Table 5.7.2.12-1) may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

INSERT A

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

f.

e. Inspection frequency - The above required inservice inspections of the SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 effective full power months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice 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;

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.7.2.12-1 at 40-month intervals 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 the subsequent inspections satisfy the criteria of Specification ~~5.7.2.12.e.1;~~ 5.7.2.12.f.1 the interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.7.2.12-1 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

b) A seismic occurrence greater than the Operating Basis Earthquake, or

c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

d) A main steam line or feedwater line break.

g.

~~f.~~ Acceptance Criteria

1. Terms as used in this specification will be defined as follows:

a) Degradation - A service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

b) Degraded Tube - A tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

c) % Degradation - The percentage of the tube wall thickness affected or removed by degradation;

d) Defect - An imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

e) Imperfection - An exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

f) Plugging Limit - The imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

ADD

This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.7.2.12.g.1.j for the repair limit applicable to these intersections.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

g) Preservice Inspection - An inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections.

h) Tube Inspection - An inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

i) Unserviceable - The condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operational Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break accident as specified in Specification 5.4.2.12.e

INSERT B

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2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.7.2.12-1.

g. Reports - The content and frequency of written reports shall be in accordance with Specification 5.9.9.

h.

(continued)

5.9 Reporting Requirements

5.9.9 SG Tube Inspection Report (continued)

The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected,
2. Location and percent of wall-thickness penetration for each indication of an imperfection, and
3. Identification of tubes plugged.

Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

ADD →

For implementation of the voltage based repair criteria to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.
2. If circumferential crack-like indications are detected at the tube support plate intersection.
3. If indication are identified that extend beyond the confines of the tube support plate.
4. If indications are identified at the tube support plate elevations that are attributable to primary water stress corrosion cracking.
5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2} , notify the NRC and provide an assessment of the safety significance of the occurrence.

INSERT A

Supplemental Inspection Requirements - Implementation of the steam generator tube to tube support plate repair criteria requires a 100-percent 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-percent random sampling of tubes inspected over their full length.

INSERT B

The Tube Support Plate Repair Limit - The Tube Support Plate Repair Limit is used for the disposition of Alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the repair limit is based on maintaining steam generator tube serviceability as described below:

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit of 1.0 volt will be allowed to remain in service.

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with the bobbin voltage greater than the lower voltage repair limit of 1.0 volt, will be repaired, except as noted in Specification 5.7.2.12.g.1.j.3 below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit of 1.0 volt but less than or equal to the upper voltage limit*, may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit* will be plugged or repaired.

4. Certain intersections as identified in Attachment 2 of WAT-D-10709 will be excluded from application of the voltage-based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.7.2.12.g.1.j.1, 5.7.2.12.g.1.j.2, and 5.7.2.12.g.1.j.3. The mid-cycle repair limits are determined from the following equations:

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

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

where:

V_{URL}	=	upper voltage repair limit
V_{LRL}	=	lower voltage repair limit
V_{MURL}	=	mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL}	=	mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt	=	length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL	=	cycle length (the time between two scheduled steam generator inspections)
V_{SL}	=	structural limit voltage
Gr	=	average growth rate per cycle length
NDE	=	95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e. a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach used in specifications 5.7.2.12.g.1.j.1, 5.7.2.12.g.1.j.2, and 5.7.2.12.g.1.j.3.

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

BASES

B 3.4 REACTOR COOLANT SYSTEM (RCS)

B 3.4.13 RCS Operational LEAKAGE

BASES

BACKGROUND

Components that contain or transport the coolant to or from the reactor core make up the RCS. Component joints are made by welding, bolting, rolling, or pressure loading, and valves isolate connecting systems from the RCS.

During plant life, the joint and valve interfaces can allow varying amounts of reactor coolant LEAKAGE, through either normal operational wear or mechanical deterioration. The purpose of the RCS Operational LEAKAGE LCO is to limit system operation in the presence of LEAKAGE from these sources to amounts that do not compromise safety. This LCO specifies the types and amounts of LEAKAGE.

10 CFR 50, Appendix A, GDC 30 (Ref. 1), requires means for detecting and, to the extent practical, identifying the source of reactor coolant LEAKAGE. Regulatory Guide 1.45 (Ref. 2) describes acceptable methods for selecting leakage detection systems.

The safety significance of RCS LEAKAGE varies widely depending on its source, rate, and duration. Therefore, detecting and monitoring reactor coolant LEAKAGE into the containment area is necessary. Quickly separating the identified LEAKAGE from the unidentified LEAKAGE is necessary to provide quantitative information to the operators, allowing them to take corrective action should a leak occur that is detrimental to the safety of the facility and the public.

A limited amount of leakage inside containment is expected from auxiliary systems that cannot be made 100% leaktight. Leakage from these systems should be detected, located, and isolated from the containment atmosphere, if possible, to not interfere with RCS leakage detection.

This LCO deals with protection of the reactor coolant pressure boundary (RCPB) from degradation and the core from inadequate cooling, in addition to preventing the accident analyses radiation release assumptions from being exceeded. The consequences of violating this LCO include the possibility of a loss of coolant accident (LOCA) or steam generator tube rupture (SGTR).

INSERT C

(continued)

BASES

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

DELETE AND INSERT D

Total primary to secondary LEAKAGE amounting to 1 gpm through all SGs produces acceptable offsite doses in the MSLB accident analysis. Violation of this LCO could exceed the offsite dose limits for this accident. Primary to secondary LEAKAGE must be included in the total allowable limit for identified LEAKAGE.

e. Primary to Secondary LEAKAGE through Any One SG

DELETE AND INSERT E

The 500 gallons per day limit on one SG is based on the assumption that a single crack leaking this amount would not propagate to a SGTR under the stress conditions of a LOCA or a main steam line rupture. If leaked through many cracks, the cracks are very small, and the above assumption is conservative.

(continued)

INSERT C

The voltage based repair limit of Specification 5.7.2.12.g.1.j implement the guidance of GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) 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 SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of Specification 5.7.2.12.g.1.j requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this Specification).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation at the 95-percent prediction interval curve reduces to account for the lower 95/95-percent tolerance bound for tubing material properties at 650 ^{0}F (i.e. the 95-percent lower tolerance limit (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 in Specification 5.7.2.12.g.1.j should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

Specification 5.9.9 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to notified prior to returning the SGs to service. 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 SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b(c) criteria.

INSERT D

The LEAKAGE limits incorporated into LCOs 3.4.13.d and 3.4.13.e are more restrictive than the standard operating LEAKAGE limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

The 600 gallons per day total primary to secondary LEAKAGE through all SGs ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10 CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.

INSERT E

The 150 gallons per day primary to secondary LEAKAGE through any one SG ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents

ENCLOSURE 3

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT (WBN) UNIT 1

PROPOSED TECHNICAL SPECIFICATION (TS) CHANGE WBN-TS-99-14
REVISED PAGES

I. AFFECTED PAGE LIST

Technical Specifications

3.4-30
5.0-15
5.0-16
5.0-17
5.0-18
5.0-19
5.0-35

Technical Specification Bases

B 3.4-74
B 3.4-76

II. REVISED PAGES

See Attached.

3.4 REACTOR COOLANT SYSTEM (RCS)

3.4.13 RCS Operational LEAKAGE

LCO 3.4.13 RCS operational LEAKAGE shall be limited to:

a. No pressure boundary LEAKAGE;

b. 1 gpm unidentified LEAKAGE;

c. 10 gpm identified LEAKAGE;

d. 600 gallons per day total primary to secondary LEAKAGE through all steam generators (SGs);

e. 150 gallons per day primary to secondary LEAKAGE through any one SG; and

f. Four SGs shall be OPERABLE.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. RCS LEAKAGE not within limits for reasons other than pressure boundary LEAKAGE.	A.1 Reduce LEAKAGE to within limits.	4 hours
B. Steam Generator Tube Surveillance Program not met.	B.1 Determine steam generator tube integrity is acceptable for continued operation	4 hours

(continued)

5.7 Procedures, Programs, and Manuals (continued)

5.7.2.12 Steam Generator (SG) Tube Surveillance Program

Each steam generator (SG) shall be demonstrated OPERABLE by performance of the following augmented inservice inspection program in addition to the provisions for inservice inspection of ASME Code Class 1, 2, and 3 components which shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a. See Specification 5.7.2.11 for applicable inspection Frequencies.

a. SG Sample Selection and Inspection - Each SG shall be determined OPERABLE during shutdown by selecting and inspecting at least the minimum number of SG tubes specified in Table 5.7.2.12-1.

b. SG Tube Sample Selection and Inspection - The SG tube minimum sample size, inspection result classification, and the corresponding action required shall be as specified in Table 5.7.2.12-1. The inservice inspection of SG tubes shall be performed at the frequencies specified in Specification 5.7.2.12.f and the inspected tubes shall be verified acceptable per the acceptance criteria of Specification 5.7.2.12.g. The tubes selected for each inservice inspection shall include at least 3% of the total number of tubes in all SGs; the tubes selected for these inspections shall be selected on a random basis except:

1. Where experience in similar plants with similar water chemistry indicates critical areas to be inspected, then at least 50% of the tubes inspected shall be from the critical areas;

2. The first sample of tubes selected for each inservice inspection (subsequent to the preservice inspection) of each SG shall include:

a) All nonplugged tubes that previously had detectable wall penetrations (greater than 20%),

b) Tubes in those areas where experience has indicated potential problems,

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

c) A tube inspection (pursuant to Specification 5.7.2.12.g) shall be performed on each selected tube. If any selected tube does not permit the passage of the eddy current probe for a tube inspection, this shall be recorded and an adjacent tube shall be selected and subjected to a tube inspection, and

d) Indications left in service as a result of the application of the tube support plate voltage repair criteria shall be inspected by bobbin probe during all future refueling outages.

c. Examination Results - The results of each sample inspection shall be classified into one of the following three categories:

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 (greater than 10%) further wall penetrations to be included in the above percentage calculations.

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d. Supplemental Sampling Requirements - The tubes selected as the second and third samples (if required by Table 5.7.2.12-1) may be subjected to a partial tube inspection provided:

1. The tubes selected for these samples include the tubes from those areas of the tube sheet array where tubes with imperfections were previously found, and

2. The inspections include those portions of the tubes where imperfections were previously found.

e. Supplemental Inspection Requirements - Implementation of the steam generator tube to tube support plate repair criteria requires a 100-percent 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

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

tube support plate intersections having ODSCC indications shall be based on the performance of at least a 20-percent random sampling of tubes inspected over their full length.

f. Inspection frequency - The above required inservice inspections of the SG tubes shall be performed at the following frequencies:

1. The first inservice inspection shall be performed after 6 effective full power months but within 24 calendar months of initial criticality. Subsequent inservice inspections shall be performed at intervals of not less than 12 nor more than 24 calendar months after the previous inspection. If two consecutive inspections, not including the preservice 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;

2. If the results of the inservice inspection of a SG conducted in accordance with Table 5.7.2.12-1 at 40-month intervals 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 the subsequent inspections satisfy the criteria of Specification 5.7.2.12.f.1; the interval may then be extended to a maximum of once per 40 months; and

3. Additional, unscheduled inservice inspections shall be performed on each SG in accordance with the first sample inspection specified in Table 5.7.2.12-1 during the shutdown subsequent to any of the following conditions:

   a) Primary-to-secondary tube leaks (not including leaks originating from tube-to-tube sheet welds) in excess of the limits of Specification 3.4.13, or

   b) A seismic occurrence greater than the Operating Basis Earthquake, or

   c) A loss-of-coolant accident requiring actuation of the Engineered Safety Features, or

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

d) A main steam line or feedwater line break.

g. Acceptance Criteria

1. Terms as used in this specification will be defined as follows:

a) Degradation - A service-induced cracking, wastage, wear, or general corrosion occurring on either inside or outside of a tube;

b) Degraded Tube - A tube containing imperfections greater than or equal to 20% of the nominal wall thickness caused by degradation;

c) % Degradation - The percentage of the tube wall thickness affected or removed by degradation;

d) Defect - An imperfection of such severity that it exceeds the plugging limit. A tube containing a defect is defective;

e) Imperfection - An exception to the dimensions, finish, or contour of a tube from that required by fabrication drawings or specifications. Eddy-current testing indications below 20% of the nominal tube wall thickness, if detectable, may be considered as imperfections;

f) Plugging Limit - The imperfection depth at or beyond which the tube shall be removed from service and is equal to 40% of the nominal tube wall thickness;

This definition does not apply to tube support plate intersections for which the voltage-based repair criteria are being applied. Refer to Specification 5.7.2.12.g.1.j for the repair limit applicable to these intersections.

g) Preservice Inspection - An inspection of the full length of each tube in each SG performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed prior to initial MODE 1 operation using the equipment and techniques expected to be used during subsequent inservice inspections.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

h) Tube Inspection - An inspection of the SG tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg; and

i) Unserviceable - The condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operational Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break accident as specified in Specification 5.7.2.12.f.

j. The Tube Support Plate Repair Limit - The Tube Support Plate Repair Limit is used for the disposition of Alloy 600 steam generator tubes for continued service that are experiencing predominantly axially oriented outside diameter stress corrosion cracking confined within the thickness of the tube support plates and flow distribution baffle (FDB). At tube support plate intersections (and FDB), the repair limit is based on maintaining steam generator tube serviceability as described below:

1. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with bobbin voltages less than or equal to the lower voltage repair limit of 1.0 volt will be allowed to remain in service.

2. Steam generator tubes, whose degradation is attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with the bobbin voltage greater than the lower voltage repair limit of 1.0 volt, will be repaired, except as noted in Specification 5.7.2.12.g.1.j.3 below.

3. Steam generator tubes with indications of potential degradation attributed to outside diameter stress corrosion cracking within the bounds of the tube support plate with a bobbin voltage greater than the lower voltage repair limit of 1.0 volt but less than or equal to the upper voltage limit,* may remain inservice if a rotating pancake coil inspection does not detect degradation. Steam generator tubes, with indications of outside diameter stress corrosion cracking degradation with a bobbin voltage greater than the upper voltage repair limit* will be plugged or repaired.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

4. Certain intersections as identified in Attachment 2 of WAT-D-10709 will be excluded from application of the voltage-based repair criteria as it is determined that these intersection may collapse or deform following a postulated LOCA + SSE event.

5. If an unscheduled mid-cycle inspection is performed, the following mid-cycle repair limits apply instead of the limits identified in 5.7.2.12.g.1.j.1, 5.7.2.12.g.1.j.2, and 5.7.2.12.g.1.j.3. The mid-cycle repair limits are determined from the following equations:

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

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

where:

V_{URL} = upper voltage repair limit
V_{LRL} = lower voltage repair limit
V_{MURL} = mid-cycle upper voltage repair limit based on time into cycle
V_{MLRL} = mid-cycle lower voltage repair limit based on V_{MURL} and time into cycle
Δt = length of time since last scheduled inspection during which V_{URL} and V_{LRL} were implemented
CL = cycle length (the time between two scheduled steam generator inspections)
V_{SL} = structural limit voltage
Gr = average growth rate per cycle length
NDE = 95-percent cumulative probability allowance for nondestructive examination uncertainty (i.e. a value of 20-percent has been approved by the NRC)

Implementation of these mid-cycle repair limits should follow the same approach used in Specifications 5.7.2.12.g.1.j.1, 5.7.2.12.g.1.j.2, and 5.7.2.12.g.1.j.3.

(continued)

5.7 Procedures, Programs, and Manuals

5.7.2.12 Steam Generator (SG) Tube Surveillance Program (continued)

* The upper voltage repair limit is calculated according to the methodology in GL 95-05 as supplemented. V_{URL} will differ at the tube support plates and flow distribution baffle.

2. The SG shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 5.7.2.12-1.

h. <u>Reports</u> - The content and frequency of written reports shall be in accordance with Specification 5.9.9.

(continued)

5.9 Reporting Requirements

5.9.9 SG Tube Inspection Report (continued)

The complete results of the SG tube inservice inspection shall be submitted to the NRC within 12 months following the completion of the inspection. The report shall include:

1. Number and extent of tubes inspected,

2. Location and percent of wall-thickness penetration for each indication of an imperfection, and

3. Identification of tubes plugged.

Results of SG tube inspections that fall into Category C-3 shall be reported to the NRC in accordance with 10 CFR 50.72. This report shall provide a description of investigations conducted to determine cause of the tube degradation and corrective measures taken to prevent recurrence.

For implementation of the voltage based repair criteria to tube support plate (and flow distribution baffle) intersections, notify the NRC prior to returning the steam generators to service should any of the following conditions arise:

1. If estimated leakage based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds the leakage limit (determined from the licensing basis dose calculation for the postulated main steam line break) for the next operating cycle.

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

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

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

5. If the calculated conditional burst probability based on the projected end-of-cycle (or if not practical, using the actual measured end-of-cycle) voltage distribution exceeds 1×10^{-2}, notify the NRC and provide an assessment of the safety significance of the occurrence.

BASES (continued)

BACKGROUND (continued)

The voltage based repair limit of Specification 5.7.2.12.g.1.j implement the guidance of GL 95-05 and are applicable only to Westinghouse-designed steam generators (SGs) 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 SG tube degradation nor are they applicable to ODSCC that occurs at other locations within the SG. Additionally, the repair criteria apply only to indications where the degradation mechanism is dominantly axial ODSCC with no significant cracks extending outside the thickness of the support plate. Refer to GL 95-05 for additional description of the degradation morphology.

Implementation of Specification 5.7.2.12.g.1.j requires a derivation of the voltage structural limit from the burst versus voltage empirical correlation and then the subsequent derivation of the voltage repair limit from the structural limit (which is then implemented by this Specification).

The voltage structural limit is the voltage from the burst pressure/bobbin voltage correlation at the 95-percent prediction interval curve reduces to account for the lower 95/95-percent tolerance bound for tubing material properties at 650°F (i.e. the 95-percent lower tolerance limit (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 in Specification 5.7.2.12.g.1.j should only be used during unplanned inspections in which eddy current data is acquired for indications at the tube support plates.

(continued)

BASES

BACKGROUND (continued)	Specification 5.9.9 implements several reporting requirements recommended by GL 95-05 for situations which the NRC wants to notified prior to returning the SGs to service. 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 SGs to service. Note that if leakage and conditional burst probability were calculated using the measured EOC voltage distribution for the purposes of addressing the GL section 6.a.1 and 6.a.3 reporting criteria, then the results of the projected EOC voltage distribution should be provided per the GL section 6.b(c) criteria.

(continued)

LCO (continued)

b. Unidentified LEAKAGE

One gallon per minute (gpm) of unidentified LEAKAGE is allowed as a reasonable minimum detectable amount that the containment air monitoring and containment pocket sump level monitoring equipment can detect within a reasonable time period. Violation of this LCO could result in continued degradation of the RCPB, if the LEAKAGE is from the pressure boundary.

c. Identified LEAKAGE

Up to 10 gpm of identified LEAKAGE is considered allowable because LEAKAGE is from known sources that do not interfere with detection of unidentified LEAKAGE and is well within the capability of the RCS Makeup System. Identified LEAKAGE includes LEAKAGE to the containment from specifically known and located sources, but does not include pressure boundary LEAKAGE or controlled reactor coolant pump (RCP) seal leakoff (a normal function not considered LEAKAGE). Violation of this LCO could result in continued degradation of a component or system.

d. Primary to Secondary LEAKAGE through All Steam Generators (SGs)

The LEAKAGE limits incorporated into LCOs 3.4.13.d and 3.4.13.e are more restrictive than the standard operating LEAKAGE limits and are intended to provide an additional margin to accommodate a crack which might grow at a greater than expected rate or unexpectedly extend outside the thickness of the tube support plate. Hence, the reduced leakage limit, when combined with an effective leak rate monitoring program, provides additional assurance that should a significant leak be experienced in service, it will be detected, and the plant shut down in a timely manner.

The 600 gallons per day total primary to secondary LEAKAGE through all SGs ensures that the dosage contribution from the tube leakage will be limited to a small fraction of 10-CFR Part 100 limits in the event of either a steam generator tube rupture or steam line break. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents.

(continued)

BASES

LCO (continued)

e. Primary to Secondary LEAKAGE through Any One SG

The 150 gallons per day primary to secondary LEAKAGE through any one SG ensures that steam generator tube integrity is maintained in the event of a main steam line rupture or under LOCA conditions. The limit is consistent with, or conservative to, the assumptions used in the analysis of these accidents

f. Steam Generator OPERABILITY

Four SGs are also required to be OPERABLE. This requirement is met by satisfying the augmented inservice inspection requirements of the Steam Generator Tube Surveillance Program (Specification 5.7.2.12).

APPLICABILITY

In MODES 1, 2, 3, and 4, the potential for RCPB LEAKAGE is greatest when the RCS is pressurized.

In MODES 5 and 6, LEAKAGE limits are not required because the reactor coolant pressure is far lower, resulting in lower stresses and reduced potentials for LEAKAGE.

ACTIONS

A.1

Unidentified LEAKAGE, identified LEAKAGE, or primary to secondary LEAKAGE in excess of the LCO limits must be reduced to within limits within 4 hours. This Completion Time allows time to verify leakage rates and either identify unidentified LEAKAGE or reduce LEAKAGE to within limits before the reactor must be shut down. This action is necessary to prevent further deterioration of the RCPB.

(continued)

ENCLOSURE 4

WATTS BAR NUCLEAR PLANT UNIT 1
OUTSIDE DIAMETER STRESS CORRISION CRACKING REPAIR CRITERIA

COMMITMENT LIST

ENCLOSURE 4

WATTS BAR NUCLEAR PLANT UNIT 1
OUTSIDE DIAMETER STRESS CORRISION CRACKING REPAIR CRITERIA

COMMITMENT LIST

TVA will revise the Updated Final Safety Analysis Report to include a reference to this letter for implementing the outside diameter stress corrosion cracking (ODSCC) repair criteria.