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SEP 1 8 2000

TVA-WBN-TS-99-014

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) - CLARIFICATION (TAC NO. MA8635)

The purpose of this submittal is to provide clarification to TVA's letter dated April 10, 2000, concerning the subject technical specification change and to respond to NRC's request for additional information dated July 24, 2000. The enclosed clarifications and responses were discussed in teleconference calls on June 12 and July 19, 2000, with the NRC Project Manager, R. Martin and NRC Reviewers J. Tsao and J. Hayes, respectively. The results of those discussions are documented in Enclosure 1 and Enclosure 3 of this letter.

Enclosure 2 provides markups of technical specification pages that required additional clarification. TVA has evaluated the changes in the proposed amendment and finds that these changes do not effect the initial proposed No Significant Hazards Consideration determination. Revised pages for these markups will be coordinated with the NRC Project Manager prior to issuance of the technical specification amendment.

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Enclosure 3 provides the responses to NRC's request for additional information dated July 24, 2000 concerning the main steam line break calculations. Attachment 1 to this enclosure provides a summary sheet of the assumptions for the revised main steam line break dose calculation provided in Attachment 2 of this enclosure. This summary sheet corrects an error which was identified by TVA on page E1A1-2 of the April 10, 2000, letter to NRC. TVA informally notified the Staff of this error on May 19, 2000. After discussions with the NRC concerning the methodology used in the main steam line break calculation, TVA has revised the calculation to conform to the Standard Review Plan and industry standards. This summary sheet also provides the assumptions for the main control room dose calculation. These assumptions were requested by the reviewer in a June 15, 2000, teleconference.

Enclosure 4 identifies the commitments in this letter. If you should have any questions concerning this matter, please contact P. L. Pace at (423) 365-1824.

Sincerely,

W. R. Lagergren

W. R. Lagergren

cc See Page 3
Enclosures

Subscribed and sworn to before me
on this 18th day of September, 2000

E. Jeannette Long

Notary Public

My Commission Expires June 27, 2001

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cc: (Enclosures):

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ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
CLARIFICATION OF PROPOSED CHANGE

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
CLARIFICATION OF PROPOSED CHANGE

The following concerns were discussed in a teleconference calls with the NRC WBN Project Manager and NRC reviewer on June 12 and July 19, 2000. This letter documents the TVA/NRC discussion of these issues and provides clarification and/or additional information as required.

QUESTION 1

Section 3.b.3 of Attachment 1 to GL 95-05, in Enclosure 1, Table 1, the licensee stated that "+Point (or equivalent) will be used for inspection of all dents >5 volts. Any indications found at such intersections with RPC [rotating pancake coil] 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." TVA has committed to perform RPC inspections of all intersections with dents greater than 2 volts. Based on experience with ODSCC at plants with dented support plate intersections, circumferential cracking has been observed in dents under 5 volts and at Sequoyah Unit 1, circumferential cracking has been observed in dents under 2 volts. For the staff to evaluate the Watts Bar dent inspection program in connection with this amendment, TVA will have to propose a program that will lead to inspection of dents for circumferential cracks in dents below 2 volts if their inspections indicate that circumferential cracking is occurring in dents above 2 volts. The proposed sampling and expansion program needs to be consistent with the inspection philosophy of the EPRI SG inspection guidelines.

RESPONSE 1

The hot leg dented tube support intersections with bobbin dent voltage greater than or equal to 2 volts will be inspected with a technique qualified to Appendix H of the EPRI Pressurized Water Reactor (PWR) Steam Generator Examination Guideline. The denting at Watts Bar Nuclear Plant (WBN) Unit 1 is not comparable to TVA's Sequoyah Nuclear Plant (SQN) Unit 1. The denting at WBN is mechanically induced, where the denting at SQN Unit 1 is corrosion induced. The denting at tube support plate intersections at WBN is more correctly characterized as "dings." The majority are located near the edge of the support plate or just outside the support plate, whereas at SQN Unit 1, the majority of the denting is located in the center of the support plate. The stresses are dramatically different, and the consequences of circumferential cracking at these dents are also

different. No cracking has been identified in dented support plate intersections at WBN.

The inspection plan for WBN's current Unit 1 Cycle 3 outage includes 100% of hot leg dented intersections greater than or equal to 2 volts. This inspection is driven by industry experience, and only includes 90 intersections. If circumferential cracking is identified at WBN in a dented tube support plate intersection that is equal to 2 volts, the inspection plan expands to hot leg dented intersections greater than or equal to 1.5 volts.

QUESTION 2

In recent information provided by Entergy in connection with an operational assessment for ANO-2, data was presented that calls into question the validity of the burst correlation developed for the voltage-based repair criteria. This data demonstrates that rapid burst pressure testing for tubes with ODSCC flaws exhibits a strong testing rate effect, i.e., an apparent strength much higher than would be found from quasi-static testing. The burst data developed for the voltage-based repair criteria contains tubes tested under very rapid pressurization. Therefore, the burst correlation relied upon for the repair criteria may also exhibit this effect. Structural integrity is governed by a requirement to demonstrate a margin of 3 to the normal operating differential pressure. Normal operating pressure is a steady state condition. Therefore, the margin is based on the same condition, rather than a rapidly changing pressure condition. TVA needs to address the preceding explanation and demonstrate how the governing margins are satisfied by their proposed repair criteria.

RESPONSE 2

This subject was discussed in a meeting between the NEI Steam Generator Task Force and the NRC technical staff in early July. At that meeting, a project was presented to the staff that has begun and a draft of the report is currently being reviewed by the industry. The section of the report that discusses the ODSCC burst correlation was given high priority in the industry review and was sent to the NRC staff by NEI the latter part of August, 2000. That section of the report was discussed on a teleconference among the NRC staff, NEI, and industry representatives on September 5. The full report is scheduled to

be discussed with the NRC staff in a meeting at the end of September, 2000. This project consists of the following tasks:

1) Verify and document all aspects of the ANO specimen burst tests to ensure that the results are accurate and evidence of a real physical phenomenon.

2) Compile a database of loading rate and hold time effects for completed tests.

3) Evaluate the significance of ANO specimen burst test data including a characterization of the phenomenon and the implications of the effect on tube burst models and alternate repair criteria (ARCs).

NRC's concerns with the ODSCC burst correlation should be eliminated by the report and the subsequent teleconference. TVA is providing excerpts of the draft EPRI Report, "Steam Generator Tubing Burst Testing Review," in the Attachment of this Enclosure as further clarification to this question. This portion of the draft report has been reviewed by the Steam Generator Task Force Group.

QUESTION 3

The licensee stated that its ARC is consistent with Generic Letter (GL) 95-05 and provided exceptions to certain sections in Attachment 1 to GL 95-05 (e.g., Enclosure 1 Table 1 in the licensee's submittal). The licensee did not address all sections in Attachment 1 to GL 95-05; therefore, the staff is not clear of the licensee's intent regarding those GL 95-05 sections that were not addressed, specifically, Sections 2.b.2; 2.b.2(1); 2.b.2(2); 2.b.3; 2.b.3(1); 2.b.3(2); and 2.c. Please clarify.

RESPONSE 3

TVA's program follows the guidance from GL 95-05 Sections 2.b.2; 2.b.2.(1); 2.b.2(2); 2.b.3; 2.b.3(2); and 2.c. with only one exception. See the revised attached Table 1. This exception is discussed in the text on page 10 of Enclosure 1 of the April 10, 2000 letter. For the initial cycle, since no prior tube support plate (TSP) ODSCC has been detected, a bounding probability distribution function of growth rates is used. More information on the bounding distribution is in the response to Question 6 below.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
CLARIFICATION OF PROPOSED CHANGE

QUESTION 4

Section 2.a.1 of Attachment 1 to GL 95-05. The licensee stated that Addendum 3 to EPRI NP-7480-L, which contains burst and leakage data for axial ODSCC, is slightly more conservative than Addendum 2. The licensee proposed to use Addendum 3 in its condition monitoring and operational assessment if the NRC approves Addendum 3 before the next refueling outage. The Staff and NEI have agreed upon a protocol in which the ODSCC database can be updated periodically without staff approval. The protocol provides guidance and acceptance criteria in which data may be added or not included in the database without staff approval. Therefore, pending exceptions taken to the protocol, the staff does not intend to update its review of the database.

RESPONSE 4

TVA understands that NRC approval of Addendum 3 to EPRI NP-7480-L is not required because it was revised based on industry protocol. TVA plans to utilize Addendum 3 for condition monitoring and operational assessments.

QUESTION 5

Section 2.b.1 of Attachment 1 to GL 95-05. The Staff has not completed its review of the probability of prior cycle detection (POPCD) methodology that the industry has proposed. The Staff does not foresee the completion of this review before the upcoming Watts Bar refueling outage in September 2000. Therefore, the licensee needs to use the probability of detection (POD) of 0.6 as specified in GL 95-05 in its condition monitoring and operational assessments.

RESPONSE 5

TVA understands that the review of POPCD was not completed prior to the current WBN Unit 1 Cycle 3 outage. As stated in TVA's April 10, 2000 License Amendment request, TVA's program calls for a POD of 0.6 for the first cycle of implementation. If the Staff approves POPCD or another alternative following the first cycle of implementation, it may be considered for use by TVA in future plant outages.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
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QUESTION 6

Section 2.b.2(2) of Attachment 1 to GL 95-06. The licensee stated that it will use a bounding curve for the voltage growth for flaws from similarly designed and operated plants. (1) Clarify if the bounding growth curve will be constructed from the maximum growth curves of all previous operating cycles from all similar plants or from a specific plant. (2)Clarify if two bounding growth curves will be used in assessments, i.e., one for TSP indications and one for FDBP (flow distribution baffle plate) indications, or if one bounding curve will be used for both TSP and FDBP indications.

RESPONSE 6

The bounding growth distribution is developed considering maximum growth rates in similarly designed and operated plants while a 1-volt repair criteria was in effect. This growth distribution was developed recently and utilized to perform an operational assessment per GL 95-05 for another plant with similar steam generators. As there is insufficient data to develop a separate growth distribution at the FBBP elevation, only a growth distribution applicable to TSP indications is utilized. This is applied to FDBP elevations, but the data is evaluated to verify that such application is appropriate.

QUESTION 7

Section 3.b of Attachment 1 to GL 95-05 - The licensee stated that it will use the +point or future equivalent probe for confirmation of bobbin indications. The licensee also stated that "rotating pancake coil for the purposes of the technical specification change also includes the use of comparable or improved nondestructive examination techniques."

(1) Clarify how comparable or improved nondestructive examination techniques will be qualified before their use in the ARC application.

(2) In its Enclosure 1 (page E1-13), the licensee stated that the +Point probe and Cecco probes are considered by the industry as acceptable alternatives to rotating pancake coil. Based on recent inspection results, the staff has concerns about issues such as the adequacy of the Cecco probe to perform training and declining call rates during production. Should

the licensee wish to obtain NRC approval to modify the methodology to use Cecco probes in the confirmation inspection, the licensee needs to submit information to address these concerns.

RESPONSE 7

(1) TVA's discussion of comparable or improved nondestructive examination techniques is meant for any future inspection techniques qualified through the EPRI Appendix H qualification program.

(2) TVA plans to use the +point probe for confirmation of bobbin indications. TVA will not utilize the Cecco probe.

QUESTION 8

Section 3.c.5 of Attachment 1 to GL 95-05. In Enclosure 1, Table 1, the licensee stated that "...it is expected that these [quantitative noise] criteria will be evolving over the inspection and as a result, are subject to change." Clarify why the quantitative noise criteria will evolve and how the acceptable noise criteria is established.

RESPONSE 8

The noise criteria is established as a basis for rejection of the data (i.e. retest) or further evaluation. TVA has voltage criteria for mix residuals (2 volts) and permeability variations (5 volts). These are initial settings. As data is evaluated, analysts identify areas of the tube (such as support plates and top of tubesheet) where interference from "noise" such as dents or permeability inhibit the analysis process. In these cases a rotating coil examination is performed. If the data is still corrupted by this noise the tube is plugged. TVA practices concerning the noise criteria follows standard industry practices.

QUESTION 9

Section 3.c.6 of Attachment 1 to GL 95-05. In Enclosure 1, Table 1, the licensee stated that its data analyst will review the mixed residuals on the standard itself and take action as required in the TVA analysis guidelines. Discuss briefly this

aspect of the TVA analysis guidelines and the basis for the action criteria in guidelines. Describe how this aspect of the guidelines is covered by the EPRI examination guidelines.

RESPONSE 9

It is the responsibility of each analyst to perform the mixes in accordance with the appropriate examination techniques specification sheets (ETSS) and ensure that the process channels are created such that the appropriate structure is minimized on the calibration standard. This may necessitate the need to move the artifact on the calibration standard being suppressed, move the conduit to provide a straighter approach or exit from the standard, or invert the orientation of the standard. Guidance on dealing with mixed residuals is not included in the EPRI examination guidelines. However, TVA practices dealing with mixed residuals meet standard industry practices.

QUESTION 10

Section 4 of Attachment I to GL 95-05. With regard to tube removal schedule, the licensee stated that it will follow the industry proposed guidance in NP-7480-L, Addendum 3, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1999 Database Update," EPRI, May 1999. The proposed tube removal schedule in NP-7480-L, Addendum 3, deviates from GL 95-05. According to the industry proposal, tube removal can be delayed for one operating cycle if no indications greater than or equal to 3 volts are found during tube inspection. The tube selection criteria and tube examination and testing program in NP-7480-L, Addendum 3, also deviates from GL 95-05. In a letter to David J. Modeen of Nuclear Energy Institute dated January 31, 2000, the Staff found that some parts of industry proposed tube removal program acceptable and some parts are not acceptable. Pending resolution of the issue, the licensee needs to follow the tube removal program in Section 4.0 of Attachment 1 to GL 95-05.

RESPONSE 10

A letter from David J. Modeen dated June 2, 2000, entitled "Steam Generator Degradation Specific Management Database, Addendum 3," replied to NRC's January 31, 2000 letter to Nuclear Energy Institute (NEI). TVA will only implement those parts of the

industry proposed alternative tube pull program that the NRC Staff has approved as indicated by Jack R. Strosnider's letter to David J. Modeen, dated January 31, 2000.

QUESTION 11

The FDBP was included in some, but not all, relevant TS sections. This inconsistency gives an impression that the ARC is applicable to flow distribution plate intersections in certain conditions but not in other conditions. To avoid potential confusion in the interpretation of the TS requirements and applicability, the licensee should either include flow distribution baffle plate intersection in all relevant TS sections and/or define that the ARC is applicable to the flow distribution baffle plate and tube support plate intersections in a generic statement in the beginning of TS 5.7.2.12.

RESPONSE 11

The ARC is applicable to all relevant TSP and FDBP intersections. The TS has been revised to include the FDBP. Enclosure 2 provides the additional markups.

QUESTION 12

The licensee referenced a report, WAT-D-10709, in TS section 5.7.2.12.g.l.j.4. This report contains those tube support plate intersections that will be excluded from ARC application. The licensee needs to provide a complete reference to this report (e.g., title, author, revision number, date of issuance).

RESPONSE 12

The complete reference is: WAT-D-10709, Letter from John W. Irons to J. E. Maddox and M. J. Lorek, "Tennessee Valley Authority Watts Bar Nuclear Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria for Westinghouse Steam Generator Tubes Affected By ODSCC at TSPs," dated January 12, 2000. The reference has been added to the TS. See Enclosure 2 marked up pages.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
CLARIFICATION OF PROPOSED CHANGE

Table 1

Acknowledgment of Individual GL 95-05 Performance Criteria

GL 95-05 Item	GL Methodology Concurrence	Comments
1.a	Followed	TVA will abide by the specified guidance in GL 95-05
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.2	Followed	TVA will abide by the specified guidance in GL 95-05
2.b.2(1)	Followed	TVA will abide by the specified guidance in GL 95-05
2.b.2(2)	Modified initial cycle. Followed there after.	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. For subsequent inspections, TVA will continue to abide by the specified guidance in GL 95-05.
2.b.3	Followed	TVA will abide by the specified guidance in GL 95-05
2.b.3(1)	Followed	TVA will abide by the specified guidance in GL 95-05
2.b.3(2)	Followed	TVA will abide by the specified guidance in GL 95-05
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.
2.c	Followed	TVA will abide by the specified guidance in GL 95-05
3.a	Followed	TVA will abide by the specified guidance in GL 95-05
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.

ENCLOSURE 1

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
CLARIFICATION OF PROPOSED CHANGE

Table 1

Acknowledgment of Individual GL 95-05 Performance Criteria

GL 95-05 Item	GL Methodology Concurrence	Comments
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.
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.

ENCLOSURE 1
ATTACHMENT

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
DRAFT EPRI REPORT EXCERPTS
STEAM GENERATOR TUBING BURST TESTING REVIEW

Review of the ANO Incident

During the inspection of the steam generator (SG) tubes at the Arkansas Nuclear One Unit 2 (ANO 2) nuclear power plant in November 1999, the tube located at row 72 and column 72 (R72C72) in SG B was pressure tested in situ to evaluate the burst resistance and leakage potential of an indication found by eddy current inspection (ECT). The maximum pressure attained during the test was 4147 psi when corrected to account for the pressure drop due to pressurization fluid flow and for instrument error. At that pressure the 100% throughwall dimension of the flaw in the tube was of such a size that the test equipment could not supply enough water to maintain the pressure. The pressure test was performed as a whole tube test, as contrasted to use of a localized test employing a sealing bladder. In such cases, the maximum pressure that can be attained during the test is limited by the flow capacity of the testing equipment.

One of the objectives of performing a pressurization test is to determine if the burst resistance of the tube with the indication meets a performance criterion value of three times the normal operation primary-to-secondary pressure difference (3DP). The criterion value for the ANO 2 SG tubes is greater than the pressure achieved during the test. The results from the pressure test were considered to be inconclusive because the information available from the test is not sufficient to demonstrate that the burst pressure of the tube had been reached even though a pressure of 3DP had not been reached. To further evaluate whether or not the R72C72 tube burst pressure was greater than or equal to 3DP, a series of surrogate specimens were fabricated and pressure tested. The flaws in the test specimens were made using EDM techniques to simulate the single, eddy current (ECT) estimated crack profile of the R72C72 indication. Separate series of pressure tests were conducted to identify both the leak (ligament tearing) and burst resisting capability of the flawed tube.

Information was presented by ANO 2 personnel to the NRC staff on June 8, 2000, which compared burst pressures obtained following ligament tearing with standard burst pressure test results obtained using a foil reinforced bladder with pressurization rates on the order of 2000 psi/s (in accord with industry guidelines). The results from the ligament tearing and burst pressure testing were as follows:

1) Ligament tearing pressure tests of the EDM specimens demonstrated ligament tearing pressures significantly below the value of 3DP for the ANO 2 plant SGs. Ligament tearing tests were performed using very slow pressurization rates (on the order of several psi per minute) in order to achieve tearing of a portion of the crack without opening the whole crack.

2) Burst pressure tests performed on specimens that had been previously pressurized to ligament tearing did not result in demonstrating a burst pressure in excess of the ligament tearing pressure. The burst pressures following ligament tearing were found to be less than the burst pressures obtained using the industry standard burst pressure test techniques. These

tests were performed per EPRI guidelines [NSD-EPRI-0646, "EPRI Guidelines for Leak & Burst Testing of SG Tubes," Westinghouse Electric Company, Madison, PA (July, 11, 1994).] using a foil reinforced plastic liner to prevent loss of pressure through the simulated flaw opening.

3) Burst pressure tests performed on non-previously tested specimens exhibited burst pressures significantly in excess of the ligament tearing pressures. The burst pressure tests were performed using a pressurization rate of about 2000 psi/s. The specimens were also lined with a flexible bladder which was reinforced in the immediate vicinity of the flaw with a lubricated brass foil patch prior to testing.

The ANO 2, R72C72 indication was about 1.4 inches long with two deep sections separated by a shallower ligament, such that when that ligament tore, the resulting through wall length did not permit a higher burst pressure than the ligament tearing pressure.

During a SGTF telephone conference on June 20, 2000, the NRC staff expressed the following concerns:

1) The results indicate that if the pressure was raised slowly, and if hold times were included in the pressurization process, the tubes would have failed at pressures significantly lower than 3DP.

2) The rapid increase in pressure used for the burst tests did not allow time for ligament failures to occur, thus artificially elevating the measured burst pressure of the surrogate specimens.

3) This issue may have safety concerns as it raises questions about the adequacy of analytical burst models and of the results of similar laboratory tests that were used as a basis for the ODSCC voltage based alternate repair criteria. If the laboratory tests are flawed, i.e., if a pressure ramp rate dependency has been overlooked, the validity of the data used to support the ARCs could be questioned.

In early July of 2000, industry representatives proceeded with a systematic approach aimed at developing an understanding of the ANO test program results and their implications relative to the results from other test programs. It is the industry's conclusion that this phenomenon is expected to effect only long and near-throughwall cracks (likely significantly longer than an acceptable length for a throughwall flaw such as about 0.7 inch). This conclusion and supporting documentation will be presented to the NRC on September 28 by NEI and the Steam Generator Task Force. The text below is submitted to address the ODSCC burst correlation only.

Industry Burst Pressure Testing

The ANO 2 test results do represent an unexplained difference relative to the results of SG tube burst testing that has been performed during the last quarter century. Specific tests have been performed to verify that the testing processes do not result in an artificial increase in the burst pressure from strain-rate effects. However, the issue here is whether or not there is some other time-dependent phenomenon active when a complex crack profile is involved. The testing database mostly consists of through wall slits and rectangular shaped profiles for developing analytical models for predicting burst pressure. However, burst data at different pressurization rates for pulled tubes, laboratory generated flaws and more complex EDM profiles exists and is interspersed in the test database with no apparent existence of the anomaly observed in the ANO testing.

Actual performance of burst tests varies from laboratory to laboratory. A joint set of guidelines was developed for EPRI in 1994 by representatives of Westinghouse, Laborelec and Packer Engineering [NSD-EPRI-0646, "EPRI Guidelines for Leak & Burst Testing of SG Tubes," Westinghouse Electric Company, Madison, PA (July, 11, 1994).]. Comments on the guidelines were received from Electricité de France (EdF) in 1995 [Berge, P., "EPRI Guidelines for Leak and Burst Testing of Steam Generator Tubes," Electricité de France, Paris, France (January 8, 1995).]. A draft revision of the guidelines was prepared in early 1996 to address special considerations associated with circumferential cracking, but has not been formally issued. One of the provisions of the guidelines is the recommendation that the pressurization rate be in the range of 200 to 2000 psi/sec. EdF noted in their comments that they restrict the pressurization rate to 116 psi/sec.

Burst testing is performed on pulled tube specimens, specimens in which cracking has been environmentally induced and on specimens for which the degradation has been fabricated by some machining technique. The use of slits as surrogate cracks is a long-standing practice and is based on the fact that SG tube materials are ductile and failure is usually typified as being by tensile overload rather than fracture mechanics processes. Moreover, crack tips in SG tube material blunt to widths greater than those of the machined slits before crack extension takes place [Hernalsteen, P., "The influence of Testing Conditions on Burst-Pressure Assessment for Inconel Tubing," IJPVP, **52**, pp. 41-57 (1992).], hence there is essentially no difference between environmentally induced and mechanically fabricated flaws during the failure process. Because EDM results in a thin, brittle surface layer at the plane of machining, it is possible that microscopic extension of such flaws occurs at loads significantly less than those that lead to gross failure of the flawed specimen. There has, however, been no systematic study aimed at investigating this possibility. One significant difference between environmentally and mechanically produced flaws is the planar nature of the mechanical flaws. Environmentally induced cracks tend to consist of arrays of smaller cracks that are nearly, but are not coplanar. Hence, there are also ligaments or material bridges between the smaller cracks that tend to strengthen the cracked tube relative to its mechanically fabricated counterpart. This is another area

where systematic research aimed at characterizing those strengthening effects has not been performed. The omission of such effects leads to some inherent conservatism in the use of planar slits to simulate non-planar cracks.

Westinghouse-Churchill Burst Testing Procedure

Burst tests are performed in accordance with the EPRI guidelines for laboratory measurements of burst strength of degraded steam generator tubing. Burst specimens are initially prepared by installing an unreinforced plastic bladder into the ID of the test specimen to prevent leakage before burst. The bladder is a clear plastic laboratory tubing with a wall thickness of approximately 0.125 inches. The outer diameter of the bladder is selected to be slightly larger than the inner diameter of the test specimen. The bladder is stretched axially, reducing its diameter, thus permitting the test specimen to be slipped over the bladder. As the stretching force is released, the bladder expands radially and seals the test specimen, the ends of the bladder are then trimmed square using a knife. The specimens are then prepared by installing a Swagelok plug on one end of the tube specimen and a modified Swagelok adapter to the other. This permits the specimen to be connected to the burst test apparatus via a high pressure connector. The test specimen is pre-filled with de-ionized water and the pressure intensification cycle is initiated with the pressure rising at a rate of 2000 psi per second, until tube burst occurs.

In the event a specimen's flaw depth could possibily rupture before burst due to through-wall or near through-wall flaws, a metal reinforcing foil made of brass and approximately 0.004 to 0.006 inches thick is used. The foil is installed over the plastic bladder after the stretching process and lubricated using vacuum grease to prevent premature extrusion of the bladder through the defect opening before actual tube burst could occur. In this test program, brass foil is used, having a thickness of 0.006 inches and is positioned such that approximately 0.500 inch on either side of the flaw.

Because of the reasonably high toughness of Alloy 600 tubing, a narrow EDM slit is assumed to be an adequate stimulant of natural stress corrosion or fatigue cracks in terms of affecting the tube burst properties. The burst pressure is dependent on the plastic flow properties of the tubing instead of the fracture toughness. Plastic collapse is reached before the onset of crack tearing. Yielding of the tube in the vicinity of the crack or slit limits the pressure bearing capacity of the tube. If this limit or collapse pressure is maintained, the crack opening will continue to increase until at some point crack tearing develops. The point where crack tearing develops does depend on the fracture toughness of the material but in the geometries of interest here, the maximum pressure capacity of the cracked tube is dominated by plastic response.

Westinghouse - Windsor Burst Testing Procedure

Burst and leak rate testing of steam generator tubes at the Westinghouse-Windsor Connecticut laboratories are conistent with EPRI guidelines. The basic procedure has been used since 1989 to test a variety of specimens representing both tubes removed from operating steam generators and flawed tubes prepared in the laboratory. Flaws tested have included ID and OD initiated SCC, intergranular attack, wastage, wear scars, and pits as well as non-flawed specimens. In addition, specimens with EDM notches (rectangular and more complex shapes) to simulate crack-like flaws have been tested.

The Westinghouse-Windsor approach to pressurizing specimens has traditionally used an air operated positive displacement pump. The earliest tests used a hand operated pump and, most recently, an accumulator has been used for a few specimens tested at very high pressurization rates. In almost all cases, the pressurizing medium has been de-ionized water. For a few specimens with axial flaws, a non-reinforced bladder was used to apply hoop loads. This approach resulted from qualification testing for in situ pressure testing of through-wall flaws with leakage in excess of pump capacity. Hoop strains in the bladder tests were identical to those recorded in capped tube hydro-tests. The positive displacement pump results in a small pressure spike whenever the pump strokes. The spike is more significant during leak testing, especially at higher leak rates where significant pressure fluctuations can occur.

Reinforcing bladders are generally not used for burst or leak rate testing. If leakage beyond pump capacity occurs, the test is interrupted and a bladder is inserted and the test resumed. The bladder is a section of tygon™ tubing. A reinforcing foil of brass (0.004 to 0.008 inch thick) or stainless steel (0.002 to 0.004 inch thick) is positioned over the flaw to prevent the bladder from extruding the flaw. The bladder is lubricated with vacuum grease for ease of insertion into the specimen. The EPRI guidelines requirement a reduction in observed burst pressure of 5 percent to account for the reinforcing effect of the bladder. As part of a CEOG study, Westinghouse-Windsor did evaluate the effect of a bladder on burst pressure by testing 6 part through-wall specimens with a bladder and 6 identical specimen without a bladder. All specimens were from the same tube of Alloy 600. There was not a significant effect (less than 5 percent difference).

A low pressurization rate was used for most of the burst tests conducted by Westinghouse-Windsor. Burst tests are actually conducted as leak rate tests, regardless of flaw type and characteristics. Specimens were slowly pressurized (target of 2000psi/minute) to a pressure simulating NODP and held for 5 minutes to observe for leakage. If leakage occurred, a leak test of up to 5 minutes duration at this pressure was conducted. Pressure was then slowly raised to MSLB pressure and held for 5 minutes to observe for leakage. A leak rate test of up to 5 minutes duration was conducted if leakage was present. The specimen was then pressurized slowly to $3DP_{NO}$ and held for 5 minutes after which the specimen was slowly pressurized to burst. An

objective of this procedure was to determine when leakage occurred (ligament tearing) and to determine if leakage at DP_{NO} occurred in specimens that were leaking at $3DP_{NO}$. Bladders were not used except as noted above.

During burst and leak testing, control of the pressure and pressurization rates is manual. Pressure can be observed by the operator on a pressure gage which taps into the test system tubing about 12 inches downstream of the pump. In addition, pressure is recorded by a transducer at the same location. Transducer output is to an X-Y recorder and to a computerized data acquisition system.

The Westinghouse-Windsor in situ pressure test system is similar to the laboratory burst test system. This system has the capability of conducting either localized tests (only a small section of the tube is pressurized or a full tube test. An air operated positive displacement pump provides pressurized de-ionized water to the tubes being tested. This system can supply 0.5 gpm of water at MSLB pressures. A high flow system for larger leaks can supply approximately 3.5 gpm at MSLB conditions.

The tools for localized testing were qualified to demonstrate that strains (axial and hoop) produced by the tools were identical to those in a capped tube hydro test. Pressure drop through the tubing and tools during leak tests are estimated based on qualification tests. If leak rates exceed the capabilities of the pump or accumulator in the high flow system, the burst strength of tubes can still be determined by positioning an integral bladder over the defect. Qualification test of this feature demonstrated that the hoop and axial strains were the same as in a capped tube hydro test.

During ISPT, pressurization is manually controlled and occurs at a slow rate. The current guidelines do not specify a pressurization rate. Pressures are indicated by a pressure gage and by the output of a pressure transducer which goes to a computerized data acquisition system. The taps for the devices are about 12 inches downstream of the pump. A pressure versus time record for each specimen tested is obtained and maintained.

Each ISPT has designated hold points and times. Typically a tube is pressurized to an adjusted DP_{NO} (adjusted for temperature and locked tube effects, as applicable) and held for some time. The tube is then slowly pressurized to MSLB pressure and held for the designated time and then pressurized to $3DP_{NO}$ and held. Other hold points may be designated by the utility. Hold times in the over 400 tests that Westinghouse-Windsor has conducted to date has varied form as little as 30 seconds to over 20 minutes. The most commonly used hold times are 2 and 5 minutes. If leakage occurs, a leak rate test of up to 5 minutes duration is conducted. The duration is restricted by the amount of water in the reservoir. Leak rates are determined by the number of pump strokes over a given time period or, for the high flow system, a flow meter downstream of the accumulator

Electricité de France and Framatome, France Burst Testing Procedure

There are two significant sources of burst pressure data from France. In both cases the data were obtained for use by Electricité de France and Framatome and later shared with EPRI. The results from extensive testing of tubes with and without various degradation morphologies, other than ODSCC at TSP intersections, is contained in Reference NP-6865-L, "Steam Generator Tube Integrity Volume 1: Burst Test Results and Validation of Rupture Criteria (Framatome Data)," EPRI, Palo Alto, CA (June 1991). The data from tests performed in support of application of alternate repair criteria (ARC) for ODSCC indications at TSPs are best summarized in Reference NP-7480-L, Addendum 1, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (November 1996).. Various additional references were used for the compilation of the final set of data which are recorded in Reference NP-7480-L, Addendum 1, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (November 1996)., but do not need to be cited here.

All of the burst pressure data used to support the ODSCC ARC were obtained from tests which were performed per the Reference D.5004/BRD/RB.90.126, Version 2, "Procedure for Performance and Analysis of Opening Tests by Placing Tubular Test-Pieces Under Internal Pressure," Electricité de France, Production Transport Laboratories Group, Paris, France (December 2, 1992). procedure specification. Additional information was provided via Reference Cattant, François, "Ref.: Steam Generator Tube Burst Testing Procedure," electronic mail to Robert Keating (Westinghouse), Electricité de France, Paris, France (August 8, 2000). The procedure provides instructions to be followed for tests performed both at ambient and elevated temperature. Specimens are pressurized with a hydraulic fluid, either water or oil, and a mandrel may be placed inside of the specimen to limit the fluid volume being pressurized. A 0.004" (0.1 mm) thick band of stainless steel foil is glued (the glue may be omitted under specific circumstances) to the inside of the specimen before testing to seal any throughwall degradation and thereby prevent loss of pressure during the test. The pressurizing medium in the tube itself is mastic (putty), which is pushed by the water in the machine. The test specimen is pressurized smoothly at a rate that does not increase the hoop stress more than 4.4 ksi/s, or about 500 psi/s for 3/4" and 7/8" diameter SG tubes. The procedure plans for a pressurization rate of 8 bar/s (116 psi/s), but practically, the device is set to achieve 0 to 2500 bars (0 to 36,300 psi) within 600 seconds, which means a rate of 4.2 bar/s (60.4 psi/s). The testing machine is equipped with a hydraulic control system. There are two (2) pressure transducers; one operates in the 0 - 200 bars range (0 - 2900 psi), the other can measure up to 4000 bars (58,000 psi). The pressure transducers are located upstream of the specimen, between the pressure multiplier and the specimen. Each of the pressure transducers can be isolated by valves. The final reported burst pressure is 85% of the measured value to account

for the presence of the foil. The crack profiles are examined by scanning electron microscopy (SEM) after the completion of the test.

There is a very extensive amount of data in the references and various geometries of machined degradation were investigated, i.e., V-notches, EDM slits, machined flats, uniform thinning, and lunar wastage (crescent shaped in the plane perpendicular to the axis of the tube). Typical burst testing rates were on the order of 35 to 60 psi/s, depending on the tests being conducted. The specimens may have been lined with a plastic bladder and the bladder may have been reinforced with a band of metal foil. The interface between the tube and the foil is lubricated to the extent that no adjustment is required from the measured burst pressure to the reported burst pressure.

Laborelec Burst Testing Procedure

Much of the information regarding burst testing performed at or for Laborelec may be gleaned from the Reference Hernalsteen, P., "The influence of Testing Conditions on Burst-Pressure Assessment for Inconel Tubing," IJPVP, **52**, pp. 41-57 (1992). discussion of the influence of test conditions. In addition, much information regarding test conditions is available in the EPRI report on the Belgian approach to PWSCC [NP-6626-SD, "Belgian Approach to Steam Generator Tube Plugging for Primary Water Stress Corrosion Cracking," EPRI, Palo Alto, CA (March 1990).], the EPRI throughwall burst pressure report [TR-105505, "Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks," EPRI, Palo Alto, CA (October 1997).] and Section *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates* of this document. Testing conditions and procedures at Laborelec are similar to those of the other laboratories. A variety of reinforcing methods have been used with reported burst pressures being reduced from the measured burst pressures to account for the strengthening effect of the reinforcement, e.g., a 5% reduction is applied to account for the presence of a plastic bladder and reinforcing foil. The pressurization rates tend to be slower than those used by Westinghouse in Pittsburgh, but are likely on the order of the rates used by Westinghouse in Windsor.

Freespan Throughwall Axial Cracking

Throughwall cracking is as the name implies, 100% through the wall of the SG tube. The evaluation model is described in Reference TR-105505, "Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks," EPRI, Palo Alto, CA (October 1997). The model is empirical, employing an equation form that is simpler than preceding models, some of which were based on theoretical evaluations or numerical solutions to the governing differential equations. The model is consistent with theoretical models, but fits the data better because of the freedom afforded in empirically fitting the coefficients.

The definitive work on measuring the burst pressure of SG tubes with throughwall cracks was performed by Laborelec (Paul Hernalsteen) [Hernalsteen, P., "The influence of Testing Conditions on Burst-Pressure Assessment for Inconel Tubing," IJPVP, **52**, pp. 41-57 (1992).]. In addition to performing multiple tests on laboratory specimens to investigate the effect of various test parameters, a series of specimens were tested to failure using the large capacity pumps at the Schelle fossil power plant in Belgium. The tests were performed without benefit of any lining to prevent leakage from the specimens. The pressure was gradually raised until unstable crack extension was attained. The results from those tests over a wide range of crack lengths confirmed the supposition that burst pressures were greater than had been reported by Westinghouse [TR-105505, "Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks," EPRI, Palo Alto, CA (October 1997).] from tests conducted at high rates and using a plastic bladder to prevent leakage prior to crack extension. They also confirmed that burst pressures were slightly lower than had been reported by others [TR-105505, "Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks," EPRI, Palo Alto, CA (October 1997).], including Laborelec, that used an unlubricated metal foil, like stainless steel, to line the plastic bladder to prevent its expulsion through the crack opening.

Other testing performed by Laborelec [Hernalsteen, P., "The influence of Testing Conditions on Burst-Pressure Assessment for Inconel Tubing," IJPVP, **52**, pp. 41-57 (1992).] did confirm that burst pressures could be artificially increased by about 15% by increasing the pressurization rate by three orders of magnitude, say 200 psi/s to 200000 psi/s. Hernalsteen also reported reductions in measured burst pressure by about 5% by decreasing the pressurization rate from say 200 psi/s to 20 psi/sec, but concluded that the result was due to *creeping* deformation of the liner and foil and crack flanks and not extension of the crack in the tube specimen.

A typical pressure versus time history for a Laborelec laboratory test is shown on Figure *Freespan Throughwall Axial Crackin*-1 and that for one of the tests performed at Schelle is on Figure *Freespan Throughwall Axial Crackin*-2. The pressurization rates from those specific tests are also illustrated on the figures. It is apparent that relatively slow pressurization rates were actually achieved during the tests. The pressure versus time histories from the other Schelle tests were requested from Laborelec, however, the response indicated that the original data are no longer available, but that all of the tests were conducted similar to the one plotted [Laire, Charles, "Re: Request for Schelle Data," elctronic mail, Laborelec, Belgium (July 06, 2000).]. The EPRI database curve for throughwall cracking is shown on Figure 6 along with the Schelle data. In fact, the Schelle data were included in the database for the regression analysis. The Laborelec laboratory and Schelle data are expected to be free from rate effects because the tests were conducted rather slowly, and thus indicate that the EPRI database for throughwall cracks is free from rate effects because there is no apparent difference in the test results (there are forty-three fast-rate and sixteen slow-rate test results illustrated on Figure *Freespan Throughwall Axial Crackin*-3).

Additional data were reported by EdF [NP-6865-L, "Steam Generator Tube Integrity Volume 1: Burst Test Results and Validation of Rupture Criteria (Framatome Data)," EPRI, Palo Alto, CA (June 1991).] for throughwall cracks with lengths of 0.276" (18 & 435 psi/s), 0.591" (18 to 234 psi/s), and 1.18" (17 to >242 psi/s). The results of the tests demonstrated that there was no dependence of the burst pressure on the pressurization rate for the range of 1.2 to 435 psi/s (5 to 1800 bar/min).

Finally, the data reported by the Battelle National Laboratory in Reference NUREG/CR-0718, "Steam Generator Tube Integrity Program Phase I Report," Prepared by the Pacific Northweat Laboratory for the U.S. Nuclear Regulatory Commission, Rockville, MD (September 1979). were obtained at a rate of 30 psi/s. These burst test results are not at variance with data obtained from other test programs at other laboratories, including Westinghouse, Combustion Engineering and Framatome.

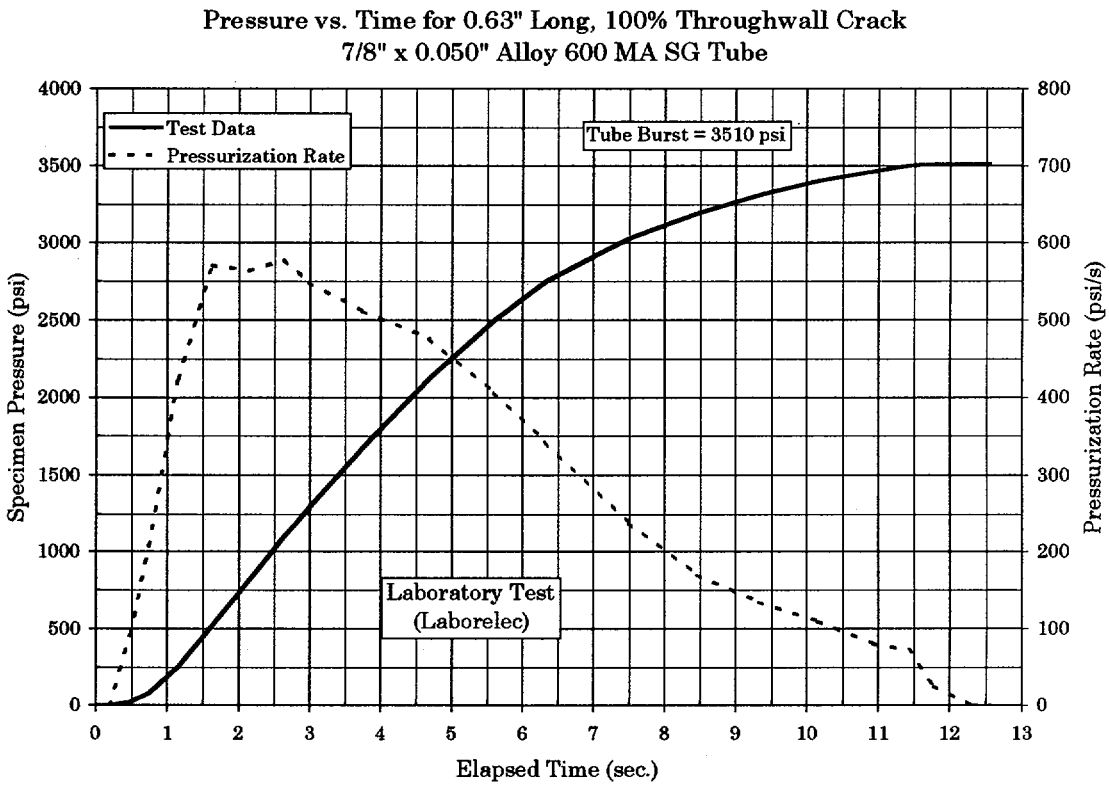

Figure *Freespan Throughwall Axial Crackin*-1: Pressure vs. time plot for Laborelec laboratory test.

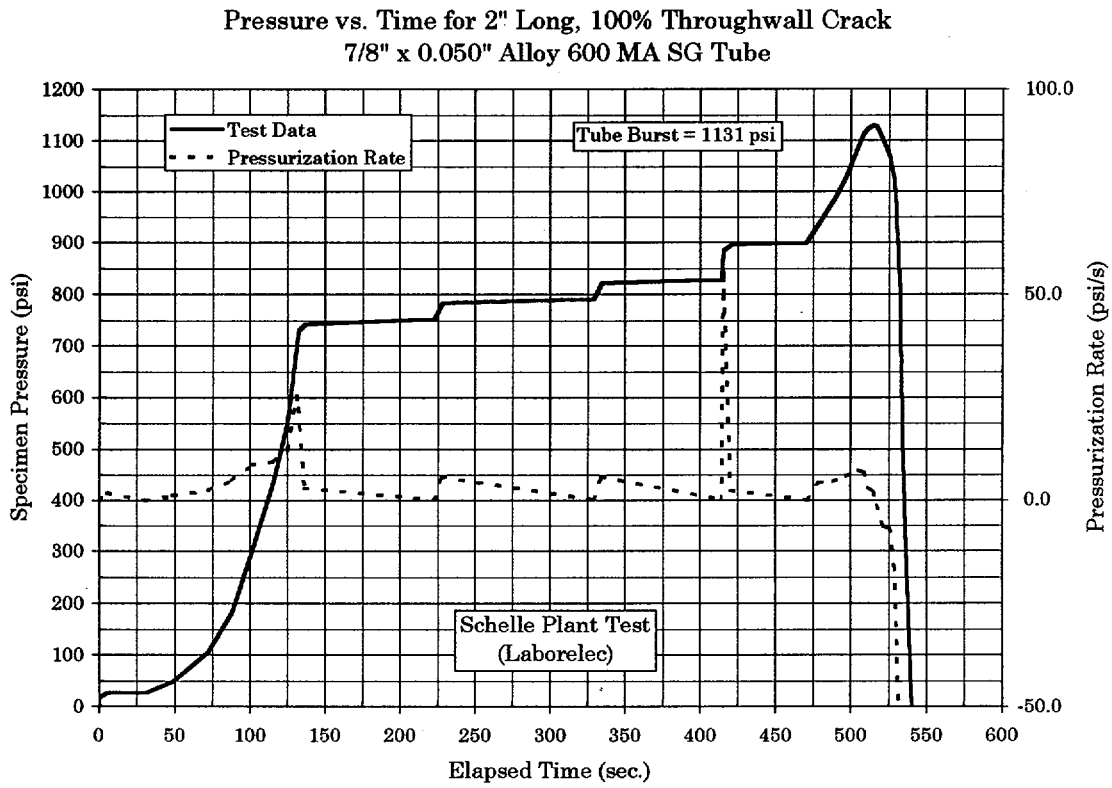

Figure *Freespan Throughwall Axial Crackin*-2: Pressure vs. time plot for Laborelec Schelle test

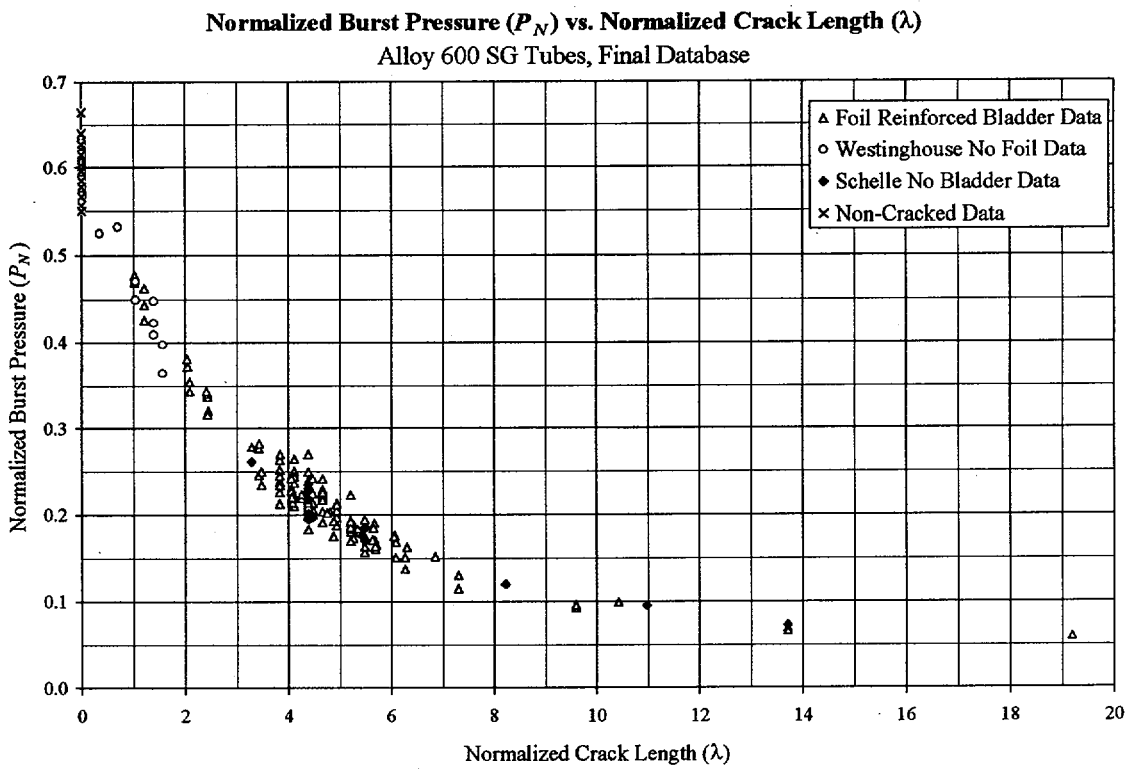

Figure *Freespan Throughwall Axial Crackin*-3: Throughwall Crack Burst Pressure Database [TR-105505, "Burst Pressure Correlation for Steam Generator Tubes with Throughwall Axial Cracks," EPRI, Palo Alto, CA (October 1997).]

Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates

Although ODSCC is also part-throughwall cracking, it has been treated separately because of the nature of the degradation and its location, and the fact that the burst pressure has been correlated to the voltage amplitude from a bobbin coil eddy current examination. Burst tests of model boiler and pulled tube specimens were, and may continue to be, performed at several different laboratories in at least three countries, the United States, Belgium and France. Although the testing procedures are different, the test results are similar. Tests performed in the US were typically performed at a rate of about 1500 to 2000 psi/s, while those in France were at a maximum of 500 psi/s and those in Belgium were most likely performed at a rates significantly less than 200 psi/s, see Reference NSD-EPRI-0646, "EPRI Guidelines for Leak & Burst Testing of SG Tubes," Westinghouse Electric Company, Madison, PA (July, 11, 1994). and the discussion of Section *Laborelec*. This latter statement is supported by the testing rate reported for throughwall degradation, see Figure *Freespan Throughwall Axial Crackin*-1.

The ODSCC voltage limits are 1.0 and 2.0 V for plants with 3/4" and 7/8" diameter tubes respectively. The comparison data for the burst resistance of tubes subject to the ARC is illustrated on Figures *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1 and -2 for 3/4" and 7/8" diameter tubes respectively. Statistical testing of the data is summarized in Tables *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1 and –2. The comparisons are based on standard statistical tests wherein a series of regression models are fit to the data independently and combined. Four models are considered to compare the data as follows:

1) General Model – The data are representative of populations for which both the intercept and slope of the regression lines are different. Four coefficients, two intercepts and two slopes, are obtained from the regression analysis of the data for this model.

2) Parallel Model – The data are representative of populations for which the intercepts are different, but the slopes are the same. Three coefficients, two intercepts and one slope, are obtained from the regression analysis of the data.

3) Concurrent Model – The data are representative of populations for which the intercepts are the same, but the slopes are different. Three coefficients, one intercept and two slopes, are obtained from the regression analysis of the data.

4) Coincident Model – The data are representative of the same population, so the intercepts and the slopes are the same. Two coefficients, one intercept and one slope are obtained from the regression analysis of the data for this model.

The first model is the most flexible in terms of trying to accommodate the variations in the data and results in the smallest value of the residual or error sum of squares (RSS). The fourth model is the most stringent and results in the largest value of the RSS.

Models two and three result in RSS values that are less than that of Model 4 and greater than the value from Model 1. A series of general *F*-tests are performed to determine if the improvement in reducing the RSS is statistically significant in changing from a more stringent model to a less stringent model, e.g., the significance of changing from Model 4 to Model 1. In other words, the results of the regression analyses are compared to determine the improvement in the residual sums of squares by adding terms to the model. For example, if all of the data are treated together, Model 4, the regression model of the burst pressure, P_B, as a function of the bobbin amplitude, V, in Volts is given as,

$P_B = b_0 + b_1 \log(V)$ *Model 4 (Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates-1)*

where b_0 and b_1 are coefficients determined from the regression analysis.

The results from the statistical tests are summarized in Tables *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1 and -2 for 3/4" and 7/8" diameter tubes respectively. The results from testing of pulled tube sections by Westinghouse and Laborelec were compared using the data from 3/4" diameter tubes [NP-7480-L, Addendum 3, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, Addendum 3, 1999 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (May 1999).]. The results are quite conclusive in indicating that there is no difference in results obtained from an analysis of the Laborelec data relative to results obtained from an analysis of the Westinghouse data. The table lists the test performed, the models compared, the RSS values calculated, the degrees of freedom of the models, the calculated *F*-test values, and the probability of exceeding those values. Test 2, for example, compares the use of Model 4 against the use of Model 1. The calculated value of *F* is 0.203 and the probability of obtaining a value of *F* greater than 0.203 if the null hypothesis that the intercepts and slopes are equal is true is 82%. The other comparisons all result in values of *F* that would be expected to occur randomly with a probability of greater than 50%. The conclusion that is indicated by the test results is that there is no difference between the set of data from testing performed by Laborelec and testing performed by Westinghouse. The conclusions from the analysis are also visibly apparent from an examination of the information presented on Figure *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1 where the two regression lines are almost coincident. The Laborelec tests were performed at a pressurization rate that is significantly less than the Westinghouse rate, see Figure *Freespan Throughwall Axial Crackin*-1 for an example of their laboratory practice, therefore implying that the rate of pressurization has no effect on the test results.

The results from the comparison of the Electricité de France (EdF) data and the Westinghouse data are listed in Table *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-2. The results of statistical comparisons of these sets of data were first reported to

the NRC in 1996 [NP-7480-L, Addendum 1, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, 1996 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (November 1996).]. The results at that time led to the recommendation that the EdF data be omitted from the ODSCC database. A significant factor leading to that recommendation was the observation that burst pressure predictions based on the EdF data appeared to have a systematic bias to be higher than those based on the Westinghouse data, i.e., the intercepts of the regression lines were indicated to be significantly different. No significant difference in the slopes of the two regression equations was found. Efforts aimed at identifying the cause of the observation did not lead to any meaningful conclusions and the recommendation was not concurred with by the NRC staff. Because of the addition of data subsequent to the 1996 evaluation, the comparisons were repeated for this evaluation. The results are depicted graphically on Figure *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-2 and presented numerically in Table *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-2. The results are similar to the 1996 findings in that no difference in the slope is implied, but a significant difference in the intercept is implied. A comparison of Model 1 (different intercepts and slopes) against Model 4 (same intercept and slope) indicates a significant difference in the RSS obtained from the regression analyses. Further comparisons of Models 4 and 1 against Model 2 (different intercept and same slope) indicates that the results from the first comparison are due almost entirely to an implied difference in the intercept and no difference in the slope. In summary, the models indicated to be parallel. The EdF pressurization rate is significantly less than the Westinghouse pressurization rate, hence the data indicate no dependence of measured burst pressure on the pressurization rate.

There was no indication of any significant difference when the variances of the prediction errors from each of the models being evaluated for each tube size were compared. For the 3/4" diameter tubes, the standard deviations of the regression residual burst pressures were 0.85 and 0.92 ksi for the US and Belgian data respectively. The variance ratio is then 1.2 which corresponds to performing a two-sided F-test at a confidence level of only 33%. The critical value of F at a 95% confidence level would be 2.2. Thus, there is no implication that a null hypothesis of equal variances should be rejected. For the 7/8" diameter tube data, the variance ratio is 1.3 based on regression residual standard deviations of 0.93 and 0.81 ksi for the US and French data respectively. The variance ratio corresponds to performing a two-sided F-test at a confidence level of 49%. The critical value of F at a 95% confidence level would be 2.2. Again, no implication that a null hypothesis of equal variances should be rejected. For the analysis of the 3/4" diameter tubes, the standard deviation of the residuals of the US data was slightly larger than that of the Belgian data. For the analysis of the 7/8" diameter tubes, the situation was reversed relative to the French data. Hence, there was no consistent difference between the values obtained from the more rapid pressurization rates relative to the slower pressurization rates.

1. Ancillary information is also available to support the conclusions reached in the preceding paragraphs. Leak rate tests which involved significant hold times were performed at a differential pressure of 3000 psi and at a temperature of 600°F without observing any time-dependent effects [NP-7480-L, Addendum 3, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, Addendum 3, 1999 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (May 1999).]. Finally, it is also noted that the average burst pressure for tubes with ODSCC indications is about 7500 psi regardless of the size of the tubing [NUREG/CR-0718, "Steam Generator Tube Integrity Program Phase I Report," Prepared by the Pacific Northweat Laboratory for the U.S. Nuclear Regulatory Commission, Rockville, MD (September 1979).

2. NUREG/CR-2336, "Steam Generator Tube Integrity Program/Steam Generator Group Project," Prepared by the Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Rockville, MD (May 1990).

3. NUREG/CR-5117, "Steam Generator Tube Integrity Program Phase II Final Report," Prepared by the Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Rockville, MD (August 1988).

4. NP-7474, "Evaluation of Leak and Burst Characteristics of Roll Transitions Containing Primary Water Stress Corrosion Cracks," EPRI, Palo Alto, CA (May 1993).

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NP-7480-L, Addendum 3, "Steam Generator Tubing Outside Diameter Stress Corrosion Cracking at Tube Support Plates Database for Alternate Repair Limits, Addendum 3, 1999 Database Update," prepared by Westinghouse Electric Company for EPRI, Palo Alto, CA (May 1999).]. This result is to be compared to an imposed pressure during a postulated SLB event of about 2560 psi with a criterion value of about 3600 psi. Therefore, the margin to tube burst during postulated accident conditions is adequate for both tube sizes.

In conclusion, there is no systematic effect of the pressurization rate on the burst pressure to voltage correlations, either the slope or the intercept or the standard

deviation of the regression residual values, over the range of interest of application of the ARC, which is bracketed by the data.

Table *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1: Comparison of Burst Testing of 3/4" Diameter Tubes

Model Testing	Model	RSS	DoF	Test Description & Results
Test 1				
Null Hypothesis: Alt. Hypothesis:	4 2	40.1868 40.1307	62 61	Same intercept, same slope. Different intercept, same slope.
Significance:	F / Pr(>F)	0.085	77.1%	**Different intercept NOT indicated.**
Test 2				
Null Hypothesis: Alt. Hypothesis:	4 1	40.1868 39.9164	62 60	Same intercept, same slope. Different intercept, different slope.
Significance:	F / Pr(>F)	0.203	81.7%	**Differences NOT indicated.**
Test 3				
Null Hypothesis: Alt. Hypothesis:	4 3	40.1868 39.9453	62 61	Same intercept, same slope. Same intercept, different slope.
Significance:	F / Pr(>F)	0.369	54.6%	**Different slope NOT indicated.**
Test 4				
Null Hypothesis: Alt. Hypothesis:	2 1	40.1307 39.9164	61 60	Different intercept, same slope. Different intercept, different slope.
Significance:	F / Pr(>F)	0.322	57.3%	**Different slope NOT indicated.**

Table *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-2: Comparison of Burst Testing of 7/8" Diameter Tubes

Model Testing	Model	RSS	DoF	Test Description & Results
Test 1				
Null Hypothesis: Alt. Hypothesis:	4 2	65.8347 60.1002	77 76	Same intercept, same slope. Different intercept, same slope.
Significance:	F / Pr(>F)	7.252	0.9%	**Different intercept is very likely.**
Test 2	Model	RSS	DoF	
Null Hypothesis: Alt. Hypothesis:	4 1	65.8347 60.0955	77 75	Same intercept, same slope. Different intercept, different slope.
Significance:	F / Pr(>F)	3.581	3.3%	**Reject Null Hypothesis.**
Test 3	Model	RSS	DoF	
Null Hypothesis: Alt. Hypothesis:	4 3	65.8347 62.5959	77 76	Same intercept, same slope. Same intercept, different slope.
Significance:	F / Pr(>F)	3.932	5.1%	**Null Hyp. NOT rejected.**
Test 4	Model	RSS	DoF	
Null Hypothesis: Alt. Hypothesis:	2 1	60.1002 60.0955	76 75	Different intercept, same slope. Different intercept, different slope.
Significance:	F / Pr(>F)	0.006	93.9%	**Different slope NOT likely!**

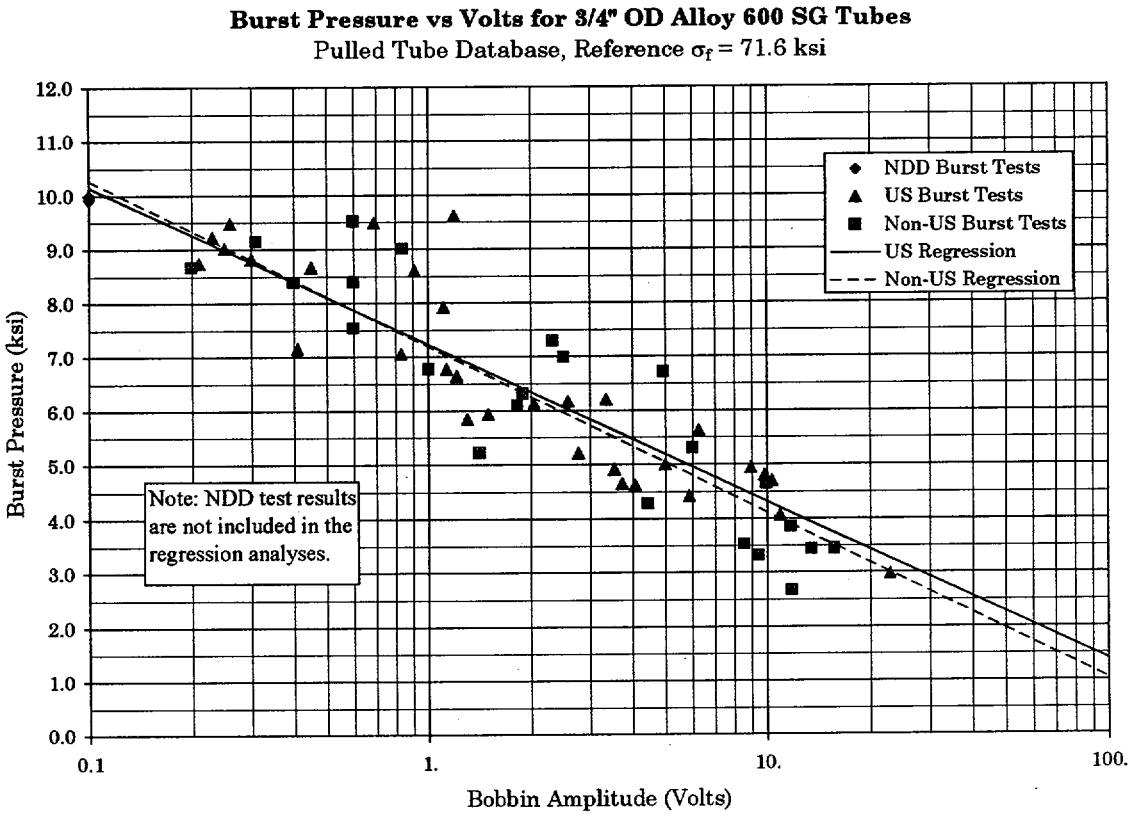

Figure *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-1:ODSCC ARC Data for 3/4" Diameter Tubes

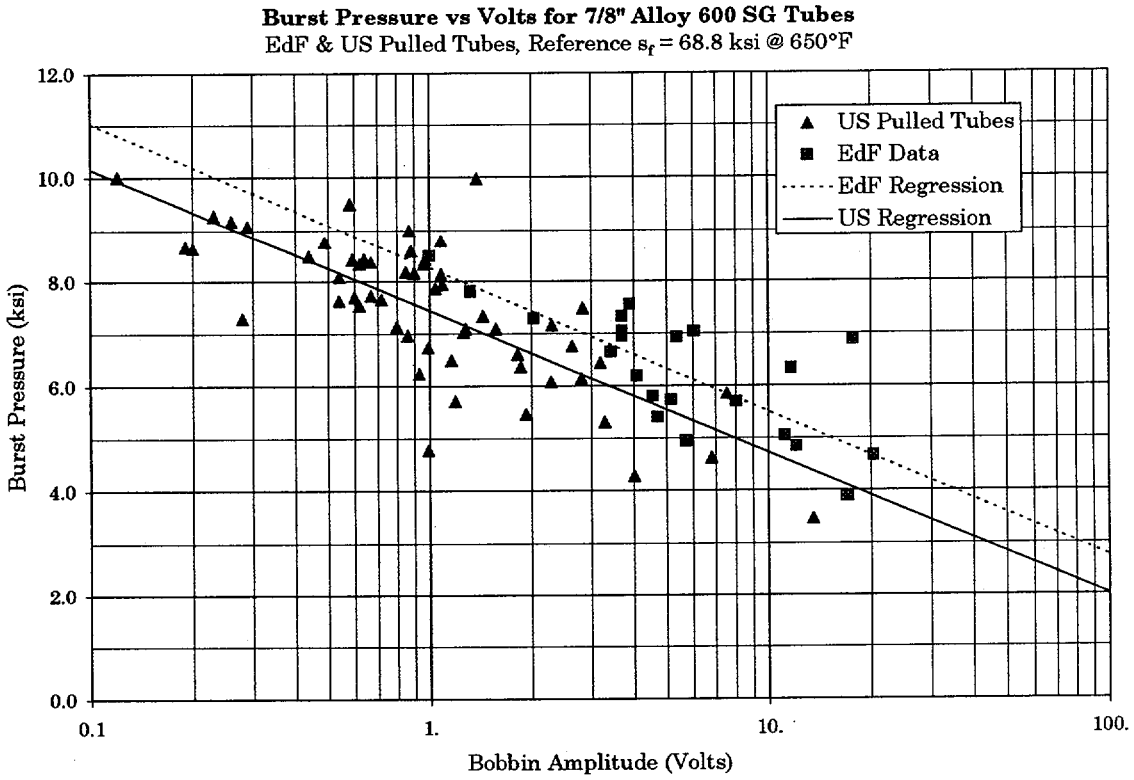

Figure *Bobbin Voltage ARC, Drilled 3/4" Thick Tube Support Plates*-2: ODSCC ARC Data for 7/8" diameter tubes.

Conclusions

The results from the previous discussions indicate no significant dependence of the burst pressure of SG tubes on pressurization rate for through wall cracks and slits or for ODSCC ARC specimen. Fast and slow pressurization rate tests, as described above, have previously been performed with no significant differences observed in the resulting burst pressures.

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ENCLOSURE 2

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
REVISED TECHNICAL SPECIFICATION PAGES

5.7 Procedures, Programs, and Manuals

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

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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

ADD

d) Indications left in service ***at the flow distribution baffles and tube support plate elevations*** 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.

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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.

(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.

f. Acceptance Criteria

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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 ***flow distribution baffles and*** 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.

INSERT A

(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 ***and flow distribution baffles.***

3. If indication are identified that extend beyond the confines of the tube support plate ***and flow distribution baffles.***

4. If indications are identified at the tube support plate ***and flow distribution baffle*** 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 to page 5.0-16

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 ***(including the flow distribution baffles)*** 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 to page 5.0-17

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 ***flow distribution baffles and*** tube support plate***s*** 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 ***flow distribution baffles and*** tube support plate***s*** 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 ***flow distribution baffles and*** tube support plate***s*** 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 ***("Tennessee Valley Authority, Watts Bar***

Nuclear Power Plant Unit 1, Application for Implementation of Voltage Based Repair Criteria, Westinghouse Steam Generator Tubes Affected by ODSCC at TSPs," J. W. Irons, Revision 0, 1/12/00) 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.

INSERT C to page B 3.4-74

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 ***and flow distribution baffles.*** 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 ***and flow distribution baffle.*** 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 ***and flow distribution baffles.***

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

WATTS BAR NUCLEAR PLANT UNIT 1
PROPOSED TECHNICAL SPECIFICATION CHANGE WBN-TS-99-014
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
MAIN STEAM LINE BREAK DOSE CALCULATION

ENCLOSURE 3

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
MAIN STEAM LINE BREAK (MSLB) ACCIDENT ANALYSIS

In a letter dated July 24, 2000, NRC requested additional information regarding the steam generator alternate repair criteria for axial outside diameter stress corrosion cracking submitted by TVA on April 10, 2000. NRC stated that the preliminary results of the staff's assessment of the methodology utilized by TVA to support the ARC amendment appeared to be unique and quite varied from the existing Standard Review Plan (SRP) and industry guidance and found the analysis unacceptable. Therefore, TVA has revised the analysis to conform to the SRP and the industry guidance. This analysis is provided as Attachment 2 to this enclosure. Attachment 1 to this enclosure provides a summary of the assumptions for the main control room dose (General Design Criteria (GDC) 19) analysis and the 10 CFR 100 offsite dose analysis. Below is TVA's response to NRC's request for additional information.

1. The calculations in WBNAPS3-077 are based upon reactor coolant activity levels from ANSI/ANS-18.1-1984. It would appear that the calculations should have been based upon the values in Final Safety Analysis Report (FSAR) Table 11.1 -2 or the values in Table 11.1 -2 needed to be changed to incorporate the ANSI/ANS-18.1-1984 values. Utilization of the ANSI values underestimates the quantity of each iodine isotope in reactor coolant relative to the existing values of Table 11.1-2.

   Response

   FSAR Section 11.1.1, "*Historical Design Model for Radioactivities in Systems and Components*" states that Tables 11.1-1 through 11.1-5 present results of the original Westinghouse Design Calculations. The results are presented as background and are superseded by calculations described in Section 11.1.2 and Tables 11.1-6 and 11.1-7. This section states that the specific activities for primary and secondary sides are calculated by ANSI/ANS-18.1-1984 methodology and given in Table 11.1-7. Table 11.1-6 provides the parameters used to describe WBN.

2. The calculations to determine secondary side activity following a year of normal operation accounting for steam generator blowdown and other removal mechanisms seems unnecessary and inconsistent with the guidance in the SRP for the MSLB. In the assessment of the consequences, the starting point for the analysis should be the assumption that the secondary side activity in the steam generators is at the technical specification value, which is typically 0.1

μCi/g. TVA's conclusion on page 10 of WBNAPS3-077, that because the secondary coolant activity level eventually becomes equal to or greater than 0.1 μCi/g meets the intent of the SRP, is erroneous and a misapplication of the SRP assumptions.

Response

TVA has revised the WBN Unit 1 MSLB calculation (WBNAPS3-077 Revision 5) and it now assumes the steam generator secondary side inventory of water is at the WBN Technical Specification LCO 3.7.14, "Secondary Specific Limit," I-131 equivalent iodine limit of 0.1 μCi/g.

3. It appears that TVA's MSLB analysis only involved the analysis of one case. To the pre-existing spike case of 60 μCi/g was incorporated an accident initiated spike which resulted in a release rate of activity to reactor coolant which was based upon the ANSI/ANS -18.1-1984 activity levels in reactor coolant. This too is contrary to the guidance in the SRP which involves assessing two cases. Based upon those two cases, a determination is made as to which case is limiting. The two cases to be evaluated include the pre-existing spike case with reactor coolant activity level at 60 μCi/g of dose equivalent ^{131}I and the accident-initiated spike case where reactor coolant level activity is at 1 μCi/g with a spiking factor resulting in release rate 500 times the release rate necessary to maintain reactor coolant activity level at 1 μCi/g. The assessment which was performed for Watts Bar seems to be a hybrid of these two cases and one which may be overly conservative and burdensome especially if the steam generators continue to degrade. Utilization of TVA's methodology in the future may be limited and future increases in accident-induced leakage for the MSLB would most likely require a change in methodology. Application of a more common calculation methodology might best be performed now.

Response

TVA has revised the WBN Unit 1 MSLB calculation such that it now contains two cases:

1) a pre-accident iodine spike of 60 μCi/g I-131 equivalent iodine, and
2) an accident initiating iodine spike 500 times the production to maintain a steady state level of 1 μCi/g I-131 equivalent iodine in the reactor coolant system.

4. In the TVA calculation the release of steam from the faulted steam generator for the first 30 minutes, the entire contents of the steam generator were released when the steamline breaks, which is the appropriate assumption. However, it appeared that TVA assumed that the radioactivity of the steam released was at the activity level of the primary coolant associated with the primary to secondary leak rather than being associated with the secondary coolant at the technical specification value of 0.1 μCi/g. If the staffs assessment is accurate, this assumption results in a very conservative prediction of the consequences of a MSLB accident.

Response

WBN's MSLB calculation (see Attachment 2 of this enclosure) now shows the calculation of the step source for the initial secondary inventory of the steam generator. The Technical Specification limit of 0.1 μCi/g was utilized.

5. Ignoring the removal of iodine from reactor coolant via decay results in a lower value for the production rate of the various iodine isotopes and is inappropriate in determining release rates associated with the spiking factors.

Response

WBN MSLB calculation now considers decay rates in the reactor coolant system in determining the production of equivalent iodine in the accident initiating iodine spike case in the section for continuous sources.

6. The basis for the removal rate value of 0.0678/hr needs to be provided.

Response

The WBN MSLB calculation in Attachment 2 of this Enclosure considers the source removal rate as equaling the sum of decay, letdown rate, and primary to secondary leakage factors in the accident initiating iodine spike case in the section for continuous sources.

ENCLOSURE 3
ATTACHMENT 1

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

It has been previously established in Generic Letter 95-05 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 calculation 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 15.18 gallons per minute (gpm) (at standard temperature and pressure) primary to secondary leakage 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 15.18 gpm value is controlled by the control room inhalation 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 Low Population Zone (LPZ) and 2-hour LPZ offsite doses - the 2 hour Exclusion Area Boundary (EAB) is determined from the 2 hr LPZ dose.

Two methods of determining the resultant doses for a main steam line break were used:

1) A pre-accident iodine spike where the iodine level in the reactor coolant spiked upward to the maximum allowable limit of 60 μCi/gm I-131 equivalent (refer to WBN Technical Specification Limiting Condition for Operation (LCO) 3.4.16) just prior to the initiation of the accident.
2) The reactor coolant at the maximum steady state dose equivalent I-131 of 1.0 μCi/gm with an accident initiated iodine spike consisting of a 500 increase on the rate of iodine release from the fuel.

In both cases, the primary-to-secondary side leak is 150 gallons per day (gpd) in the unfaulted loops, and the secondary side activity is at the WBN Technical Specification LCO 3.7.14 limit of ≤0.1 μCi/gm dose equivalent I-131.

Assumptions for the Postulated MSLB accident

1. RCS Letdown maximum flow of 120 gpm is used.
2. RCS Letdown demineralizer efficiency is assumed to be 1 for iodines.
3. ANSI/ANS-18.1-1984 spectrum was used and was scaled up to 1 or 60 μCi/gm equivalent iodine.

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

4. Two cases were used. In the first, pre-Accident Iodine Spike of 60 μCi/gm dose equivalent I-131 in the RCS was used. In the second case, 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.
5. Primary side to secondary side leakage of 150 gpd standard temperature and pressure per steam generator in the intact loops.
6. It is assumed that the primary-to-secondary leakage mass release to the environment is 15.18 gpm standard temperature and pressure from the faulted loop.
7. 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 standard temperature and pressure primary-to-secondary leakage)
   4. total from the faulted steam generator (30 min - 8 hr), 42,290 lb standard temperature and pressure primary-to-secondary leakage
8. Iodine partition coefficients for steaming of steam generator water
   1. non-defective steam generators (initial inventory) and primary-to-secondary leakage, 0.01
   2. faulted steam generator (initial inventory and primary-to-secondary leakage), 1.0
9. Atmospheric dilution factors, X/Q, are as follows for LPZ (sec/m^3):
   1. 0-2 hrs - 1.41E-4
   2. 2-8 hrs - 6.68E-5
   3. 8-24 hrs - 4.59E-5
   4. 1-4 days - 2.04E-5
   5. 4-30 days - 6.35E-6
10. Atmospheric dilution factor, X/Q, for 2-hr EAB is 6.07E-4 sec/m^3.
11. Atmospheric dilution factors, X/Q, are as follows for control room (sec/m^3):
    1. 0-2 hrs - 4.07E-3
    2. 2-8 hrs - 2.47E-3
    3. 8-24 hrs - 1.97E-3
    4. 1-4 days - 1.37E-3
    5. 4-30 days - 9.84E-4

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
MAIN STEAM LINE BREAK DOSE CALCULATION SUMMARY

12. Main Control Room related assumptions
    1. Volume - 257,198 cu ft
    2. Makeup/pressurization flow - 711 cfm
    3. Recirculation flow - 3600 cfm
    4. Unfiltered intake - 51 cfm
    5. Filter efficiency - 95% first pass, 70% second pass, 0% for noble gases
    6. Isolation time, T=0
    7. ICRP-2 dose conversion factors
    8. Occupancy factors:
        1. 0-24 hrs - 100%
        2. 1-4 days - 60%
        3. 4-30 days - 40%

Main Steam Line Break Dose Calculation Results

15.18 gpm Primary-to-Secondary leakage	**Control Room Operator (rem)**	**SRP Guidance for GDC 19 Limits (rem)**	**30-Day LPZ (rem)**	**2-Hour LPZ (rem)**	**2-Hour EAB (Site boundary) (rem)**	**SRP Guidance for 10CFR100 Limits (rem)**
Accident Initiated Iodine Spike Case						
Gamma:	0.017	5	0.252	0.043	0.185	2.5
Beta:	0.153	30	0.061	0.010	0.042	30
Inhalation:	29.99	30	26.68	2.407	10.365	30
Pre-Accident Iodine Spike Case						
Gamma:	0.005	5	0.050	0.031	0.134	25
Beta:	0.057	30	0.014	0.008	0.033	300
Inhalation	13.780	30	11.750	4.795	20.642	300

ENCLOSURE 3
ATTACHMENT 2

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
MAIN STEAM LINE BREAK CALCULATION

QA Record

ORIGINAL

TVAN CALCULATION COVERSHEET

Title: Offsite and Control Room Operator Doses Due to a Main Steam Line Break	Plant WBN Unit 1/2	Page 1

Preparing Organization Mechanical Design	Key Nouns (For EDM) MSLB, Offsite, Control Room, STP, COROD, FENCDOSE
Calculation Identifier **WBNAPS3-077**	Each time these calculations are issued, preparer must ensure that the original (R0) RIMS/EDM accession number is filled in.

Applicable Design Document(s)	Rev	(for EDM use)	EDM Accession Number
NA	R0	Oct 25 1993	B26 931014 409
	R4		B26 000223 411
UNID System(s)	R5		T71 000909 802
NA	R6		

	R0	R4	R5	R6			
					Quality Related?	Yes ■	No ☐
DCN, EDC, NA	NA	50473	50473		Safety related? If yes, mark Quality Related yes	Yes ■	No ☐
Prepared	Marc C. Berg	Marc C. Berg	Marc C. Berg				
Checked	Regis M. Nicoll	Janusz W. Kot			These calculations contain unverified assumption(s) that must be verified later?	Yes ☐	No ■
Design Verified	Regis M. Nicoll	Janusz W. Kot			These calculations contain special requirements and/or limiting conditions?	Yes ☐	No ■
Approved	Richard J. McMahon	BGBriody			These calculations contain a design output attachment?	Yes ☐	No ■
Approval Date	10/13/93	2/18/2000	9/8/2000		Calculation Classification	Essential	
SAR Affected?	Yes ☐ No ■	Yes ■ No ☐	Yes ■ No ☐	Yes ☐ No ☐	Microfiche generated	Yes ■	No ☐
Revision applicability	Entire calc ■	Entire calc ■ Selected pgs ☐	Entire calc ■ Selected pgs ☐	Entire calc ☐ Selected pgs ☐	Number R5: TVA-F-C000316		

Statement of Problem:
Determine the offsite and control room operator dose due to a Main Steam Line Break (MSLB).

Abstract A Main Steam Line Break at the Watt Bar Nuclear Plant will result in a significant steam release to the environment. This calculation is performed to show that the offsite and control room operator doses do not exceed the 10CFR100 and 10CFR50 App.A GDC 19 dose limits. The computer code STP determines the activity releases. The STP output is used as input to computer codes FENCDOSE and COROD which determine the offsite and control room doses.

There are several cases modeled. The first case has a pre-accident iodine spike where the iodine level in the reactor coolant is at the maximum allowable of 60 μCi/gm I-131 equivalent. The second has the reactor coolant at the maximum steady state I-131 equivalent of 1 μCi/gm with an accident initiated iodine spike consisting of a 500 increase in the rate of iodine release from the fuel. In both cases, the primary to secondary side leak is 150 gpd in the unfaulted loops, and the secondary side activity is at the Technical Specification limit of 0.1μCi/gm and the primary to secondary side leak in the faulted loop is 10 gpm. Additional cases are performed with a maximum leakage, to be determined in this calculation by iteration, so as to not exceed the 10CFR100 or 10CFR50 App.A GDC 19 limits. Finally, the above cases were performed with 10 gpm leak (or maximum leakage with maximum allowable iodine concentration of 60*0.35 = 21 μCi/gm I-131, or with 0.35 μCi/gm I-131 equivalent with 500 factor of spiking.

With the Technical Specification limits of 1 μCi/gm I-131 equivalent steady state (and 60 μCi/gm maximum), the maximum allowable leak so as to not exceed the dose limits was determined to be 15.18 gpm in the faulted line. With the Technical Specification limits of 0.35 μCi/gm I-131 equivalent steady state (and 21 μCi/gm maximum) the maximum allowable leak so as to not exceed the dose limits was determined to be 45.70 gpm in the faulted line.

■ Microfilm and return calculation to Calculation Library: Address: ☐ Microfilm and return calculation to:	☐ Microfilm and destroy.

ORIGINAL

TVAN CALCULATION COVERSHEET

Title: Offsite and Control Room Operator Doses Due to a Main Steam Line Break	Plant WBN Unit 1/2	Page 1

Preparing Organization Mechanical Design	Key Nouns (For EDM) MSLB, Offsite, Control Room, STP, COROD, FENCDOSE
Calculation Identifier **WBNAPS3-077**	Each time these calculations are issued, preparer must ensure that the original (R0) RIMS/EDM accession number is filled in.

Rev	(for EDM use)	EDM Accession Number
R0	Oct 25 1993	B26 931014 409
R4		B26 000223 411
R5		
R6		

Applicable Design Document(s): NA

UNID System(s): NA

	R0	R4	R5	R6			
					Quality Related?	Yes ■	No ☐
DCN, EDC, NA	NA	50473			Safety related? If yes, mark Quality Related yes	Yes ■	No ☐
Prepared	Marc C. Berg	Marc C. Berg					
Checked	Regis M. Nicoll	[signature]			These calculations contain unverified assumption(s) that must be verified later?	Yes ☐	No ■
Design Verified	Regis M. Nicoll	[signature]			These calculations contain special requirements and/or limiting conditions?	Yes ☐	No ■
Approved	Richard J. McMahon	[signature]			These calculations contain a design output attachment?	Yes ☐	No ■
Approval Date	10/13/93	2/18/2000			Calculation Classification	Essential	
SAR Affected?	Yes ☐ No ■	Yes ■ No ☐	Yes ☐ No ☐	Yes ☐ No ☐	Microfiche generated	Yes ■	No ☐
Revision applicability	Entire calc ■	Entire calc ■ Selected pgs ☐	Entire calc ☐ Selected pgs ☐	Entire calc ☐ Selected pgs ☐	Number R4: TVA-F-C000296		

Statement of Problem:
Determine the offsite and control room operator dose due to a Main Steam Line Break (MSLB).

Abstract

A Main Steam Line Break at the Watt Bar Nuclear Plant will result in a significant steam release to the environment. The steam will contain radionuclides if a primary to secondary side leak occurs prior to the MSLB event. This calculation is performed to show that the offsite and control room operator doses do not exceed the 10CFR100 and 10CFR50 App.A GDC 19 dose limits. R1 is performed because X/Q values changed. R3 changes the primary to secondary side leak to 150 gpd/steam generator and 10 gpm in the faulted loop.

This calculation used the computer code STP to determine the activity releases. The STP output was used as input to computer codes FENCDOSE and COROD. Computer code FENCDOSE was used to determine the offsite dose. Computer code COROD was used to determine the control room operator dose. This calculation assumed that the reactor had operated with a primary to secondary side leak at the technical specification limit of 150 gpd/steam generator for a duration of 1 year in order to maximize the radionuclide inventory in the secondary side. The primary coolant consisted of expected reactor coolant activity levels (0.12% failed fuel) multiplied by 8 to bring the inventory up to design limits. The base STP model to determine the secondary side activity was taken from WBNAPS3-053. The STP model was modified to add the MSLB steam releases after 1 year of steady state operation with 150 gpd/steam generator primary to secondary side leakage and 10 gpm in the faulted loop. A iodine spike to increase the iodine concentration to 60 uCi/cc prior to the accident is applied. An accident initiated spiking factor of 500 increase in iodine release from the fuel to the reactor coolant was incorporated in the model. The results can be found in the text.

The offsite doses due to a MSLB are significantly less (<10%) than the 10CFR100 limits of 25 rem gamma, 300 rem beta, and 300 rem inhalation. The control room operator doses are also less than the 10CFR50 App.A GDC 19 limits of 5 rem gamma, 30 rem beta, and 30 rem inhalation.

■ Microfilm and return calculation to Calculation Library: Address:
☐ Microfilm and return calculation to:
☐ Microfilm and destroy.

ORIGINAL

TVAN CALCULATION COVERSHEET

Title Offsite and Control Room Operator Doses Due to a Main Steam Line Break	Plant WBN Unit 1/2	Page 1 of	

Preparing Organization Mechanical Design	Key Nouns (For EDM) MSLB, Offsite, Control Room, STP, COROD, FENCDOSE		
Calculation Identifier **WBNAPS3-077**	Each time these calculations are issued, preparer must ensure that the original (R0) RIMS/EDM accession number is filled in.		
	Rev	(for EDM use)	EDM Accession Number
Applicable Design Document(s) NA	R0	Oct 25 1993	B26 931014 409
	R1	May 15 1995	B26 950501 390
UNID System(s) NA	R2		B26 950720 359
	R3		B26 991117 408

	R0	R1	R2	R3			
					Quality Related?	Yes ■	No ☐
DCN, EDC, NA	NA	NA	NA	~~NA~~ 087 11/16/99 50473	Safety related? If yes, mark Quality Related yes	Yes ■	No ☐
Prepared	Marc C. Berg	Marc C. Berg	Marc C. Berg	Marc C. Berg			
Checked	Regis M. Nicoll	RMNicoll	DGFickey	RMNicoll	These calculations contain unverified assumption(s) that must be verified later?	Yes ☐	No ■
Design Verified	Regis M. Nicoll	RMNicoll	DGFickey	RMNicoll	These calculations contain special requirements and/or limiting conditions?	Yes ☐	No ■
Approved	Richard J. McMahon	FAKoontz, Jr.	PSSmith	[illegible]	These calculations contain a design output attachment?	Yes ☐	No ■
Approval Date	10/13/93	4-27-95	7/20/95	11/17/99	Calculation Classification	Essential	
SAR Affected?	Yes ☐ No ■	Yes ■ No ☐	Yes ■ No ☐	Yes ■ No ☐	Microfiche generated	Yes ☐	No ■
Revision applicability	Entire calc ■	Entire calc ■ Selected pgs ☐	Entire calc ■ Selected pgs ☐	Entire calc ■ Selected pgs ☐	Number		

Statement of Problem:
Determine the offsite and control room operator dose due to a Main Steam Line Break (MSLB).

Abstract

A Main Steam Line Break at the Watt Bar Nuclear Plant will result in a significant steam release to the environment. The steam will contain radionuclides if a primary to secondary side leak occurs prior to the MSLB event. This calculation is performed to show that the offsite and control room operator doses do not exceed the 10CFR100 and 10CFR50 App.A GDC 19 dose limits. R1 is performed because X/Q values changed. R3 adds alternate primary side leak cases of 5 and 10 gpm.

This calculation used the computer code STP to determine the activity releases. The STP output was used as input to computer codes FENCDOSE and COROD. Computer code FENCDOSE was used to determine the offsite dose. Computer code COROD was used to determine the control room operator dose. This calculation assumed that the reactor had operated with a primary to secondary side leak at the technical specification limit of 1 gpm for a duration of 1 year in order to maximize the radionuclide inventory in the secondary side. The primary coolant consisted of expected reactor coolant activity levels (0.12% failed fuel) multiplied by 8 to bring the inventory up to design limits. The base STP model to determined the secondary side activity was taken from WBNAPS3-053. The STP model was modified to add the MSLB steam releases after 1 year of steady state operation with 1 gpm primary to secondary side leakage. The results can be found in the text.

The offsite doses due to a MSLB are significantly less (<<10%) of the 10CFR100 limits of 25 rem gamma, 300 rem beta, and 300 rem inhalation. The control room operator doses are also significantly less than the 10CFR50 App.A GDC 19 limits of 5 rem gamma, 30 rem beta, and 30 rem inhalation.

■ Microfilm and return calculation to Calculation Library: Address: ☐ Microfilm and return calculation to:	☐ Microfilm and destroy.

TVA 10697

CALCULATIONS Q. Record

TITLE	Offsite and Control Room Operator Doses Due to a Main Steam Line Break	PLANT/UNIT	WBN 1/2

PREPARING ORGANIZATION	KEY NOUNS (Consult CCRIS LIST)
CE/MNE/Analysis	MSLB, Offsite, Control Room, STP, COROD, FENCDOSE

BRANCH/PROJECT IDENTIFIERS: WBNAPS3-077

APPLICABLE DESIGN DOCUMENT(S): NA

SAR SECTION(S): 15.5 | UNID SYSTEM(S): NA

Each time these calculations are issued, preparers must ensure that the original (R0) RIMS accession number is filled in.

Rev	(for RIM'S use)	RIMS accession number
R0	Oct 25 1993	B26 931014 409
R1	May 15, 1995	B26 950501 390
R2	AUG 03 1995 (13)	B26 '950720 359
R3		

Revision 0	R1	R2	R3
ECN No. (or indicate Not Applicable) NA	NA	NA	
Prepared Marc C. Berg	Marc C. Berg	Marc C. Berg	
Checked Regis M. Nicoll	RMNicoll	D. Fibey	
Reviewed Regis M. Nicoll	RMNicoll	D. Fibey	
Approved Richard J. McMahon	FAKoontz,Jr	PSSmith	
Date 10/13/93	4-27-95	20 July 95.	
Use form TVA 10534 if more space required. List all pages added by this revision	see rev log	see rev log	
List all pages deleted by this revision	see rev log	see rev log	
List all pages changed by this revision	see rev log	see rev log	

Safety-related? Yes ☑ No ☐

Calculation Revision: ☒ Entire Calculation ☐ Selected Pages

Statement of Problem

Determine the offsite and control room operator dose due to a Main Steam Line Break (MSLB).

ORIGINAL

These calculations contain unverified assumption(s) that must be verified later. Yes ☐ No ☑

Calculation contains special requirements or limiting conditions. Yes ☐ No ☑

Abstract

A Main Steam Line Break at the Watt Bar Nuclear Plant will result in a significant steam release to the environment. The steam will contain radionuclides if a primary to secondary side leak occurs prior to the MSLB event. This calculation is performed to show that the offsite and control room operator doses do not exceed the 10CFR100 and 10CFR50 App.A GDC 19 dose limits. R1 is performed because X/Q values changed

This calculation used the computer code STP to determine the activity releases. The STP output was used as input to computer codes FENCDOSE and COROD. Computer code FENCDOSE was used to determine the offsite dose. Computer code COROD was used to determine the control room operator dose. This calculation assumed that the reactor had operated with a primary to secondary side leak at the technical specification limit of 1 gpm for a duration of 1 year in order to maximize the radionuclide inventory in the secondary side. The primary coolant consisted of expected reactor coolant activity levels (0.12% failed fuel) multiplied by 8 to bring the inventory up to design limits. The base STP model to determined the secondary side activity was taken from WBNAPS3-053. The STP model was modified to add the MSLB steam releases after 1 year of steady state operation with 1 gpm primary to secondary side leakage. The results were:

	Control Room Operator [rem]	30-Day LPZ [rem]	2-Hour EAB [rem]
Gamma:	2.854E-5	2.631E-5	9.441E-5
Beta:	3.436E-4	1.866E-5	6.828E-5
Inhalation:	6.817E-4	6.014E-4	2.202E-3

The offsite doses due to a MSLB are significantly less (<<10%) of the 10CFR100 limits of 25 rem gamma, 300 rem beta, and 300 rem inhalation. The control room operator doses are also significantly less than the 10CFR50 App.A GDC 19 limits of 5 rem gamma, 30 rem beta, and 30 rem inhalation.

R2: 13 total pages

☐ Microfilm and store calculations in RIMS Service Center. Microfilm and destroy. ☐

☐ Microfilm and return calculations to: Address: NTB

TVAN CALCULATION RECORD OF REVISION

CALCULATION IDENTIFIER WBNAPS3-077

Title **Offsite and Control Room Operator Doses Due to a Main Steam Line Break**

Revision No.	DESCRIPTION OF REVISION
0	Initial Issue
1	Revision 1 is performed because the X/Q values have changed. R1 pages 6 and 7 are new, and all pages are renumbered. Only changed text have revision bars. pages added: 6, 7 pages changed: all pages deleted: none
2	Revision 2 was performed because of new control room makeup flow (711 cfm). pages added: none pages deleted: none pages changed: 1-7, 9, 13
3	Revision 3 was performed to incorporate larger Technical Specification primary to secondary side leakage as part of the alternate steam generator tube plugging project. Since the final leakage value has not been established, a 5 gpm and a 10 gpm case was performed. All pages were renumbered, however only pages with text changes will have revision bars and will be listed as having been changed: Pages added: none pages deleted: classification forms pages changed: 1-9, 12 R3: 13 total pages
4	Revision 4 was performed to change the iodine partition factor for the faulted steam generator from 100 to 1 to account for steam generator dryout. The steady state preaccident leakage is changed to 150 gpd/steam generator. A preaccident iodine spike of 60 uCi/gm is applied to the reactor coolant. An accident initiated spiking factor of 500 increase in iodine release from the fuel to the reactor coolant was incorporated in the STP model. This revision is part of the corrective action for WBN PER 99-017510-000. Pages added: new coversheet (page 1) Pages deleted: none Pages changed: 1a (old coversheet page 1), 1b (old page 1a), 2-9, 12 R4: 15 total pages
5	Revision 5 is performed to incorporate new iodine production rates, split the iodine spiking model into two separate cases, and also to perform an additional analysis for a maximum Technical Specification limit of 0.35 μCi/gm I-131 equivalent (steady state). The non-steady state maximum limit of 60 μCi/gm I-131 * 0.35 = 21 μCi/gm I-131 is also analyzed. Additionally, the maximum allowable primary to secondary side leakage is determined for all cases. Justification for usage of the ANSI-ANS 18.1-1984 spectrum is also provided. Due to the nature of the revision, all pages were changed, with significant additions (Appendix A, and Attachment 1). Pages added: all pages deleted: all pages changed: all R5: 27 total pages

TVAN CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

Calculation Identifier **WBNAPS3-077** Revision 5

Method of design verification (independent review) used:

1. Design Review ■
2. Alternate Calculation ❑
3. Qualification Test ❑

Comments:

This revision incorporated revised iodine production rates, divided the iodine spiking model into two separate cases (one for maximum iodine allowed during normal operation and one for a 500 times production rate occurring at the time of the event). The revision also determined the maximum primary to secondary leakage rate that would result in the control room and site dose just meeting the allowable dose (this was determined for the current Technical Specification iodine limit of 1 uCi/gm and for a lower limit of 0.35 uCi/gm). Appropriately conservative assumptions were documented. The calculation is technically adequate and the conclusions are reasonable.

Design Verifier

8-25-00
Date

TVAN CALCULATION TABLE OF CONTENTS

Calculation Identifier: WBNAPS3-077	Revision: 5	

TABLE OF CONTENTS

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TVAN COMPUTER INPUT FILE STORAGE INFORMATION SHEET

Document WBNAPS3-077	Rev. 5	Plant: WBN	

Subject:
Offsite and Control Room Operator Doses Due to a Main Steam Line Break

☐ Electronic storage of the input files for this calculation is not required. Comments:

■ Input files for this calculation have been stored electronically and sufficient identifying information is provided below for each input file. (Any retrieved file requires re-verification of its contents before use.)

The R1 input files are archived on FILEKEEPER under reference I.D.# 263447

The R2 input files are archived on FILEKEEPER under reference I.D.# 263715

The R3 input files are permanently stored in FILEKEEPER file # 302577

The R4 input files are permanently stored in FILEKEEPER file # 302740

The R5 input files are permanently stored in FILEKEEPER file # 303180

TVAN COMPUTER OUTPUT
MICROFICHE INFORMATION SHEET

Document WBNAPS3-077	Rev. 5	Plant: WBN	

Subject:
Offsite and Control Room Operator Doses Due to a Main Steam Line Break

Microfiche Number	Description
R0 TVA-F-G104614	R0 Computer Runs: APS77S STP source terms for MSLB APS77C COROD control room operator dose for MSLB APS77F FENCDOSE offsite dose for MSLB
R1: TVA-F-C000078	R1 Computer Runs: APS77C1 COROD control room operator dose for MSLB APS77F1 FENCDOSE offsite dose for MSLB
R2: TVA-F-C000119	R2 Computer Runs: APS77C2 COROD control room operator dose for MSLB
R3: TVA-F-C000294	R3: Computer Runs APS77S3A STP source terms for 5 gpm primary side leak APS77S3B STP source terms for 10 gpm primary side leak APS77C3A COROD control room dose 5 gpm leak APS77C3B COROD control room dose 10 gpm leak APS77F3A FENCDOSE offsite dose 5 gpm leak APS77F3B FENCDOSE offsite dose 10 gpm leak
R4: TVA-F-C000301	R4: Computer Runs APS77S4 STP source terms for 10 gpm primary side leak APS77S4X STP source terms to test leakage model APS77C4 COROD control room dose 10 gpm leak APS77F4 FENCDOSE offsite dose 10 gpm leak
R5: TVA-F-C000316	R5: Computer Runs APS77S5A STP source terms, 10 gpm leak, preaccident 60 μCi/gm I-131 spike APS77S5B STP source terms, 10 gpm leak, 1 μCi/gm I-131, 500 spike APS77S5C STP source terms, 15.18 gpm leak, preaccident 60 μCi/gm I-131 spike APS77S5D STP source terms, 15.18 gpm leak, 1 μCi/gm I-131, 500 spike APS77S5E STP source terms, 10 gpm leak, preaccident 21 μCi/gm I-131 spike APS77S5F STP source terms, 10 gpm leak, 0.35 μCi/gm I-131, 500 spike APS77S5G STP source terms, 45.70 gpm leak, preaccident 21 μCi/gm I-131 spike APS77S5H STP source terms, 45.70 gpm leak, 0.35 μCi/gm I-131, 500 spike APS77C5A COROD control room dose, 10 gpm leak, preaccident 60 μCi/gm I-131 spike APS77C5B COROD control room dose, 10 gpm leak, 1 μCi/gm I-131, 500 spike APS77C5C COROD control room dose, 15.18 gpm leak, preaccident 60 μCi/gm I-131 spike APS77C5D COROD control room dose, 15.18 gpm leak, 1 μCi/gm I-131, 500 spike APS77C5E COROD control room dose, 10 gpm leak, preaccident 21 μCi/gm I-131 spike APS77C5F COROD control room dose, 10 gpm leak, 0.35 μCi/gm I-131, 500 spike APS77C5G COROD control room dose, 45.70 gpm leak, preaccident 21 μCi/gm I-131 spike APS77C5H COROD control room dose, 45.70 gpm leak, 0.35 μCi/gm I-131, 500 spike APS77F5A FENCDOSE offsite dose, 10 gpm leak, preaccident 60 μCi/gm I-131 spike APS77F5B FENCDOSE offsite dose, 10 gpm leak, 1 μCi/gm I-131, 500 spike APS77F5C FENCDOSE offsite dose, 15.18 gpm leak, preaccident 60 μCi/gm I-131 spike APS77F5D FENCDOSE offsite dose, 15.18 gpm leak, 1 μCi/gm I-131, 500 spike APS77F5E FENCDOSE offsite dose, 10 gpm leak, preaccident 21 μCi/gm I-131 spike APS77F5F FENCDOSE offsite dose, 10 gpm leak, 0.35 μCi/gm I-131, 500 spike APS77F5G FENCDOSE offsite dose, 45.70 gpm leak, preaccident 21 μCi/gm I-131 spike APS77F5H FENCDOSE offsite dose, 45.70 gpm leak, 0.35 μCi/gm I-131, 500 spike note: the titles in the actual runs for the 15.18 gpm and 45.70 gpm runs are incorrect (the titles specify different flows than that used). The runs utilize the correct flow rates. The incorrect titles were not corrected since there is no impact on the results.

Purpose

The purpose of this calculation is to determine the offsite and control room operator dose due to a Main Steam Line Break (MSLB). The results will be used in FSAR ch.15.5 to show compliance with 10CFR100 and 10CFR50 App.A GDC 19. Revision 1 is performed because X/Q values have changed. Revision 2 is performed because the control room makeup flow changed from 325 to 711 cfm.

Revision 3 is performed to provide doses for alternate primary to secondary leakage rates. This is to support potential future Technical Specification changes. Revision 4 is performed as to correct the iodine partition factor in the faulted steam generator from 100 to 1 due to dry out and to incorporate an accident initiated iodine spike. The steady state leakage is changed to 150 gpd/steam generator and the maximum leakage after the accident in the faulted loop is 10 gpm. Revision 5 is performed to incorporate new iodine production rates, split the iodine spiking model into two separate cases, and also to perform an additional analysis for a maximum Technical Specification limit of 0.35 μCi/gm I-131 equivalent (steady state). The non-steady state maximum limit of 60 μCi/gm I-131 * 0.35 = 21 μCi/gm I-131 is also analyzed.

Introduction

A Main Steam Line Break at the Watt Bar Nuclear Plant will result in a significant steam release to the environment. The steam will contain radionuclides if a primary to secondary side leak occurs prior to the MSLB event. This calculation is performed to show that the offsite and control room operator doses do not exceed the 10CFR100 and 10CFR50 App.A GDC 19 dose limits.

This calculation uses the computer code STP (ref.3) to determine the activity releases. The STP output is used as input to computer codes FENCDOSE and COROD. Computer code FENCDOSE (ref.4) is used to determine the offsite dose. Computer code COROD (ref.5) is used to determine the control room operator dose. The base FENCDOSE and COROD models are taken from WBNTSR-008 (ref.9).

There are several cases modeled. The first case has a pre-accident iodine spike where the iodine level in the reactor coolant is at the maximum allowable (60 μCi/gm I-131 equivalent, ref.1). The second has the reactor coolant at the maximum steady state I-131 equivalent of 1 μCi/gm with an accident initiated iodine spike consisting of a 500 increase in the rate of iodine release from the fuel. In both cases, the primary to secondary side leak is 150 gpd in the unfaulted loops, and the secondary side activity is at the Technical Specification limit of 0.1μCi/gm and the primary to secondary side leak in the faulted loop is 10 gpm. Additional cases are performed with a maximum leakage, to be determined in this calculation by iteration, so as to not exceed the 10CFR100 or 10CFR50 App.A GDC 19 limits. Finally, the above cases were performed with 10 gpm leak (or maximum leakage, to be determined) with maximum allowable iodine concentration of 60*0.35 = 21 μCi/gm I-131, or with 0.35 μCi/gm I-131 equivalent with 500 factor of spiking. The relative isotopic spectrum is taken from WBNNAL3-003 (Reactor Coolant Activities in Accordance with ANSI/ANS-18.1-1984). Justification of the usage of this spectrum as opposed to the historical design spectrum as found in chapter 11 of the FSAR can be found in Appendix A.

Assumptions

1. The primary side to secondary side leakage is 150 gpd/steam generator, steady state, with 10 gpm (or maximum leak rate determined so as not to exceed the 10CFR100 or 10CFR50 App.A GDC 19 dose limits) on the faulted steam generator starting at the beginning of the accident.
Technical Justification: 150 gpd/steam generator and the 10 gpm (or other value based on this calculation) are for potential future Technical Specification changes. These values are assumed values for this analysis.
2. The maximum letdown of 120 gpm (ref.16) is used.
Technical Justification: This will maximize the removal rate of iodines from the primary coolant, and therefore will maximize the production rate of iodine (production = removal at steady state). See Calculation section for the formulas used. Note, this value is used for calculation of iodine production/removal rates. The letdown is assumed to be isolated at the beginning of the accident to maximize the reactor coolant inventories.
3. The primary to secondary side leak rates are based on Standard Temperature and Pressure (STP).
Technical Justification: None needed.
4. It is assumed that the faulted steam generator dries out at the start of the accident, resulting in an iodine partition factor of 1.0 per ref.10.
Technical Justification: Following an accident, the Main Steam Line will be isolated and the Main and Auxiliary Feedwater will also be isolated. Since the worst case accident occurs with the line associated with a Steam Generator with Technical Specification leakage, that Steam Generator will dry out. In reality, this dry out will not occur until all feedwater has been isolated, and then water boiled off. Assuming dry out conditions at time zero is clearly conservative.
5. In the intact steam generators, the iodine partition factor is assumed to be 100.
Technical Justification: The mass of primary to secondary leakage which occurs to the intact steam generators is small relative to the mass of secondary coolant. Therefore none of this leakage is assumed to flash and the release to the environment is through the steaming process. Reference 10 allows a partition factor of 100 for such cases.
6. In one case, a preaccident iodine spike of 60 μCi/gm I-131 equivalent (or 21 μCi/gm for the reduced concentration case) is assumed at the start of the accident. In the other case, an accident initiated iodine spike of 500 increase in the iodine release rate from the fuel is assumed in the accident initiated case with the reactor coolant starting at 1 μCi/gm I-131 equivalent (or 0.35 μCi/gm for the reduced concentration case).
Technical Justification: SRP 15.1.5 subsection 4a specifies the maximum allowable preaccident spike is required (60 uCi/gm is permissible for 48 hours). SRP 15.1.5 subsection 4b specifies that following an accident, the iodine release rate from the fuel to the reactor coolant is increased by a factor of 500.
7. The letdown demineralizer efficiency is assumed to be 1 for iodines.
Technical Justification: This will maximize iodine removal (=production) rate, and therefore result in larger iodine spiking.
8. The control room isolates in 14 seconds (ref.9) due to high radiation in the Control Building Ventilation intake (400 cpm. ref.18). This will result in an unfiltered puff into the control room for that 14 seconds. This effect is not taken into account in the analysis because of the following:
Technical Justification: a) The main leak of 10 gpm does not occur until the faulted loop steam generator is depressurized. From ref.19, the steam generator will not significantly depressurize (less than 25-50% of peak pressure) until after 14 seconds. Therefore, the release in 14 seconds will only be the secondary side activity, which is insignificant compared to the reactor coolant releases later in the accident.
b) If the 10 gpm leak were to occur at time=0, then the release in 14 seconds will be less than 0.04% of the total release (see table below). Taking into account control room filter efficiency of 95%, the 14 second release is at most 0.75% of the total contributor to the total dose. These values are based on examination of the computer runs for 10 gpm leak with preaccident spike of 60 μCi/gm I-131 equivalent and the accident initiated iodine spike of 500 times I release rate from the fuel.

	14 sec I release [Ci]	2 hr release	2-8 hr release	total release	14 sec fraction of total
preaccident I spike, 60 μCi/gm	0.32838	326.06	549.1	875.16	3.75E-04
accident I spike, 1μCi/gm*500	0.30359	362.68	4282.6	4645.28	6.54E-05
with filters taken into account:					
preaccident I spike, 60 μCi/gm	0.32838	16.303	27.455	43.758	7.50E-03
accident I spike, 1μCi/gm*500	0.30359	18.134	214.13	232.264	1.31E-03

Special Requirements/Limiting Conditions

There are no special requirements or limiting conditions in this calculation.

Calculations

The STP models consist of a pre-accident iodine spike (see figure 1) model and an accident initiated iodine spike model (see figure 2). The model(s) consist of the following:

Volumes:

#1: Reactor Coolant: 5.4E5 lb (ref.2) = 2.4494E8 gm
#2: Steam Generator w/Leak: 4.745E7 gm (ref.6)
#3: Steam Generators w/out Leak: 1.424E8 (ref.6)
#4: Environment: 1 gm (arbitrary) (This volume is made into an accumulator through the "A" card to suppress radioactive decay)

Step Sources:

The following equation is used to set up the initial activities (in Ci) for each component using the initial ANSI/ANS-18.1-1984 source (which is in units of μCi/gm):

S = Component Volume [gm] * 1E-6 Ci/μCi * I-131 equivalent conversion factor

To obtain the I-131 equivalent conversion factor, the ANSI/ANS-18.1-1984 spectrum must be converted to I-131 equivalence. From WBNTSR-008, the I-131 equivalence is:

I-131 equivalent = dose conversion factor (D/A) * concentration / I-131 dose conversion factor

	D/A mrads/Ci (ref.9)	μCi/gm coolant ANS-18.1 (ref.2)	I-131 equivalent μCi/gm	1 μCi/gm I-131 equivalent
1-131	1.48E+09	4.77E-02	4.77E-02	3.80E-01
1-132	5.35E+07	2.25E-01	8.13E-03	6.48E-02
1-133	4.00E+08	1.49E-01	4.03E-02	3.21E-01
1-134	2.50E+07	3.64E-01	6.15E-03	4.90E-02
1-135	1.24E+08	2.78E-01	2.33E-02	1.86E-01
total		1.06E+00	1.255E-01	1.000E+00
		inverse	7.965E+00	

To convert to I-131 equivalence, the reactor coolant I-131 equivalent conversion factor is (1/0.1255) = 7.965gm/μCi

For the secondary side concentrations from WBNNAL3-003, the same procedure is performed to determine the I-131 equivalence:

	D/A mrads/Ci	μCi/gm secondary side, water ansi 18.1	I-131 equivalent μCi/gm
1-131	1.48E+09	1.41E-06	1.41E-06
1-132	5.35E+07	3.37E-06	1.22E-07
1-133	4.00E+08	4.03E-06	1.09E-06
1-134	2.50E+07	2.93E-06	4.95E-08
1-135	1.24E+08	6.19E-06	5.19E-07
total		1.79E-05	3.189E-06
		inverse	3.136E+05

To convert to I-131 equivalence, the secondary side I-131 equivalent conversion factor is (1/3.189E-6) = 3.136E5 gm/μCi

The step sources to initialize the reactor coolant and the secondary side activities are:

Pre-accident iodine spike case (initial concentration = 60 μCi/gm):

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *60 = 1.171E5 (iodines)

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *1 = 1.951E3 (noble gasses)

Pre-accident iodine spike case (initial concentration = 21 μCi/gm):

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *21 = 4.097E4 (iodines)

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *0.35 = 6.828E2 (noble gasses)

Accident initiated iodine spike case (initial concentration = 1 μCi/gm):

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *1 = 1.951E3

Accident initiated iodine spike case (initial concentration = 0.35 μCi/gm):

S=2.4494E8 gm*1E-6 Ci/μCi * 7.965 *0.35 = 6.828E2

Secondary side, all cases, steam generator with leak (initial concentration = 0.1 μCi/gm):

S = 4.745E7 gm * 1E-6 Ci/μCi * 3.136E5 * 0.1 = 1.488E6

Secondary side, all cases, steam generators without leak (initial concentration = 0.1 μCi/gm):

S = 1.424E8 gm * 1E-6 Ci/μCi * 3.136E5 * 0.1 = 4.466E6

Continuous Sources:

For the accident initiated iodine spike case, the iodine spike is 500 times the iodine release rate from the fuel. At steady state, the iodine release (production) rate is equal to the removal rate. The iodine removal is due to a) radioactive decay, b) removal by the letdown system, and c) removal through leakage to the secondary side. These terms are expressed as:

$P = \Sigma$removal rates = decay + letdown + leakage

or $P = \lambda + f_L\varepsilon/V + p_s/V$

where P = production rate [hr^{-1}]

λ = decay constant for the isotope in question [hr^{-1}]

f_L = letdown flow rate = 120 gpm

ε = letdown demineralizer efficiency = 1 (assumed so as to maximize removal/production rate)

V = volume of primary coolant = 11375 cuft (ref.17) * 7.4805 gal/cuft = 8.51E4 gal

p_s = removal rate of iodine from primary side due to primary to secondary leakage = 4*150gpd = 600 gpd

and $\lambda = \ln(2)/T_{1/2}$

where $T_{1/2}$ = halflife taken from ref.15

	Half Life	λ [1/hr]	$f_L\varepsilon/V$ [1/hr]	p_s/V [1/hr]	prod rate P [1/hr]
I-131	8.04 d	3.59E-03	8.46E-02	2.94E-04	0.0885
I-132	2.28 h	3.04E-01	8.46E-02	2.94E-04	0.3889
I-133	20.9 h	3.32E-02	8.46E-02	2.94E-04	0.1181
I-134	52.6 m	7.91E-01	8.46E-02	2.94E-04	0.8756
I-135	6.61 h	1.05E-01	8.46E-02	2.94E-04	0.1898

The accident initiated iodine spike of 500 times the increase in the iodine release (production) rate from the fuel is modeled as a continuous source:

C = Volume * 1E-6 Ci/μCi * Prod. Rate * 500 * 1 μCi/gm I-131 equivalent conversion factor

where Volume = 2.4494E8 gm

Prod Rate = see table above

1 μCi/gm I-131 equivalent conversion factor = 7.965 (value determined above, this is to get the ANSI/ANS-18.1-1984 source into 1 μCi/gm I-131 equivalent

Continuous Source [1/hr] for Accident Initiated Iodine Spike:

	1 μCi/gm I-131 case	0.35 μCi/gm I-131 case
I-131	8.632E+04	3.021E+04
I-132	3.794E+05	1.328E+05
I-133	1.152E+05	4.031E+04
I-134	8.541E+05	2.989E+05
I-135	1.851E+05	6.479E+04

Flow Rates:

The following flow rates/leakage rates for each component are:

Flow from Reactor Coolant #1 to Steam Generator w/ Leak #2 all classes:

F = 10 gpm * 60 min/hr * 3785.48 cc/gal = 2.271E6 g/hr

For maximum 10CFR100/10CFR50 App.A GDC19 cases, it was determined (through iteration of various leakage rates) that the maximum leakage rates are 15.18 gpm with the reactor coolant at 1 μCi/gm I-131 equivalent and 45.70 gpm at 0.35 μCi/gm I-131 equivalent, which translate to the following:

Flow from Reactor Coolant #1 to Steam Generator w/ Leak #2 all classes (15.18 gpm case):

F = 15.18 gpm * 60 min/hr * 3785.48 cc/gal = 3.448E6 g/hr

Flow from Reactor Coolant #1 to Steam Generator w/ Leak #2 all classes (45.70 gpm case):

F = 45.70 gpm * 60 min/hr * 3785.48 cc/gal = 1.038E7 g/hr

Flow from Reactor Coolant #1 to Steam Generator w/ no Leak #3 all classes:

F = 3 steam generators * 150 gpd * 3785.48 cc/gal / 24 hr/day = 7.098E4 g/hr

From reference 8, the steam released for a 1 gpm primary to secondary leak case from the defective steam generator is 150,000 lb (0-30 min), and 1000 lb (30 min - 8 hr), and from the non-defective steam generators is 480,000 lb (0-2 hr), and 871,000 lb (2-8 hr). From reference 11 and 12, the steam released for a 10 gpm primary to secondary leak case is an additional 2509 lb (at STP) in the 0-30 min time period, and a total of 42290 lb (at STP) from 30 min to 8 hr. To take into account uncovery of the faulted steam generator, there is no iodine partitioning in the release to the environment (iodine partition coefficient = 1). However, the reactor coolant release to the unfaulted steam generator is small relative to the secondary side mass, therefore partitioning is allowed. The iodine partition coefficient due to steaming for the unfaulted steam generators to the environment is 100. These mass releases translate into the following flows:

Flow from Steam Generator w/ Leak #2 to Environment #4 (0-30 min):

F = (150,000 lb+2509 lb)(453.59 g/lb)/(0.5 hr) = 1.3835E8 g/hr (all isotopes)

Flow from Steam Generator w/ Leak #2 to Environment #4 (30 in-8 hr):

F = (42290 lb)(453.59 g/lb)/(7.5 hr) = 2.5576E6 g/hr (all isotopes)

Flow from Steam Generators w/out Leak #3 to Environment #4:

F = (480,000 lb)(453.59 g/lb)/(2 hr) = 1.0886E8 g/hr (0-2 hr, noble gasses)

F = (480,000 lb)(453.59 g/lb)/(100*2 hr) = 1.0886E6 g/hr (0-2 hr, iodines)

F = (871,000 lb)(453.59 g/lb)/(6 hr) = 6.5846E7 g/hr (2-8 hr) (noble gasses)

F = (871,000 lb)(453.59 g/lb)/(100*6 hr) = 6.5846E5 g/hr (2-8 hr) (iodines)

The following table presents the variables that change for each case:

Case	Step Source S	Flow Rate F #1 to #2	Continuous Source C	Description
A	1.171E+05 I 1.951E+03 NG	2.271E+06	Not Applicable	10 gpm leak, 60 uCi/gm I-131
B	1.951E+03	2.271E+06	see Table above	10 gpm leak, 1 uCi/gm I-131, 500 I spiking
C	1.171E+05 I 1.951E+03 NG	3.448E+06	Not Applicable	15.18 gpm leak, 60 uCi/gm I-131
D	1.951E+03	3.448E+06	see Table above	15.18 gpm leak, 1 uCi/gm I-131, 500 spiking
E	4.097E+04 I 6.828E+02 NG	2.271E+06	Not Applicable	10 gpm leak, 21 uCi/gm I-131
F	6.828E+02	2.271E+06	see Table above	10 gpm leak, 0.35 uCi/gm I-131, 500 I spiking
G	4.097E+04 I 6.828E+02 NG	1.038E+07	Not Applicable	45.70 gpm leak, 21 uCi/gm I-131
H	6.828E+02	1.038E+07	see Table above	45.70 gpm leak, 0.35 uCi/gm I-131, 500 spiking

The STP output is used as input to COROD (which determines control room operator dose) and FENCDOSE (which determines 30-day and 2-hour LPZ offsite dose). The 2-hr EAB (Exclusion Area Boundary/Site Boundary) dose is determined by multiplying the 2 hour LPZ dose by 4.305 (ref.9).

Some pertinent information from the COROD and FENCDOSE models used (but not changed) in this analysis are (from ref.9):
30-day LPZ Offsite X/Q values [sec/cum]: 1.41E-4 0-2hr, 6.68E-5 2-8 hr, 4.59E-5 8-24 hr, 2.04E-5 1-4 day, 6.35E-6 4-30 day
2-hr EAB X/Q values: 6.07E-4
Control Room X/Q: 4.07E-3 0-2 hr, 2.47E-3 2-8 hr, 1.97E-3 8-24 hr, 1.37E-3 1-4 day, 9.84E-4 4-30 day
Control Room volume: 257198 cuft
Control Room makeup/pressurization flow: 711 cfm
Control Room recirculation flow: 3600 cfm
Control Room unfiltered intake: 51 cfm
Control Room filter efficiency: 95% first pass, 70% second pass
Control Room occupancy factors: 100% 0-24 hr, 60% 1-4 days, 40% 4-30 days
ICRP-2 dose conversion factors

Figure 1: STP Model

Pre-accident Iodine Spike

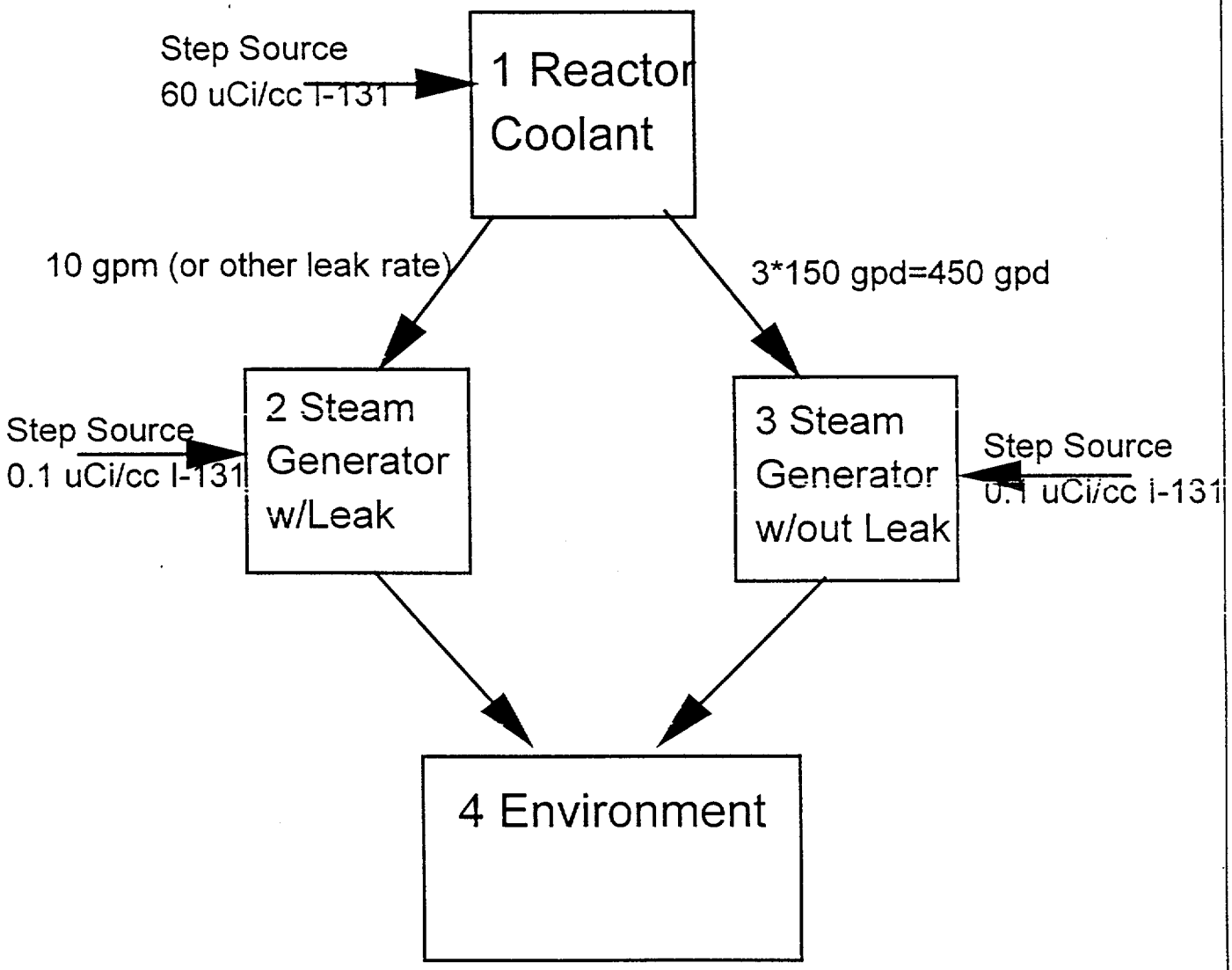
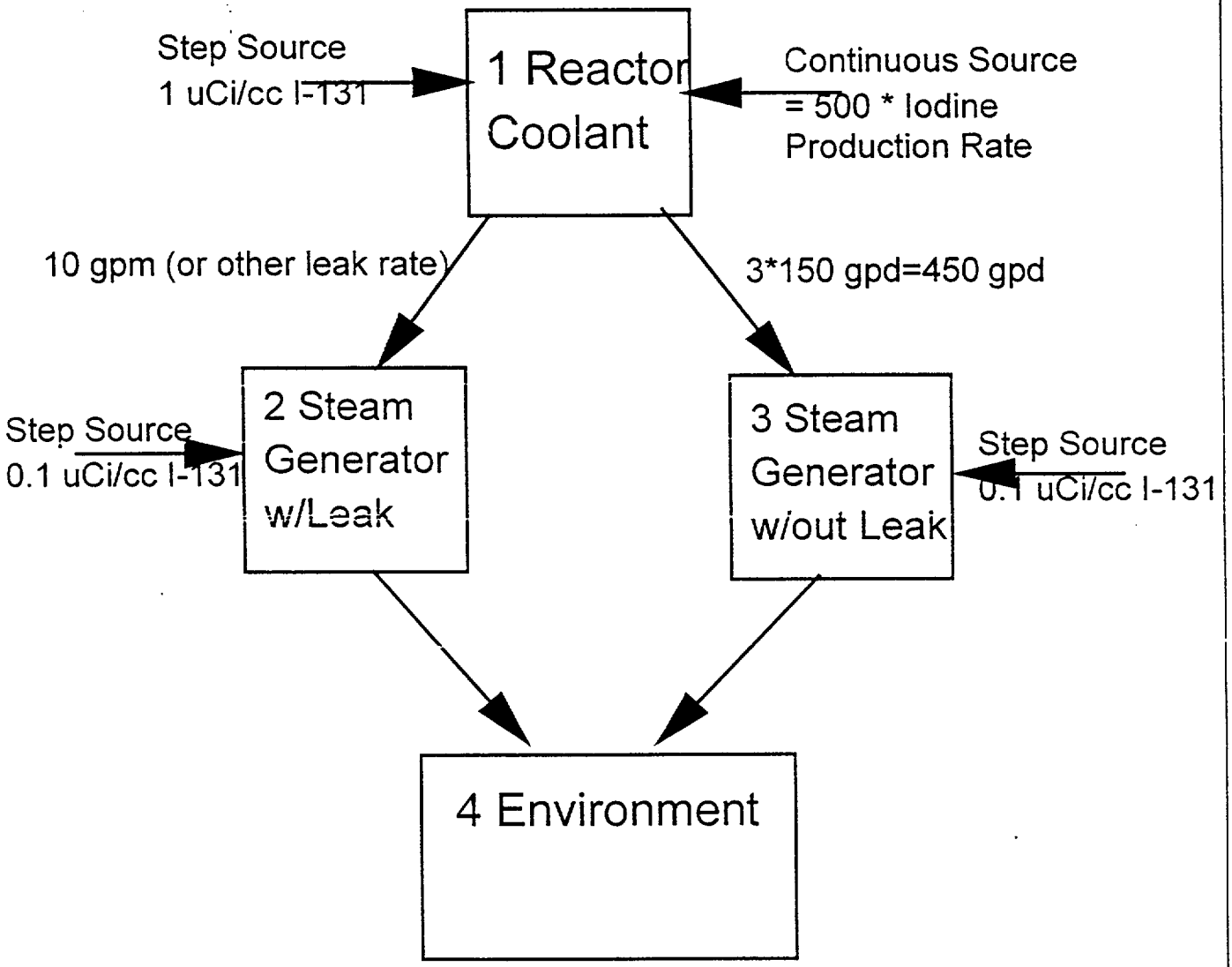


Figure 2: STP Model

Accident Initiated Iodine Spike



Results

The results for 60 μCi/gm I-131 equivalent with the pre-accident iodine spike and 1.0 μCi/gm I-131 equivalent (with 500 iodine spiking), and 150 gpd leakage/steam generator and 10 gpm (or 15.18 gpm) in the faulted loop following the accident were:

Doses due to MSLB (rem)

Pre-accident Iodine spiking, 60 μCi/gm I-131 equivalent case				Accident Initiated Iodine Spiking (500) case, Initially 1 μCi/gm I-131 equivalent			
CR	10 gpm leak	15.18 gpm leak	limit	CR	10 gpm leak	15.18 gpm leak	limit
gamma	0.004	0.005	5	gamma	0.012	0.017	5
beta	0.040	0.057	30	beta	0.106	0.153	30
Inhalation	9.287	13.780	30	Inhalation	20.07	29.99	30
30-day LPZ				30-day LPZ			
gamma	0.033	0.050	25	gamma	0.168	0.252	2.5
beta	0.009	0.014	300	beta	0.041	0.061	30
Inhalation	7.915	11.750	300	Inhalation	17.85	26.68	30
2-hr LPZ				2-hr LPZ			
gamma	0.021	0.031	25	gamma	0.029	0.043	2.5
beta	0.005	0.008	300	beta	0.007	0.010	30
Inhalation	3.261	4.795	300	Inhalation	1.685	2.407	30
2-hr EAB				2-hr EAB			
gamma	0.090	0.134	25	gamma	0.124	0.185	2.5
beta	0.023	0.033	300	beta	0.028	0.042	30
Inhalation	14.039	20.642	300	Inhalation	7.254	10.362	30

Note: the above values are rounded off from the computer output. Calculated values (2-hr EAB) are based on number from output.

The results for 21 μCi/gm I-131 equivalent with the pre-accident iodine spike and 0.35 μCi/gm I-131 equivalent (with 500 iodine spiking), and 150 gpd leakage/steam generator and 10 gpm (or 45.70 gpm) in the faulted loop following the accident were:

Doses due to MSLB (rem)

Pre-accident Iodine spiking, 21 μCi/gm I-131equivalent case				Accident Initiated Iodine Spiking (500) case, Initially 0.35 μCi/gm I-131 equivalent			
CR	10 gpm leak	45.70 gpm leak	limit	CR	10 gpm leak	45.70 gpm leak	limit
gamma	0.001	0.005	5	gamma	0.004	0.017	5
beta	0.015	0.054	30	beta	0.039	0.146	30
Inhalation	3.469	13.73	30	Inhalation	7.241	29.99	30
30-day LPZ				30-day LPZ			
gamma	0.013	0.050	25	gamma	0.060	0.255	2.5
beta	0.004	0.014	300	beta	0.014	0.061	30
Inhalation	2.965	11.720	300	Inhalation	6.44	26.72	30
2-hr LPZ				2-hr LPZ			
gamma	0.008	0.032	25	gamma	0.011	0.045	2.5
beta	0.002	0.008	300	beta	0.003	0.010	30
Inhalation	1.324	4.988	300	Inhalation	0.773	2.497	30
2-hr EAB				2-hr EAB			
gamma	0.035	0.140	25	gamma	0.047	0.193	2.5
beta	0.009	0.035	300	beta	0.011	0.044	30
Inhalation	5.700	21.473	300	Inhalation	3.327	10.750	30

Discussion and Conclusion

The offsite doses due to a MSLB with preaccident iodine spiking has 10CFR100 limits of 25 rem gamma, 300 rem beta, and 300 rem inhalation. The offsite doses due to a MSLB with accident initiated iodine spike (factor of 500) has limits of 10% of the 10CFR100 limits or 2.5 rem gamma, 30 rem beta, and 30 rem inhalation (ref.10). The control room operator doses limits from 10CFR50 App.A GDC 19 are 5 rem gamma, 30 rem beta, and 30 rem inhalation.

With the Technical Specification limits of 1 μCi/gm I-131 equivalent steady state (and 60 μCi/gm maximum), the control room and offsite doses are less than the limits with a 10 gpm leak in the faulted line and 150 gpd in the unfaulted lines. The maximum allowable leak so as to not exceed the dose limits was determined to be 15.18 gpm in the faulted line.

With the Technical Specification limits of 0.35 μCi/gm I-131 equivalent steady state (and 21 μCi/gm maximum), the control room and offsite doses are less than the limits with a 10 gpm leak in the faulted line and 150 gpd in the unfaulted lines. The maximum allowable leak so as to not exceed the dose limits was determined to be 45.70 gpm in the faulted line.

References

1. WBN Technical Specification 3.4.16, Amendment 11
2. WBNNAL3-003 R1 "Reactor Coolant Activities in Accordance with ANSI/ANS-18.1-1984" RIMS# B26 930210 254
3. Computer Code STP R6, code I.D. 262165, controlled user's manual # 2
4. Computer Code FENCDOSE R3,code I.D. 262358, controlled user's manual # 2
5. Computer Code COROD R3, code I.D. 262347, controlled user's manual # 1
6. WBNAPS3-053 R2 "Steam Generator Leakage Detection with the Condenser Vacuum Pump Air Exhaust Monitor (1,2-RM-90-119)"
7. WBNAPS3-043 R1 "Shielding Calculation For the Steam Generator Blowdown Demineralizer System" RIMS# B26 900620 200
8. Memorandum from J.W. Irons to W.L. Elliott, WAT-D-9489, "Verification of Data in FSAR Table 15.5-16" RIMS# T33 930927 823
9. WBNTSR-008 R6 "Control Room Operator and Offsite Dose Due to a Steam Generator Tube Rupture"
10. NUREG-0800 R2 section 15.1.5
11. WAT-D-10690, Nov.9, 1999, Memorandum from John W. Irons to J.E. Maddox "SLB Leak Rates" RIMS# B44 991109 002
12. WAT-D-10724, February 10, 2000, Memorandum from John W. Irons to J.E. Maddox "SLB Leak Rates Conversion" RJMS# T71 000217 928
13. N3-15-4002 R5 System Description For "Steam Generator Blowdown System"
14. FSAR Table 11.1-2 (note: this information is used only for comparison with reference 2, and not used as design input).
15. Lederer and Shirley, "Table of Isotopes", seventh ed.
16. N3-62-4001 R5 System Description for "Chemical and Volume Control System"
17. WBNNAL3-002 R2 "100-Day LOCA-DBA Source Terms for the EGTS and ABGTS Filters, Containment, Sump, and Shield Building Annulus" Note: this calculation is currently at R3, however the information is found in R2.
18. WBNAPS3-050 R2 "Determine the Main Control Room Intake Monitor (0-RE-90-125, -126) Setpoints and Post Accident Air Intake Concentrations"
19. WB-DC-40-70 R1 "Accident Analysis Parameter Checklist (AAPC)", Figures 4.3.2-13 and -25

Appendix A: Justification for Using ANSI/ANS-18.1-1984 Expected Coolant Spectrum

The choice of iodine spectrum is fairly important, since several isotopes have short halflives. Results may be affected when accident times are on the order of the decay of the short lived isotopes. There are several possible spectra available. The spectrum chosen for this analysis is the one that most closely resembles the actual spectrum present at WBN. From the surveillance test 1-SI-68-28 performed on 7/10/00 (see Attachment 1), the following concentrations were determined:

	µCi/gm RCS Gaseous		µCi/gram RCS Degassed
Ar-41	1.303E-02	F-18	1.179E-01
Kr-85M	1.915E-04	Na-24	9.169E-04
Kr-87	4.575E-04	Mn-56	9.313E-05
Xe-133	9.565E-04	Co-58	5.019E-04
Xe-135	1.429E-03	Nb-95	3.132E-05
Xe-135M	7.364E-04	I-131	6.070E-05
Xe-138	1.796E-03	I-132	1.459E-03
		I-133	8.208E-04
		I-134	2.694E-03
		I-135	1.608E-03
		Xe-135	8.914E-05
		Xe-135M	1.406E-02
		Cs-138	2.395E-03

Two potential spectra are from WBNNAL3-003 (Reactor Coolant Activities in Accordance with ANSI/ANS-18.1-1984) and from the FSAR Table 11.1-2 (Historical Design Activities). The iodine concentrations and relative concentrations for each spectrum are as follows:

	WBN actual µCi/gm	WBN actual relative fraction	ANS 18.1 µCi/gm	ANS 18.1 relative fraction	FSAR 11.1-2 µCi/gm	FSAR 11.1-2 relative fraction
I-131	6.070E-05	0.0091	0.0477	0.0448	2.5	0.2461
I-132	1.459E-03	0.2196	0.225	0.2115	0.9	0.0886
I-133	8.208E-04	0.1236	0.149	0.1401	4	0.3937
I-134	2.694E-03	0.4056	0.364	0.3422	0.56	0.0551
I-135	1.608E-03	0.2421	0.278	0.2614	2.2	0.2165
sum:	6.643E-03		1.0637		10.16	

As can be seen, the FSAR historical design concentrations do not reflect the actual measured concentrations. The FSAR values are weighted too strongly in favor of I-131 (24.6% of total as opposed to < 1% of the actual total). By comparison, the ANSI/ANS-18.1-1984 fractions are very close to the actual fractions. The worst fit was for I-134 which was 40.1% actual versus ANSI/ANS-18.1-1984 34.22%. The I-131 is slightly over predicted by ANS-18.1 (0.9% actual versus 4.48%), however this difference is not as large compared to the FSAR fraction. The ANSI/ANS-18.1-1984 spectrum overall fit is much better than the FSAR spectrum, therefore it can be concluded that the use of the ANSI/ANS-18.1-1984 spectrum is acceptable.

Attachment 1: Surveillance test 1-SI-68-28 performed on 7/10/00

JUL-24-2000 13:52 TVA PLANT MGRS OFC 423 365 1984 P.02/07

SURVEILLANCE TASK SHEET (SPP-8.2) PAGE 1 OF

WORK ORDER: 000679300
SI KEY: P0531
PROCEDURE#: 1-SI-68-28
TITLE: PRIMARY RADIOCHEMISTRY REQUIREMENTS
PERF SECT: CEM
TEST REASON: PERIODIC PERFORMANCE
DUE: 07/10/00
WBN EXT: 07/11/00
MAX EXT: FORM SPP-8.2-2
FREQ: W
EQ: N
ASME XI: N
APP MODE: 123
PERF MODE: 1234
SUBSCNT RVWS:
INSTRUCTIONS: Do NOT start prior to scheduled due date

N/A
AUTHORIZATION TO BEGIN: SRO — DATE — N/A TIME

7/10/00 / 0720
START DATE — TIME

7/10/00 / 1118
COMPLETION DATE — TIME

TEST PERFORMERS

NAME	SIGNATURE	INIT	SECT
Lynda Monroe	Lynda Monroe	LM	CHEM

WAS THIS A COMPLETE OR PARTIAL PERFORMANCE? (EXPLAIN "PARTIAL" IN REMARKS) COMPLETE: ✓ PARTIAL: ___

WERE ALL TECH SPEC/TECH REQ/ODCM/FIRE PROT REQ ACCEPTANCE CRITERIA SATISFIED? YES: ✓ NO: ___ N/A: ___

WERE ALL OTHER ACCEPTANCE CRITERIA SATISFIED? YES: ___ NO: ___ N/A: ✓

ALERT SCHEDULING REQUIRED? YES: ___ NO: ___ N/A: X

IF ALL TECH SPEC/TECH REQ/ODCM/FIRE PROT REQ WERE NOT SATISFIED, WAS AN LCO/TR/ODCM/OR ACTION REQD? (EXPLAIN IN REMARKS) YES: ___ NO: ___ N/A: ✓

Lynda Monroe 7/10/00
TEST DIRECTOR / LEAD PERFORMER — DATE

N/A
ACCEPTANCE CRITERIA REVIEW: SRO — DATE / TIME N/A

7/11/00
CHEM SUPV IND REVIEW — DATE

REMARKS:

COPY OF STS SENT TO SCHEDULING: BLL / 7/11/00
INITIALS — DATE

SECTION/WHEN/DUR HRS

SECTION/WHEN/DUR HRS

SECTION/WHEN/DUR HRS

SECTION/WHEN/DUR HRS

RECORDS TRANSMITTAL#:

JUL-24-2000 13:52 TVA PLANT MGRS OFC 423 365 1904 P.03/07

10-JUL-2000 09:23:05.19
TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT
****

SAMPLE TITLE : U1 - RCS - GASEOUS ACTIVTY
FILE IDENT : DKB600:[TVA.SAMPLE.CHEM.NEW]W0007108796_C401.CNF;1

SAMPLE ID : W0007108796_C401
SAMPLE TIME : 10-JUL-2000 08:53
SAMPLE TYPE : 1240 CC GAS MARI

* OPERATOR : LLMANNONE
* SAMPLE GEOMETRY : GM1K
* SHELF HEIGHT : 0
* EFFICIENCY FILE : GM1K0
* SAMPLE QUANTITY : 1.00000E+00 CC

****

ACQ DATE & TIME : 10-JUL-2000 09:12
PRESET LIVE TIME : 0 00:10:00
ELASPED REAL TIME : 0 00:10:01
ELAPSED LIVE TIME : 0 00:10:00

* DEADTIME (%) : 0.3%
* SENSITIVITY : 4.00000
* GAUSSIAN SEN : 10.00000
* NBR ITERATIONS : 10

****

DETECTOR : DET #3, GSS-3286
EFFIC CAL DATE : 29-JUL-1994 13:47
DCAL DATE & TIME : 9-JUL-2000 15:52:
KEV/CHAN : 4.99928E-01
OFFSET : -1.48334E-01 keV
Q COEFFICIENT : 3.22120E-08
PEAK START CHAN : 140

* LIBRARY : NOBLEGAS
* EFFIC CERT DATE : 29-JUL-1994 13:47
* ENERGY TOLER : 1.25
* HALF LIFE RATIO : 8.00000
* ABUNDANCE LIMIT : 80.0%
* CORRECTION FACTOR : 1.00000E+00
* PEAK END CHAN : 4096

****

ANALYSES : PEAK V16.9 NID V3.3 MINACT V2.8 WTMEAN/KEY V1.8

****

COUNTED ON : LION
COLLECTED BY :
COUNTED BY : LLMANNONE
REVIEWED BY :
COMMENTS :

****

Post-NID Peak Search Report

It	Energy	Area	Bkgnd	FWHM	Channel	Left	Pw	%Err	Fit	Nuclides
0	81.12	230	92	1.01	162.55	157	12	10.8		XE-133
0	151.37	88	53	0.97	303.07	299	8	17.7		KR-85M
0	196.38	70	80	0.88	393.10	388	10	26.7		
0	227.79	22	47	1.35	455.92	448	10	62.9		
0	249.81	549	83	0.94	499.98	494	12	5.4		XE-135
0	258.71	72	40	0.89	517.78	514	10	20.2		XE-138
0	305.21	19	34	0.76	610.77	607	8	57.2		KR-85M
0	402.80	48	34	2.39	805.96	801	12	29.3		KR-87
0	435.19	26	16	1.31	870.75	866	11	35.3		XE-138
0	511.06	390	49	2.28	1022.51	1014	18	6.6		
0	526.45	39	24	1.21	1053.28	1048	12	29.8		XE-135M
0	898.31	22	16	1.29	1796.96	1791	9	38.8		
0	1293.58	904	9	1.60	2587.40	2578	16	3.4		AR-41

JUL-24-2000 13:53 TVA PLANT MGRS OFC 423 365 1904 P.04/07

REPORT DATE : 10-JUL-2000 09:23
REQUESTOR : LLMANNONE

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT

POST NID QA ANALYSIS

TITLE : U1 - RCS - GASEOUS ACTIVTY

SAMPLE No. : W0007108796 C401
SAMPLE TYPE : 1240 CC GAS MARI
COUNT TIME : 10-JUL-2000 09:12:51
SAMPLE TIME : 10-JUL-2000 08:53:00
LIBRARY : NOBLEGAS

OPERATOR NAME : LLMANNONE
SAMPLE GEOMETRY : GM1K
SAMPLE QUANTITY : 1.00000E+00
DETECTOR : DET #3, GSS-3286

ISOTOPE	PEAK ENERGY	ENERGY DIFF (KEV)	DECAY CORR uCi/CC	COMMENTS
AR-41	1293.64	-0.06	1.303E-02	QA Results OK
KR-85M	151.18	0.19	1.915E-04	QA Results OK
KR-87	402.58	0.22	4.575E-04	QA Results OK
XE-133	81.00	0.12	9.565E-04	QA Results OK
XE-135	249.79	0.02	1.429E-03	QA Results OK
XE-135M	526.56	-0.11	7.364E-04	QA Results OK
XE-138	258.31	0.40	1.796E-03	QA Results OK
AVG ENERGY DIFF =		0.11	1.859E-02 = TOTAL GAMMA ACTIVITY	
			0.000E+00 = Total DGL Activity	
			1.859E-02 = Total Gas Activity	

UNIDENTIFIED/REJECTED PEAKS

ENERGY	NET AREA	FWHM	GAMMA/SEC	GAMMA/SEC /CC	% ERROR	FLAG	POTENTIAL ID	ACTIVITY
196.38	70.	0.88	4.728E+00	4.728E+00	26.7	R	KR-88	5.438E-04
227.79	22.	1.35	1.675E+00	1.675E+00	62.9	U	TE-132	5.163E-05
						U	CS-138	5.110E-03
						U	NP-239	4.252E-04
511.06	390.	2.28	7.016E+01	7.016E+01	6.60	U	ANNIL	0.000E+00
898.31	22.	1.29	7.079E+00	7.079E+00	38.8	U	RB-88	1.512E-03
						U	Y-88	2.049E-04

No Peaks of interest
7/10/00

JUL-24-2000 13:53 TVA PLANT MGRS OFC 423 365 1904 P.05/07

10-JUL-2000 09:37:45.71
TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT

SAMPLE TITLE : U1 - RCS - DEGASSED LIQUID ACTIVITY
FILE IDENT : DKB600:[TVA.SAMPLE.CHEM.NEW]W0007108795_C402.CNF;1

SAMPLE ID : W0007108795_C402
SAMPLE TIME : 10-JUL-2000 07:25
SAMPLE TYPE : RCS 20ML LSV

* OPERATOR : LLMANNONE
* SAMPLE GEOMETRY : LSV20
* SHELF HEIGHT : 1
* EFFICIENCY FILE : LSV201
* SAMPLE QUANTITY : 5.00000E+00 GRAMS

ACQ DATE & TIME : 10-JUL-2000 08:36
PRESET LIVE TIME : 0 01:00:00
ELASPED REAL TIME : 0 01:01:20
ELAPSED LIVE TIME : 0 01:00:00

* DEADTIME (%) : 2.2%
* SENSITIVITY : 4.00000
* GAUSSIAN SEN : 10.00000
* NBR ITERATIONS : 10

DETECTOR : DET #4, GSS-3310
EFFIC CAL DATE : 5-AUG-1994 11:11:
DCAL DATE & TIME : 9-JUL-2000 15:52:
KEV/CHAN : 5.00474E-01
OFFSET : -3.73924E-01 keV
Q COEFFICIENT : -1.14092E-07
PEAK START CHAN : 140

* LIBRARY : RCSLIQUID
* EFFIC CERT DATE : 5-AUG-1994 11:11:
* ENERGY TOLER : 1.25
* HALF LIFE RATIO : 8.00000
* ABUNDANCE LIMIT : 80.0%
* CORRECTION FACTOR : 1.00000E+00
* PEAK END CHAN : 4096

ANALYSES : PEAK V16.9 NID V3.3 MINACT V2.8 WTMEAN/KEY V1.8

COUNTED ON : LION
COLLECTED BY :
COUNTED BY : LLMANNONE
REVIEWED BY :
COMMENTS :

Post-NID Peak Search Report

It	Energy	Area	Bkgnd	FWHM	Channel	Left	Pw	%Err	Fit	Nuclides
0	135.60	697	25268	0.92	271.70	269	7	38.0		I-134
0	249.64	887	20225	0.79	499.61	498	6	25.6		XE-135
0	287.87	540	22521	0.98	576.01	574	7	46.3		I-135
										I-135
0	364.21	455	13323	1.06	728.61	726	8	44.3		I-131
0	405.43	310	6932	1.05	810.99	808	8	46.9		I-134
0	417.67	485	6009	1.20	835.46	832	8	28.1		I-135
0	462.73	545	5368	1.28	925.53	922	8	23.7		CS-138
0	511.00	838084	22601	2.65	1022.01	1014	19	0.1		F-18
0	522.65	824	999	1.38	1045.31	1043	7	7.3		I-132
1	526.58	1048	823	1.29	1053.15	1050	18	5.2	1.06E+00	I-135
										XE-135M
1	529.88	4510	1009	1.34	1059.77	1050	18	1.9		I-133

JUL-24-2000 13:53 TVA PLANT MGRS OFC 423 365 1904 P.06/07

REPORT DATE : 10-JUL-2000 09:37
REQUESTOR : LLMANNONE

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT

POST NID QA ANALYSIS

TITLE : U1 - RCS - DEGASSED LIQUID ACTIVITY

SAMPLE No. : W0007108795_C402
SAMPLE TYPE : RCS 20ML LSV
COUNT TIME : 10-JUL-2000 08:36:15
SAMPLE TIME : 10-JUL-2000 07:25:00
LIBRARY : RCSLIQUID
OPERATOR NAME : LLMANNONE
SAMPLE GEOMETRY : LSV20
SAMPLE QUANTITY : 5.00000E+00
DETECTOR : DET #4, GSS-331C

ISOTOPE	PEAK ENERGY	ENERGY DIFF (KEV)	DECAY CORR uCi/GRAM	COMMENTS
F-18	511.00	0.00	1.179E-01	QA Results OK
NA-24	1368.53	0.07	9.169E-04	QA Results OK
MN-56	1810.69	-0.63	9.313E-05	QA Results OK
CO-58	810.76	-0.01	5.019E-04	QA Results OK
NB-95	765.79	0.89	3.132E-05	QA Results OK
I-131	364.48	-0.27	6.070E-05	QA Results OK
I-132	667.69	0.03	1.459E-03	QA Results OK
I-133	529.87	0.01	8.208E-04	QA Results OK
I-134	847.03	-0.05	2.694E-03	QA Results OK
I-135	1260.41	0.03	1.608E-03	QA Results OK
XE-135	249.79	-0.16	8.914E-05	QA Results OK
XE-135M	526.56	0.02	1.406E-02	QA Results OK
CS-138	1435.86	-0.11	2.395E-03	QA Results OK
AVG ENERGY DIFF =		-0.01	1.426E-01	= TOTAL GAMMA ACTIVITY
			1.218E-01	= Total DGL Activity
			2.427E-03	= Total FP Activity
			1.512E-03	= Total AP Activity
			1.415E-02	= Total Gas Activity
			6.643E-03	= Total HFP Activity

Dose Equivalent Iodine-131 = 2.905E-04
Iodine 131/133 Ratio = 7.395E-02
Iodine 133/135 Ratio = 5.105E-01

DGA= 3.94E-3

287.87 KeV Peak was used in identifying 2 isotopes
526.58 KeV Peak was used in identifying 2 isotopes
546.88 KeV Peak was used in identifying 2 isotopes
766.68 KeV Peak was used in identifying 2 isotopes
810.75 KeV Peak was used in identifying 2 isotopes
846.97 KeV Peak was used in identifying 2 isotopes
857.15 KeV Peak was used in identifying 2 isotopes
1136.29 KeV Peak was used in identifying 2 isotopes

JUL-24-2000 13:53 TVA PLANT MGRS OFC 423 365 1904 P.07/07

REPORT DATE : 10-JUL-2000 09:37

REQUESTOR : LLMANNONE

TENNESSEE VALLEY AUTHORITY
WATTS BAR NUCLEAR PLANT

POST NID QA ANALYSIS

UNIDENTIFIED/REJECTED PEAKS

ENERGY	NET AREA	FWHM	GAMMA/SEC	GAMMA/SEC /GRAM	% ERROR	FLAG	POTENTIAL ID	ACTIVITY
681.90	334.	2.71	1.143E+01	2.285E+00	25.7	U	I-133	
684.87	217.	2.04	7.446E+00	1.489E+00	31.1	R	W-187 Found	1.448E-04
1288.56	38.	1.81	2.239E+00	4.478E-01	36.5	Cs-137		
1291.07	30.	1.03	1.812E+00	3.624E-01	41.6	R	FE-59	2.244E-05
1566.76	75.	2.12	5.204E+00	1.041E+00	25.3	U	I-135	
1835.61	31.	1.08	2.411E+00	4.822E-01	36.4	R	RB-88 Found	9.196E-05
						U	Y-88	1.312E-05

Total Unidentified/Rejected Peaks = 6
% Unidentified/Rejected Peaks = 10.17

Flags: U - Unknown Line
R - Rejected During Analysis
P - Positively Identified (line not in analysis library)

No Peaks of interest
7/10/00

TOTAL P.07

ENCLOSURE 4

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
COMMITMENT LIST

ENCLOSURE 4

WATTS BAR NUCLEAR PLANT (WBN) UNIT 1
AXIAL OUTSIDE DIAMETER STRESS CORROSION CRACKING (ODSCC)
COMMITMENT LIST

The hot leg dented tube support intersections with bobbin dent voltage greater than or equal to 2 volts will be inspected with a technique qualified to Appendix H of the EPRI Pressurized Water Reactor (PWR) Steam Generator Examination Guideline.