Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

July 11, 2003
NOC-AE-03001557
10CFR50

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

South Texas Project
Units 1 & 2
Docket Nos. STN 50-498, STN 50-499
Additional Information Regarding STP's Commitment to Investigate and Repair
Bottom Mounted Instrumentation Penetration Indications

Reference: Letter dated April 24, 2003 from J. J. Sheppard, STPNOC, to NRC Document Control Desk, "Commitment to Investigate and Repair Bottom Mounted Instrumentation Penetration Indications" (NOC-AE-03001521)

On April 12, 2003, STP Unit 1 was in cold shutdown and making preparations to restart from its eleventh refueling outage (1RE11). While performing a routine inspection of the reactor coolant system (RCS) and associated systems as part of the STP boric acid control program, a system engineer found what appeared to be very small amounts of boric acid residue on two bottom mounted instrumentation (BMI) penetrations on the reactor pressure vessel. STPNOC immediately began an aggressive effort to determine the origin of the residue, including radio-isotopic and chemical analyses using both on-site and off-site resources. STPNOC apprised the NRC Resident Inspector of the condition. On April 13, 2003, STPNOC formally notified the NRC in accordance with 10CFR50.72, of the potential for RCS pressure boundary leakage.

Non-destructive examination (NDE) subsequently confirmed the presence of a leak path in each of the affected penetrations.

Following the discovery of the condition, STPNOC aggressively investigated the condition, identified likely causes, and implemented effective corrective action. STPNOC enlisted the support of a number of industry experts in conducting the investigation. A thorough non-destructive examination (NDE) campaign was conducted to determine the extent of the condition, including NDE techniques not previously used in the United States. STPNOC repaired the affected penetrations with a proven technology designed and installed by a company with extensive experience in the repair method. The NRC, local officials, and the public have been kept informed as the STPNOC effort progressed. Based on the assessment summarized in this letter and its attachment, STPNOC is now preparing to commence restart of Unit 1.

A001

In the referenced correspondence STPNOC committed to a response to this condition that would include the following elements: 1) investigation of the root cause, 2) determination of the extent of the condition, 3) identification and implementation of effective corrective actions, and 4) briefing the NRC prior to restarting the unit. The discussion below outlines how STPNOC has addressed each of these elements. The attachment provides additional detail regarding the actions STPNOC has taken to resolve the condition and the bases for the conclusions described in this letter.

1. <u>Investigation of root cause</u>: STPNOC has completed the investigation. STPNOC utilized the services of a number of industry experts in addition to its own highly qualified and experienced staff to determine the likely causes. STPNOC investigated the condition using several non-destructive examination techniques, including ultrasonic testing (UT), eddy current testing (ET), and enhanced visual examination. Metallurgical analyses of boat samples from the affected penetrations and the lower half of the penetrations that were removed for the half-nozzle repair are being conducted to provide additional insight on the origin and propagation of the cracks.

   All of the Unit 1 BMI penetrations were examined. Five cracks were found in the two affected penetrations, with one axial crack in each penetration providing a leak path. No cracks were found in the 56 other BMI penetrations.

   Based on the results of inspections and examinations that have been completed, STPNOC has concluded that the cracks that resulted in the leak paths most likely resulted from manufacturing (welding) flaws resulting in excessive stress in the nozzle/weld material leading to crack initiation with low cycle fatigue/primary water stress corrosion cracking supporting crack propagation. A final root cause determination will be completed following receipt of the results of the metallurgical analysis.

2. <u>Determination of extent of condition</u>: STPNOC found cracks only in the two leaking BMI penetrations. The nature of the condition suggests that it is not likely to be found in other reactor coolant pressure boundary applications. UT examinations confirmed the visual inspection conclusions that the condition has not caused wastage of the reactor vessel. Consequently, STPNOC determined that repair of only the two affected BMI penetrations would be necessary to prepare the unit for restart.

   Bare metal visual inspections of the STP Unit 2 BMI penetrations were conducted on three occasions during the Unit 2 Fall 2002 refueling outage and its subsequent turbine repair forced outage. There was no evidence of leakage. Due to the slow growth and low safety significance of the cracks in the affected Unit 1 penetrations, the fact that the other 56 penetrations on Unit 1 showed no indications, and the absence of leakage in Unit 2, STPNOC concluded that no immediate action is required for Unit 2. As a conservative measure, STPNOC plans to perform non-destructive examination of the Unit 2 BMI penetrations in the next refueling outage where the core barrel is planned to be removed (currently 2RE11 scheduled for 2005). STPNOC will also visually inspect the Unit 2 BMI penetrations during non-refueling shutdowns that meet the duration criteria of the STP Boric Acid Leaks Program.

In addition, STPNOC evaluated the safety significance of the BMI degradation and the potential failure modes and consequences. Large margins of safety assure that detectable minor leakage would occur long before a nozzle would lose integrity. Consequently, the risk of a loss of coolant accident at the BMI nozzles is not significant. The same margins also justify a conclusion that the potential for loose parts in the reactor vessel is not significant. No wastage of the reactor vessel was found and leakage from the penetrations were detected long before wastage would be significant.

Based on this review, STPNOC concluded there was no credible mechanism that would result in a loss of coolant accident, loose parts in the reactor vessel, or excessive wastage of the reactor vessel. Consequently, the condition is not safety-significant.

3. Identification and implementation of effective corrective action: STPNOC repaired the leaking penetrations with a "half-nozzle" design that relocated the pressure boundary to the outside of the reactor vessel. The half-nozzle repair method has been successfully applied to reactor coolant pressure boundary penetrations at other nuclear facilities. The repair is in accordance with the ASME Code as required by 10CFR50.55a and is designed for the life of the plant.

   STPNOC evaluated the repair configuration considering the likely causes and other potential causes and determined that the relocation of the pressure boundary effectively isolates the leak path and the source of the leak. Based on the cause evaluation, STPNOC does not believe there is a potential for a cause that would challenge the effectiveness of the repair.

   STPNOC will take the following action to address the long-term management of potential BMI degradation:

   - Continue long-term monitoring of BMI penetrations by performing the System Pressure Testing program that requires a Reactor Coolant System leakage pressure test at the end of each refueling outage, and the Boric Acid Leaks Program that requires regular inspections of the RPV. These inspection programs have proven effective in the early detection and location of BMI nozzle leaks. No new actions or commitments are required to assure these continued actions.
   - Perform volumetric and enhanced visual examinations of the Unit 1 penetrations at the next in-service inspection of the Unit 1 reactor pressure vessel.
   - Perform ultrasonic examinations of the reactor pressure vessel base metal around the repaired penetrations at future selected Unit 1 refueling outages to confirm there are no indications of pressure vessel wastage from RCS water in the gap area of the repaired penetrations. The details of these examinations will be described in ASME Code relief requests for the repairs.
   - Examine Unit 2 penetrations with a volumetric NDE method at the next Unit 2 refueling outage where the core barrel is planned to be removed (currently 2RE11 scheduled for 2005).

4. Briefing the NRC: STPNOC met with the NRC on May 1, 2003 to discuss plans for examination, analysis and repair. STPNOC met with the NRC again on June 5, 2003 to discuss the results of the examination and repair plans. Another meeting has been scheduled for July 17, 2003 in Rockville, to review the results of the investigation of the root cause, evaluation of extent of condition and corrective action prior to restarting the unit.

In addition to the meetings described above, STPNOC and NRC headquarters and Region IV personnel have conducted teleconferences at least weekly, and there were daily interactions with the NRC's Special Inspection Team.

With the completion of the July 17 meeting described above, STPNOC will have accomplished the actions set forth in our April 24, 2003 letter.

Conclusion:

Based on the repairs performed on the affected BMI penetrations and the plans for monitoring the BMI penetrations for degradation, STPNOC is confident that restart and operation of STP Unit 1 can be conducted safely. With completion of the repair in accordance with the ASME Code requirements, as modified by NRC-approved ASME Code relief requests submitted in accordance with 10CFR50.55a, STP Unit 1 is in compliance with its Technical Specification requirements for operability.

STPNOC will not commence heat-up of STP Unit 1 to Mode 4 until it receives written confirmation from the NRC that all necessary NRC actions are complete.

Please call me at (361) 972-8757 or Gary Parkey at (361) 972-7800 if you have any questions.

JJ Sheppard

J. J. Sheppard
President and
Chief Executive Officer

Awh

Attachment

cc:

(paper copy)

Ellis W. Merschoff
Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, Texas 76011-8064

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Richard A. Ratliff
Bureau of Radiation Control
Texas Department of Health
1100 West 49th Street
Austin, TX 78756-3189

Cornelius F. O'Keefe
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

C. M. Canady
City of Austin
Electric Utility Department
721 Barton Springs Road
Austin, TX 78704

(electronic copy)

A. H. Gutterman, Esquire
Morgan, Lewis & Bockius LLP

L. D. Blaylock
City Public Service

Mohan C. Thadani
U. S. Nuclear Regulatory Commission

R. L. Balcom
Texas Genco, LP

A. Ramirez
City of Austin

C. A. Johnson
AEP Texas Central Company

Jon C. Wood
Matthews & Branscomb

South Texas Project Status Report on Resolving the Unit 1 BMI Issue

I. Table of Contents

I. TABLE OF CONTENTS I

II. EXECUTIVE SUMMARY 1

III. CONDITION IDENTIFIED 2

IV. CONDITION SIGNIFICANCE 2

V. INVESTIGATION AND EVALUATION 3

A. APPARENT CAUSES 3

B. PROBLEM IDENTIFICATION AND INITIAL EVALUATIONS 4

C. ESTABLISHMENT OF INVESTIGATION STRATEGY 7

D. EXAMINATION CAMPAIGN 8

E. FAILURE MODES AND EFFECTS ANALYSIS (FMEA) 11

1. FMEA - DEVELOPMENT 11

2. FMEA – REVIEW OF FABRICATION AND OPERATIONAL RECORD AND RESULTS OF ANALYTICAL EVALUATIONS 12

3. FMEA – UNRESOLVED INITIATORS 13

4. FMEA - EVALUATION OF POTENTIAL CONSEQUENCES OF UNIDENTIFIED LEAKAGE 13

5. CURRENT STATUS OF EXAMINATIONS AND EVALUATIONS 16

F. FAULT TREE ANALYSIS (FTA) AND CONCLUSIONS 17

G. RESULTS OF FMEA AND FTA 17

VI. INVESTIGATION SUMMARY 18

VII. REPAIR 19

VIII. STP GENERIC IMPLICATIONS 21

IX. CORRECTIVE ACTIONS 22

X. ATTACHMENTS 23

South Texas Project Status Report on Resolving the Unit 1 BMI Issue

II. Executive Summary

On April 12, 2003, with South Texas Project Unit 1 in a refueling outage, personnel discovered small deposits at two Bottom Mounted Instrument nozzles of the reactor vessel. These deposits were confirmed to be residue from very small reactor coolant leaks. Utilizing isotopic aging techniques, South Texas Project estimated the age of the leaks at time of identification as at least three to five years old.

The South Texas Project defined and implemented an extensive program of analysis, non-destructive inspections, and review of design, fabrication and operational records to understand the extent of the condition and its potential causes. The non-destructive examinations identified cracks in only two nozzles. These were the same two nozzles identified by the visible residue, penetration nozzles number 1 and 46. No other cracks or flaws were identified in the other 56 Bottom Mounted Instrument nozzles. A total of five cracks were found in the two nozzles.

Nozzle 1 had three cracks and each crack was axially oriented to the J-groove weld. One crack extended above and below the J-groove weld and penetrated the inside diameter of the nozzle. The second and third cracks were much smaller and were located on the outside diameter of the nozzle close to the initial root weld passes. The second and third cracks did not extend either above or below the J-groove weld and did not penetrate the inside diameter of the nozzle.

Nozzle 46 had two cracks and each crack was axially oriented to the J-groove weld. One crack extended above and below the J-groove weld but did not penetrate the inside diameter of the nozzle. The second crack extended for most of the length of the J-groove weld from the outside diameter, did not extend either above or below the J-groove weld, and did not penetrate through to the inside diameter of the nozzle.

South Texas Project performed half-nozzle repairs on both Bottom Mounted Instrument nozzles number 1 and 46, using an Alloy 690 material with demonstrated superior corrosion resistance compared to the original Alloy 600 nozzle. The repair relocated the pressure boundary to the outside of the reactor pressure vessel surface where future visual and non-destructive examinations can be readily performed. The upper section of the original Alloy 600 nozzle remains in service as a non-pressure boundary component.

South Texas Project has ongoing activities to perform metallurgical analysis of the nozzle portions removed by the half-nozzle repair, as well as metallurgical analysis of boat samples captured from penetration nozzles number 1 and 46 to further characterize the cracks identified by the non-destructive examination program.

At this point in the investigation South Texas Project has identified the apparent cause of the cracks in the two nozzles as manufacturing (welding) flaws resulting in excessive stress in the nozzle/weld material leading to crack initiation with low

cycle fatigue/primary water stress corrosion cracking then supporting crack propagation.

Several noteworthy observations can be made from this problem thus far:

- South Texas Project's RCS Pressure Boundary Inspection for Boric Acid Leaks Program identified these leaks at an early stage.
- These leaks were very small.
- Leakage paths for these leaks were identified.
- No other cracks of any kind were found in the other 56 Bottom Mounted Instrument Nozzles.
- No circumferential cracking was identified.

If these cracks were the product of a truly random time dependent mechanism such as primary water stress corrosion cracking, then additional cracks would have been expected in a number of other nozzles.

The South Texas Project concludes that these cracks in Bottom Mounted Instrument nozzles 1 and 46 resulted from isolated circumstances applicable to these two nozzles. The half-nozzle repair that was selected moves the pressure boundary from the existing J-groove weld inside of the reactor pressure vessel to a new temperbead pad and J-groove weld on the outside of the reactor pressure vessel for penetration nozzles number 1 and 46, and facilitates ongoing regular inspection of the repairs by existing programs and requirements.

III. Condition Identified

On April 12, 2003, with South Texas Project Unit 1 in a refueling outage, personnel discovered deposits at two Bottom Mounted Instrument (BMI) nozzles (number 1 and 46) of the reactor vessel. Analysis indicated that these deposits contained boron and elevated levels of lithium confirming these deposits were reactor coolant system (RCS) leakage.

IV. Condition Significance

Leakage at these BMI locations constitutes a failure of the RCS to satisfy Technical Specification (TS) 3.4.6.2.a requirements for no pressure boundary leakage. Since the Unit 1 leak indications were discovered during a refueling outage and did not require additional RCS inventory control actions or a plant shutdown evolution, there was no actual risk increase associated with this condition.

Due to the indication of leakage at BMI penetrations 1 and 46, Technical Specification 3.4.10 for reactor coolant system structural integrity was not met.

Action a. was implemented limiting RCS temperature to 130 degrees (based on minimum temperature from the curve in TS Fig. 3.4-2).

Although the cracks in Penetrations 1 and 46 grew to a sufficient extent to create a leakage path, the cracks were not structurally significant. The design of the BMI nozzle penetrations provides large structural design margins. Through-wall circumferential cracks around greater than 80% of the circumference of the nozzle would be required to reduce this design margin to a safety factor of 3. Similarly, a through-wall axial crack would have to exceed 5.4 inches in length before reducing the design margin to a safety factor of 3. There were no circumferential cracks, and the largest axial crack was just over 1-1/4 inches in length. Therefore, the most significant consequence of the cracks in Penetrations 1 and 46 was minor leakage.

There were no adverse safety or radiological consequences associated with this condition. Other than the degradation of the two affected BMI penetrations, no equipment damage occurred as a result of this condition, and the condition did not affect the operability of any other safety-related equipment.

Large margins of safety assure that detectable minor leakage would occur long before a nozzle would lose integrity. Consequently, the risk of a LOCA at the BMI nozzles is not significant. The same margins also justify a conclusion that the potential for loose parts in the RPV is not significant.

No wastage of the RPV was found and leakage from the penetrations would be detected long before wastage would be a significant.

Based on this review, STPNOC concluded there was no credible mechanism that would result in a loss of coolant accident, loose parts in the reactor vessel, or excessive wastage of the reactor vessel. Consequently, the condition is not safety-significant.

V. Investigation and Evaluation

This section describes the investigation and evaluations performed to identify the extent and cause of the condition.

V. Investigation and Evaluation

A. Apparent Causes

The apparent cause for the identified BMI leaks is manufacturing (welding) flaws resulting in excessive stress in the nozzle/weld material leading to crack initiation with low cycle fatigue/primary water stress corrosion cracking then supporting crack propagation.

V. Investigation and Evaluation

B. Problem Identification and Initial Evaluations

The reactor vessel at STP Unit 1 is a Westinghouse design constructed by Combustion Engineering. The reactor is licensed for thermal power output of 3853 MW. The reactor pressure vessel (RPV) has an operating pressure of 2235 psig and a design pressure of 2485 psig. Unit 1 began commercial operation on August 25, 1988. The Unit 1 reactor accumulated 11.15 effective full power years (EFPY) of operation when the plant was shut down for the eleventh refueling outage, 1RE11, on March 26, 2003.

The bottom head of the reactor was inspected on April 12, 2003 as a routine part of the station's regular bare metal inspection of the reactor vessel bottom penetrations. The inspection is part of the RCS Pressure Boundary Inspection for Boric Acid Leaks program, 0PGP03-ZE-0033. The bottom head of the reactor is contained in an insulating box structure with no insulation directly in contact with the bottom head. The inspection is accomplished by removing some of the insulation panels forming the box. Three different vantage points are used to inspect all 58 BMI nozzles. The inspection found small amounts (150 milligrams on nozzle 1 and 3 milligrams on nozzle 46) of white residue around Bottom Mounted Instrument (BMI) nozzles number 1 and 46 at the annulus between the nozzle and the bottom head, (Figure 1 and Figure 2).

Figure 1 (Nozzle 46)

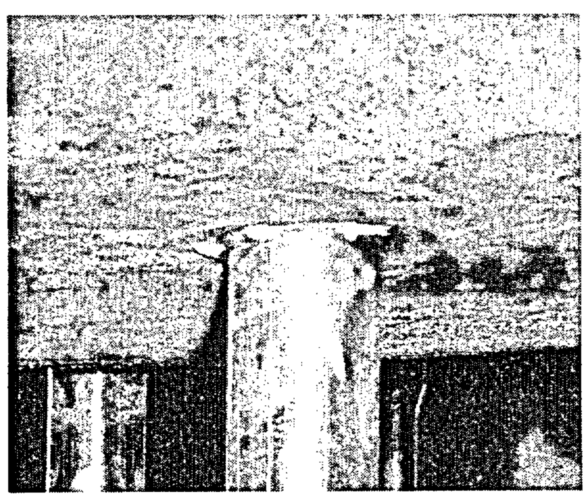
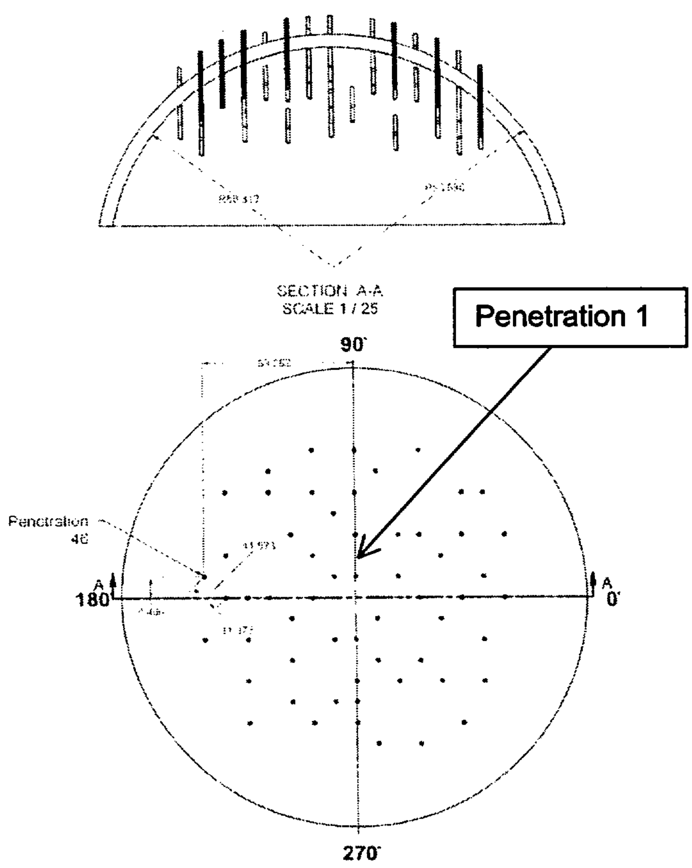


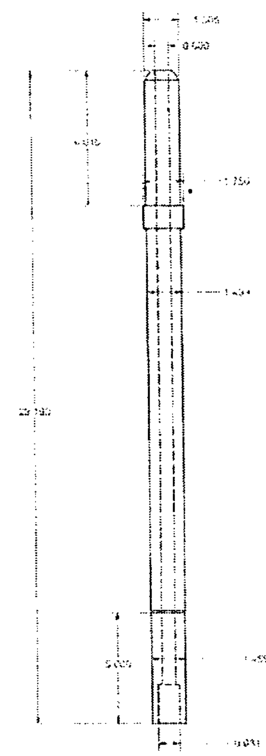
Figure 2 (Nozzle 1)

South Texas Project Status Report on Resolving the Unit 1 BMI Issue

Nozzle number 1 is located about 8.5 inches due north of the center of the bottom head. The nozzle is nearly perpendicular to the reactor bottom head. Penetration number 46 is about 8.5 inches north of the east-west center line and just over 59 inches west of the bottom head center (Figure 3). Nozzle number 46 penetrates the head at 42.6 degrees.



Viewed From Outside Head



Penetration 46
BMI Tube
Mtl. = SB-166

The BMI nozzles are machined from 1.75-inch diameter Alloy 600 bar stock. The nozzles have an outside diameter of 1.499 inches and an inside diameter of 0.60 inches. The BMI nozzles are structurally secured by an interior J-groove weld consisting of Alloy 600 base weld materials designated as 182 with fabrication

weld repairs being made with either 182 or 82 weld material. The vessel itself is 5.38-inch thick low alloy carbon steel with 0.22 inches of stainless steel cladding on the interior surface. Each penetration bore in the reactor bottom head is nominally 1.5 inches in diameter. There is an open annulus between the nozzle and the reactor head below the J-groove weld. Figure 4 provides a simplified view of a typical nozzle configuration

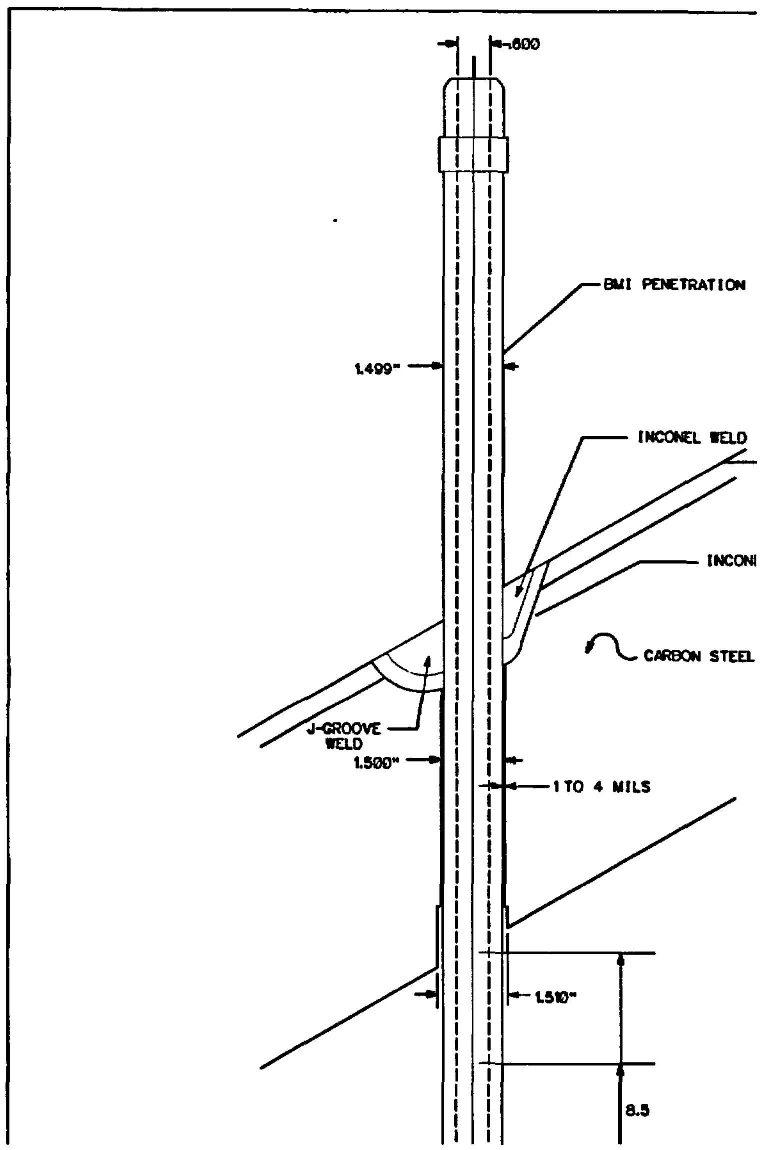


Figure 4

The presence of lithium and boron in the deposits provided the initial indication that the source of the deposits was operational reactor coolant since these elements comprise the majority of the dissolved solids resulting from evaporated reactor coolant. To eliminate the possibility that the lithium could have originated from a source other than the reactor coolant system, a lithium isotopic analysis was also performed. The analysis showed the lithium to be approximately 99.9% Lithium-7, which could only have come from the reactor coolant system.

To determine the approximate age of the residue the ratio of Cesium-134 to Cesium-137 (Cs-134/Cs-137) was calculated. Cesium-134 has a half-life of 2.06 years and Cesium-137 has a half-life of 30.17 years. The ratio of Cesium-134 to 137 in the primary cooling system is approximately 1. The Cs-134/137 ratios in the samples were 0.30 and 0.25 for penetrations 1 and 46 indicating the <u>average</u> ages of the samples are three and five years, respectively. The actual age of each leak could be considerably older depending on how the leak developed over time.

The reactor vessel bottom head is inspected every refueling outage, an approximately 18 month interval. For very small leaks, it would take time to fill enough of the annulus below the J-groove weld to become visible below the vessel. The residue was only visible during the last operating cycle.

It is important to note that the visual inspection program identified the leakage from these small cracks long before there would have been safety-significant structural degradation.

V. Investigation and Evaluation

C. Establishment of Investigation Strategy

While the BMI deposits were being analyzed, reviews of the design and fabrication records confirmed that the BMI nozzles were constructed from Alloy 600. The associated J-groove welds were made from 182 weld material with in-process weld repairs being made with either 182 or 82 weld material.

Station personnel were aware of extensive industry activity related to Alloy 600 materials used on reactor closure head penetrations and Control Rod Drive Mechanisms (CRDMs) related to Primary Water Stress Corrosion Cracking (PWSCC). Utilizing this information as a starting point, off-site experts from peer stations and supporting organizations were assembled during the week of April 13. These experts validated STP's processes confirming the residues as products of RCS leakage and assisted in developing an investigation/cause strategy.

Several of these industry representatives are members of the Electric Power Research Institute (EPRI) Material Reliability Project (MRP) or supporting contractors to the EPRI MRP efforts. These members identified that the MRP had a draft Failure Modes and Effects Analysis (FMEA) for upper head penetrations

which might be helpful and worked with STP personnel to develop a BMI FMEA. STP adapted this FMEA for bottom heads to assist in the investigation of the causes of the BMI leakage. The revised FMEA chart utilized by STP for this purpose is included as Attachment 1.

V. Investigation and Evaluation

D. Examination Campaign

An extensive nondestructive examination (NDE) program was developed and implemented in order to evaluate the condition of the Unit 1 BMI nozzles, J-groove welds, and the RPV wall and cladding immediately adjacent to BMI nozzles. The three primary NDE methods used were volumetric (UT), visual (VT), and surface (ET). In addition, both a rod test and eddy current profilometry measurements were performed in order to characterize the nozzle straightness and ovality. Finally, helium gas was used to pressurize the annulus between the lower nozzle and vessel wall at nozzles 1 and 46 to see if the small helium molecules would pass through any cracks that connected the annulus with either the nozzle ID or any points inside the vessel. During the repair process, visual and borescope examinations were conducted of the inside surface of the nozzles being repaired to check for visible cracks, and of the vessel wall in the nozzle bore holes to check for wastage.

Full-scale mock-ups were constructed with appropriate flaws incorporated to validate and calibrate the NDE tools. The mock-ups were used to evaluate prospective NDE vendors. The vendors' ability to identify and quantify the flaws built into the mock-ups was evaluated. This process provides confidence that the NDE results reported below accurately describe the condition of the Unit 1 BMI nozzles and J-groove welds.

The examinations identified cracks in only the two nozzles identified by the visual residue, penetration nozzles 1 and 46. No other cracks or flaws were identified in the other 56 Bottom Mounted Instrument nozzles. A total of five cracks were found in the two nozzles.

Nozzle 1 had three cracks. Each crack was axially oriented to the J-groove weld. One crack extended above and below the J-groove weld and penetrated the inside diameter of the nozzle. The second and third cracks were much smaller and on the outside diameter of the nozzle close to the initial weld passes. The second and third cracks did not extend above or below the J-groove weld and did not penetrate the inside diameter of the weld.

Nozzle 46 had two cracks. Each crack was axially oriented to the J-groove weld. One crack extended above and below the J-groove weld but did not penetrate the inside diameter of the nozzle. The second crack was smaller, did not extend above or below the J-groove weld, and did not penetrate the inside diameter of the weld.

Volumetric Examination: Nozzle ID and Vessel Wastage

Two methods of volumetric examination were used. The primary inspection approach was by means of Ultrasonic (UT) probe examination of all 58 BMI nozzles from the ID to locate, size and characterize any cracks in the nozzle wall. A secondary UT approach used a Phased Array inspection technique that is under industry development as a NDE technique for locating areas of vessel wastage. Development and application for each of these inspections were first time performances for the nuclear industry in the United States. Qualifying and performing underwater UT examinations of thick wall tubing products required significant coordination between STP and supporting vendors. These efforts extended into data evaluation efforts. Phased Array UT was a first time application to examine the RPV for potential material wastage.

Ultrasonic (UT): Nozzle ID

For the primary UT inspections, UT probes were inserted in each of 57 nozzles from the top down inside the vessel. Nozzle 31, which had a stuck thimble, was tested separately from the bottom after removing the thimble guide tube. Three different UT probes were used:

- Time of flight diffraction (TOFD) Axial Probe interrogation for axial oriented flaws
- TOFD Circumferential Probe interrogation for circumferential oriented flaws
- Straight beam (0-degree) examination for weld profiling and characterization of lack of fusion at the tube to weld interface.

All three probes confirmed the presence of five cracks, three in nozzle 1 and two in nozzle 46. Attachments 5 and 6 are two-dimensional representations of the cracks. There were no cracks indicated in any of the remaining nozzles, and no circumferential cracks were found. In addition, the UT probes indicated anomalous conditions at the nozzle weld interface in all of the nozzles, possibly indicating lack of weld fusion or some other anomalous weld condition. The possible significance of lack of fusion in the BMI nozzle welds is discussed later in Section V. E. 4. The extensive data derived from the primary UT inspection are electronically recorded for detailed analysis and preserved as a baseline for future UT inspections.

Phased Array UT: Reactor Pressure Vessel Wastage

A developmental Phased Array UT technique was also used to detect and characterize potential wastage of the RPV at the BMI nozzle bore region due to boric acid leakage. This Phased Array UT examination at the exit points of nozzles 1 and 46 found no indications of vessel wall wastage. Since these were the only two nozzles leaking, they were the only ones tested.

Visual Examination: Nozzle OD and Weld Surface

Thorough visual examination of the outside of the vessel bottom head found indications of leakage at only nozzles 1 and 46.

Inside the RPV enhanced visual examination by remote camera with a minimum resolution requirement to detect a 0.0005 inch diameter wire was used to examine all 58 nozzles, J-groove welds, and adjacent cladding. This examination found one suspect indication in the weld at nozzle 33. Later ET examination of this weld (below) did not indicate a crack at this location. The visual investigation did indicate grind marks on many of the nozzles. These grind marks are the result of normal fabrication processes and preparation for dye penetrant testing (PT).

Surface Examinations: Weld Surface and Nozzle ID

An ET (Array Surface Probe Coil) probe was used to examine the J-groove weld for surface or slightly subsurface flaws in the weld material around nozzles 1, 9, 12, 33, 34, 38, 41, and 46. Nozzles 1 and 46 were selected because of the identified flaws. Nozzle 33 was selected as a nozzle of interest based on the results from the visual examinations. The remaining 5 nozzles were selected as an additional sample. The ET examination found no anomalous conditions on the weld surface at any of the examined locations.

ET (Bobbin Coil) of the full length of the BMI nozzle was conducted to identify any surface or slightly subsurface flaws on the nozzle ID. Nozzles 1 and 46 were selected due to flaws found by the UT method, and nozzles 2 and 4 were selected as an additional sample. The ET examination of the nozzle ID found the through wall flaw in nozzle 1. The ID examination of nozzle 46 confirmed that its largest crack was subsurface to the ID. There were no other indications of ID anomalies.

Rod Test: Nozzle Distortion

A rod with a diameter approximately 0.040 inches less than the nozzle ID was inserted from the top of 57 of the nozzles. Nozzle 31 was tested later because of a stuck thimble. The rod passed smoothly through all the tested nozzles confirming no significant bending or distortion of the nozzle ID.

Eddy Current Profilometry: Nozzle Distortion

ET profilometry examinations were conducted from the bottom up in nozzles 1 and 46 after removing the guide tubes during the repair process. Nozzle ID measurements were made in the weld region and just over 6 inches above the top of the weld surface. The measured maximum variation in ID in nozzle 1 is about 0.016 inch at the weld and about 0.016 inch at the location 6 inches above the

weld. The measured maximum variation in ID in nozzle 46 was about 0.016 inch at the weld and just over 0.013 inch at the location 6 inches above the weld.

Helium Gas Bubble Test: Leak Path Verification

Nozzles 1 and 46 were tested to identify any leak paths from the annulus between the lower nozzle and vessel wall and the vessel interior. A gas pressure chamber was attached to the nozzle and vessel exterior wall and pressurized with helium gas. The same camera that was used for the interior-vessel visual examination was used to see if bubbles could be observed either outside the nozzle in the weld/nozzle OD region, or from inside the nozzle due to cracks through the wall in nozzle 1. The first test was at 100 psi, and at this pressure no bubbles were seen exiting at any of the nozzle locations.

The pressure was then raised to 150 psi. There were no bubbles observed at nozzle 46. However, bubbles were seen at nozzle 1 at the azimuth of the large crack in the nozzle detected by UT in the fillet region of the J-groove weld at a rate of about 1 bubble per second. The bubble test confirms the fact that the cracks believed responsible for the observed reactor coolant leakage are extremely tight.

Additional Visual Examinations

Borescopic examinations were used to look for signs of ID cracking in Nozzles 1 and 46. These examinations were performed during the repair process when the tops of the nozzles were plugged. No crack indications were identified.

Visual and borescopic examinations were also performed at the same time to inspect the condition of the vessel wall in the nozzle penetration holes for nozzles 1 and 46. After the half-nozzle sections were removed from these two locations the vessel wall was examined for signs of wastage. No indications of wastage were found.

V. Investigation and Evaluation

E. Failure Modes and Effects Analysis (FMEA)

1. FMEA - Development

STP personnel working with EPRI MRP members modified the MRP draft CRDM FMEA to reflect the BMI configuration. STP then used this modified FMEA to develop a plan for detailed and comprehensive investigation, inspection and analysis in order to determine the possible cause(s) of the identified BMI leaks.

V. Investigation and Evaluation

E. Failure Modes and Effects Analysis (FMEA)

2. FMEA – Review of Fabrication and Operational Record and Results of Analytical Evaluations

The investigation and review of fabrication and operational records, together with the results of analytical work, allow the elimination of chemistry control, nozzle fabrication defects, operational transients and flow-induced vibration as possible causes of the BMI nozzle cracks.

STP Unit 1 RCS *chemistry control* history was reviewed to identify any occurrences of extreme water chemistry transients or loss of resins into the RCS system. Such occurrences could increase Alloy 600 susceptibility to PWSCC. STP Unit 1 RCS chemistry has been consistently maintained to industry standards with infrequent occurrences of action level responses and no occurrences of extreme pH levels or lost resins within the RCS system.

Fabrication records for BMI nozzle material supply and nozzle manufacturing processes were reviewed. 100% OD surface examination of the shop fabricated BMI nozzles by UT and PT examinations prior to their installation and welding into the vessel showed no indications. No indications of off-normal processing or non-conforming conditions that might indicate a particular susceptibility to cracking were identified on the material heats or specific nozzles installed. However the heat treatment, chemical and mechanical properties characterize the nozzle Alloy 600 materials as being in the range known to be susceptible to PWSCC.

Operational transients for the Unit 1 RPV have been within the design limits. The initial hydrostatic pressure test was conducted at a pressure of 3117 psig and a temperature between 180 and 200 degrees Fahrenheit. In addition, since commencing commercial operation, the Unit 1 RPV has experienced 51 cool down and depressurization cycles from design operating temperature and pressure as part of normal operations. The RPV design assumes 200 such transients over the 40-year life of the vessel. Nonetheless, this does not preclude the possibility that a pre-existing fabrication flaw could have propagated through the pressure boundary as a result of the additional stresses associated with the initial hydrostatic pressure test or as a result of the stresses induced by normal operational transients.

Flow-induced vibration has been eliminated as a possible cause of the cracks found by NDE. Analysis indicates that RCS flow rates would induce approximately a 44 Hz excitation of the BMI nozzles. The BMI nozzles have been analyzed to have a natural frequency of approximately 237 Hz. The frequency separation between excitation and natural frequency is sufficiently large to effectively eliminate flow-induced vibration as a possible cause of the observed cracks.

Additionally, flow induced loading and vibration produce predominantly axial stresses in the BMI nozzles, which would tend to result in circumferential cracks. In contrast, the cracks identified in nozzles 1 and 46 are axially oriented, providing

further evidence that flow-induced vibration forces are not possible causes of the BMI cracks.

V. Investigation and Evaluation

E. Failure Modes and Effects Analysis (FMEA)

3. FMEA – Unresolved Initiators

The primary focus of the initial FMEA (Attachment 1) was on potential contributors to PWSCC crack initiation and propagation, which could lead to nozzle failures if the leakage were not detected and corrective action taken.

However, the results of the investigation, inspection and analysis to date have significantly reduced the scope of the initial FMEA shown in Attachment 1. Furthermore, the NDE inspections identified three cracks at the interface between the nozzle and J-groove weld that do not extend either beyond the weld or to the nozzle ID, and have likely not been in contact with the primary coolant environment. This finding suggests that PWSCC may not be the initiating mechanism for these particular cracks.

Based on the accumulated investigation, inspection and analysis results, the FMEA was streamlined (Attachment 2). This streamlined FMEA allows the evaluation of potential consequences of unidentified leakage. The potential consequences to be considered fall into three main categories: Loss-Of Coolant Accidents (LOCAs), RPV Wastage and Loose Parts.

V. Investigation and Evaluation

E. Failure Modes and Effects Analysis (FMEA)

4. FMEA - Evaluation of Potential Consequences of Unidentified Leakage

As noted above, unidentified BMI nozzle cracks could potentially lead to nozzle failure. The potential consequences of such failures fall into three major categories:

- A nozzle break or failure resulting in a loss of coolant accident (LOCA);
- A major nozzle leak resulting in significant wastage of RPV material finally resulting in a LOCA; and
- A nozzle break or failure releasing loose parts within the RCS causing significant damage.

LOCA Considerations

The investigation, inspection and analysis of the STP BMI nozzle cracks to date demonstrate that the development of a crack in either the BMI nozzle or the

associated weld large enough to result in a nozzle failure and resultant LOCA is not a credible event for the following reasons:

- The primary stress on the BMI nozzles is residual stress resulting from the J-groove welding process.
- Excess stresses are relieved by material cracking. Additionally cracks generally appear at right angles to the stress being relieved.
- The cracks identified in penetrations 1 and 46 were axial in nature appearing at right angles to the hoop stresses created by J-groove welding. No circumferential crack that could lead to nozzle failure was detected.
- No developing cracks or flaws were identified in any nozzle other than nozzles 1 and 46.
- Analysis indicates that through wall circumferential cracking of the nozzle below the weld would have to exceed 304° before exceeding a factor of 3 safety margin to nozzle failure. No circumferential cracks were identified.
- Analysis indicates that a complete lack of fusion between the BMI nozzle and the J-groove weld would have to exceed 302° before exceeding a factor of 3 safety margin to nozzle ejection. This analysis further assumes that the nozzle wall and inside weld surface are smooth and straight while in fact the fabrication methods used have likely created irregular contours defining the weld and nozzle interface effectively keying them together to provide further resistance to nozzle ejection. Anomalies were detected in the weld region of all BMI nozzles, no significant lack of fusion was identified by the volumetric (UT) examinations.
- Analysis indicates that an axial through wall crack below the J-groove weld would have to exceed 5.4 inches before exceeding a factor of 3 safety margin to nozzle integrity. This analysis does not take any credit for the support provided by the vessel structure to nozzle within the nozzle penetration annular space. Analysis also indicates that the hoop stresses below the J-groove weld region are not sufficient to cause continued axial crack growth beyond the weld region.
- The FMEA assumes that, when a leak first occurs, no action is taken to prevent a LOCA. However as evidenced both by the leaks identified in nozzles 1 and 46 at STP Unit 1 and by extensive industry experience with CRDMs and pressurizer heater penetrations, Alloy 600 cracks are typically very tight and slow to develop. These leak characteristics, together with the access configuration for bottom head inspection at South Texas ensure that early leak detection and repair would occur.

The discussion above demonstrates that a BMI leak leading to nozzle failure and a resultant LOCA is not a credible event.

Vessel Wastage Considerations

The investigation, inspection and analysis of the STP BMI nozzle cracks to date demonstrate that the development of a reactor coolant leak large enough to cause vessel wastage to the point where a nozzle failure and a resultant LOCA occur is not a credible event for the following reasons:

- The visual examinations of all BMI nozzles, J-groove welds and adjacent cladding performed during the current investigation did not identify any underlying vessel wastage.
- Surface ET inside the vessel around eight BMI nozzles did not identify any surface breaking flaws in the J-groove welds, the interfacing Alloy 600 butter zone on the outer edges of the J-groove welds or on the adjacent cladding surfaces.
- Analysis models developed for the EPRI MRP show that the reactor coolant leakage rate must be high enough to lower the vessel shell metal temperature down to about 212°F before boric acid can concentrate and allow rapid RPV steel corrosion rates to develop.
- The identified leaks in nozzles 1 and 46 were very small compared to the EPRI leak rates identified as being necessary to cause vessel wastage.
- The residue from the leaks at nozzles 1 and 46 had very little color, i.e. it was primarily white. In contrast, boric acid residue associated with carbon steel corrosion has a reddish rusty color.
- UT inspection of the reactor vessel shell around nozzles 1 and 46 using developmental UT inspection-equipment found no indication of vessel wastage.
- Visual examination of the RPV nozzle bore holes using both a high resolution video camera and a borescope during the repair of nozzles 1 and 46 did not identify any signs of wastage of the nozzle penetration bore hole steel surfaces.

The discussion above demonstrates that the development of a BMI nozzle leak large enough to cause vessel wastage to the point where a nozzle failure and a resultant LOCA occur is not a credible event.

Loose Parts Considerations

The investigation, inspection and analysis of the STP BMI nozzle cracks to date demonstrate that the development of BMI nozzle cracks above the J-groove weld large enough to result in a nozzle failure, the release of loose parts into the RCS, and possible significant consequential damage is not a credible event for the following reasons:

- The cracks identified in penetrations 1 and 46 were axial in nature, appearing at right angles to the residual hoop stresses created by J-groove welding.

- No developing cracks or flaws were identified in any nozzle other than 1 and 46.
- No circumferential cracks were detected as a result of the extensive NDE program implemented at STP Unit 1.
- Analytical stress analysis results show that the growth of axial cracks above the J-groove weld to the point where nozzle failure could potentially occur is highly unlikely.

The discussion above demonstrates that a BMI nozzle failure above the J-groove weld releasing loose parts to the RCS is not a credible event.

Conclusion on Consequences

The above sections discuss, evaluate and eliminate consequences beyond minor leakage. This establishes minor leakage as the consequence of BMI nozzle cracks. Coupling this consequence with effective visual inspection programs enhances STP's ability to identify and manage any future BMI indications.

V. Investigation and Evaluation

E. Failure Modes and Effects Analysis (FMEA)

5. Current Status of Examinations and Evaluations

All NDE activities have been completed. The remaining identified potential crack initiators fall into three major areas:

- Excessive residual stresses;
- Fatigue; or
- PWSCC.

Excessive residual stresses include those resulting from fabrication activities such as welding, grinding, nozzle straightening, etc., which can leave small flaws at the nozzle OD/weld interface or embedded within the J-groove weld itself. Fatigue includes fatigue mechanisms primarily associated with plant operations such as heatup, cool down and pressurization/depressurization cycles.

The early data and analysis from the NDE program (primarily the volumetric UT inspection) suggested that the NDE program alone does not provide sufficient conclusive information to allow a definitive root cause determination of the BMI nozzle cracks to be made. Samples of identified flaws and detailed metallurgical analysis of those samples should provide additional data and evidence to allow further characterization of the crack initiation and propagation mechanisms. Complete removal of a nozzle and J-groove weld was deemed impractical because

of a lack of industry experience with this magnitude of repair on the RPV bottom head.

Boat sampling was selected as the most viable and practical means to capture a portion of nozzle and weld material containing a crack. Analysis has been performed to assure that the RPV integrity will be maintained even if the boat-sampling operation were to uncover an unrepairable flaw. STP is planning to remove one boat sample from each of nozzles 1 and 46 to obtain cracked material from both the nozzle and weld regions. Boat samples are to be taken mid-July 2003, and STP anticipates completion of the metallurgical analysis of the removed boat samples by late September 2003.

In effecting the half-nozzle repairs of penetrations 1 and 46, about 9 inches of the old nozzles below the RPV were removed. Metallurgical analyses of the removed half nozzles are scheduled to confirm the chemical, mechanical and metallurgical characteristics of the nozzle material, establish their relative susceptibility to PWSCC and to aid in the evaluation of potential generic concerns.

V. Investigation and Evaluation

F. Fault Tree Analysis (FTA) and Conclusions

As a process validation effort, a Fault Tree Analysis (FTA) was developed to parallel the FMEA evaluations. Attachment 3 contains this FTA and associated notes.

V. Investigation and Evaluation

G. Results of FMEA and FTA

Review of the Streamlined FMEA, Attachment 2, and the FTA, Attachment 3, results in the following conclusions:

- Cracks were detected in nozzles 1 and 46 only.
- The cracks in nozzles 1 and 46 are tight, and have resulted in extremely low leak rates that have been determined to be at least 3 to 5 years old.
- No cracks have been found in the other 56 BMI nozzles.
- No circumferential cracks have been found
- By analysis, nozzle 1 would be least likely to have the highest residual stresses since it is nearest the center of the vessel.
- The cracks in nozzles 1 and 46 do not appear to originate near the highest stress region in those nozzles as identified by analysis.
- If PWSCC by itself were the initiating mechanism, statistical analysis indicates that other cracks would be identifiable and would have to be addressed.

- PWSCC does not explain the three cracks that do not appear to make contact with primary coolant.
- Analytical prediction of residual stresses due to welding indicates that these stresses are insufficient to initiate ductile tearing cracks in the BMI nozzles.
- Analysis demonstrates that stresses at the J-groove weld aggravated by fabrication operations such as grinding and weld repairs can approach or exceed material yield strength.
- Residual stresses alone are not sufficient to explain why the cracks appear in only nozzles 1 and 46. Nozzles further out on the vessel head are analyzed to have higher residual stresses than either nozzle 1 or 46.

These considerations facilitated a further refinement of possible FMEA conditions. The Refined FMEA (Attachment 4) identifies the most apparent cause for the BMI cracks as manufacturing (welding) flaws resulting in excessive stress in the nozzle/weld material leading to crack initiation with low cycle fatigue/primary water stress corrosion cracking then supporting crack propagation.

VI. Investigation Summary

There are two possible causes for the cracks identified in Penetrations 1 and 46.

As shown in the Failure Modes and Effects Analysis chart (Attachment 1), a number of causal factors could increase the PWSCC susceptibility of the Alloy 600 material. Consequently, PWSCC is a possible crack initiating mechanism.

However, several facts are inconsistent with PWSCC as the primary mechanism.

- No cracks were identified in any penetrations other than 1 and 46. This is not indicative of a random time dependent, progressive mechanism, such as PWSCC.
- Based on UT results, cracks 2 and 3 in Penetration 1 and crack 2 in Penetration 46 do not appear to be in contact with primary water wetted surfaces.

A number of facts suggest that fabrication flaws associated with the J-groove welding or weld repairs could be significant.

- The cracks did not occur at the highest predicted residual stress locations. Some other influence must be responsible for higher local stresses.
- Penetrations 1 and 46 are different in terms of residual stresses. PWSCC cracks are a higher probability in Penetration 46 and other "hillside" penetrations, than in Penetration 1.
- The cracks that resulted in leakage in both penetrations are relatively old based on leak residue analysis and are about the same age. This observation

suggests a single event or point in time, rather than a random, time dependent process.

As noted earlier, the visual examination of the BMI nozzles inside the reactor vessel revealed a number of grinding marks on the outside walls of the tubes. Considering that dye penetrant examinations were performed at least three times during the installation of each J-groove weld and that grinding was performed to prepare the weld for the penetrant examinations, welded-over grinding areas within the J-groove weld can be expected. Small "discontinuities" or UT reflectors were noted in Penetration 1 and 46 as well as other nozzles. These discontinuities are likely minute flaws, possibly porosity or areas of lack of fusion, associated with installation of the J-groove weld. Because the penetrant examination is a surface examination, such flaws would not necessarily be detected.

The cracks in Penetrations 1 and 46 are located near the edges of the weld discontinuities indicated by UT examinations. Because there are no cracks associated with similar discontinuities in other nozzles, it appears that only certain fabrication flaws result in locally high stresses and associated cracks.

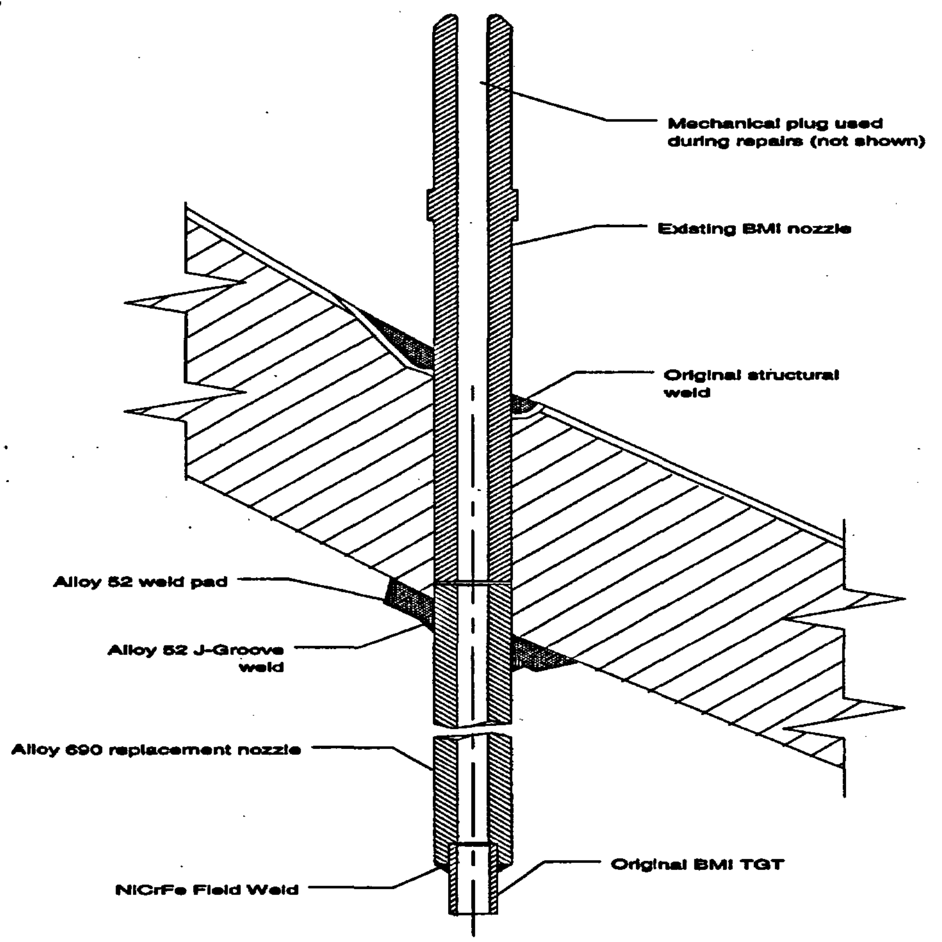
Due to the observations which are inconsistent with PWSCC as a primary mechanism, manufacturing (welding) flaws resulting in excessive stress in the nozzle/weld material are probably responsible for initiating the cracks in Penetrations 1 and 46.

VII. Repair

Industry Operating Experience reviews identified several methods of repairing J-groove installed penetrations on thick walled vessels. These repairs were performed on CE PWR pressurizer penetrations. Several contacts were made to industry repair organizations regarding availability, effectiveness, processes, analysis requirements and potential future consequences of the different repair methods. After thorough evaluation, STP chose to utilize a half-nozzle repair configuration. The STP design of the half-nozzle repair/replacement is consistent with the designs used in the nuclear industry for both pressurizer heater sleeves and RCS piping instrumentation nozzle leaks.

As shown in Figure 5, the half-nozzle repair/replacement consists of removing the lower portion of the Alloy 600 BMI nozzle partially inside the vessel head penetration. A weld pad was then installed on the outside surface of the vessel using alloy 52 filler material and the temperbead process. A replacement lower nozzle fabricated from SB-166, Alloy 690 material was then attached to the weld pad by a partial penetration weld of Alloy 52, thereby forming the new pressure boundary at the outer surface of the vessel. The repair leaves a portion of the low alloy steel reactor vessel exposed to primary water by way of the existing nozzle cracks and the gap between the new Alloy 690 half-nozzle and the existing Alloy 600 nozzle. STP performed a site-specific component corrosion analysis

addressing exposure of the reactor vessel to primary water. The results of this analysis show that the effects of general corrosion on the reactor vessel for the remainder of the 40-year design life and a hypothetical 60-year extended life are acceptable.



Typical
Half-Nozzle Repair/Replacement Design

FIGURE 5

This repair design is similar to designs used in the nuclear industry to repair Alloy 600 nozzle cracks in pressurizers and piping instrumentation nozzles. STP Engineering has reviewed the design basis and repair analyses for the half-nozzle repair/replacement, and the results of this review show that ASME code allowable stress limits are met.

Moving the pressure boundary to the outside of the vessel is consistent with ASME Code Section III. An analysis of the half-nozzle repair is the basis for an amendment to the certified design reports. Relief Requests for alternative requirements that were needed to facilitate the half-nozzle method repair/replacement have been submitted for NRC approval.

BMI nozzle weld cracking results in the unsatisfactory condition of pressure boundary leakage. Analysis indicates that the cracks do not impact either the structural integrity of the BMI nozzles or the primary function of the nozzles to provide a guidance passage for instrumentation thimbles to enter the core region of the vessel in support of flux mapping.

Retaining the structural integrity and functions of the existing BMI nozzles and eliminating leakage was accomplished by relocating the pressure boundary to the outside surface of the RPV. The half-nozzle repair uses improved materials and has no adverse impact on inspectability of the lower head.

VIII. STP Generic Implications

The investigation, inspection and analysis of the STP BMI nozzle cracks to date indicate a low likelihood that the observed cracks were initiated by a random time dependent mechanism such as PWSCC. No cracks were identified in any nozzle other than nozzles 1 and 46. Limited cracking in nozzles 1 and 46 and the absence of cracks in other nozzles indicate that irregularities resulting from the normal fabrication processes generated atypical residual stresses or stress risers which were confined to these two nozzles.

Other RCS locations with the same configuration, materials and installation processes have the same vulnerability to this type of failure. Experience from the Unit 1 bottom head is that leaks develop slowly, and that established inspection programs are adequate to identify and locate any leakage that might develop long before the leakage or any adverse effects of leakage on the structural integrity of affected components become significant.

To validate these conclusions, STP will examine Unit 2 penetrations with a volumetric NDE method at the next Unit 2 refueling outage where the core barrel is planned to be removed (currently 2RE11 scheduled for 2005).

IX. Corrective Actions

Due to the indication of leakage at BMI penetrations 1 and 46, Technical Specification 3.4.10 for reactor coolant system structural integrity was not met. Action a. was implemented limiting RCS temperature to 130 degrees.

Half-nozzle repairs of penetrations 1 and 46 have been implemented. Post modification NDE will be completed before unit restart efforts commence. Post modification inservice pressure testing will be part of the required Reactor Coolant System Leakage Pressure Test in accordance with the station's System Pressure Testing Program.

Metallurgical analysis is currently being performed on the nozzle ends removed during the repair of penetrations 1 and 46.

A boat sample from each of nozzles 1 and 46 and their associated J-groove weld region is scheduled mid-July 2003. These boat samples are intended to capture nozzle and weld material containing cracks. These boat samples will be subjected to extensive metallurgical examination and analysis. All information gathered will be evaluated and reconciled with current information to determine to the extent possible the mechanism(s) of both crack initiation and propagation. These boat sample metallurgical examinations are expected to be complete by mid-September 2003.

An updated cause report will be submitted during October 2003 following completion of the boat sample metallurgical analyses to incorporate any new findings and insights gained.

The System Pressure Testing Program that requires a Reactor Coolant System Leakage Pressure Test at the end of each refueling outage, together with the Boric Acid Leaks Program that requires regular inspections of the RPV, provides long term monitoring of the BMI penetrations. These inspection programs have proven effective in the early detection and location of BMI nozzle leaks. No new actions or commitments are required to assure these continued actions.

To confirm beyond the comprehensive inspections already completed that the identified cracks in nozzles 1 and 46 are isolated flaws, STP will perform enhanced visual and volumetric (UT) inspections of the Unit 1 BMI nozzles during the next inservice inspection of the vessel.

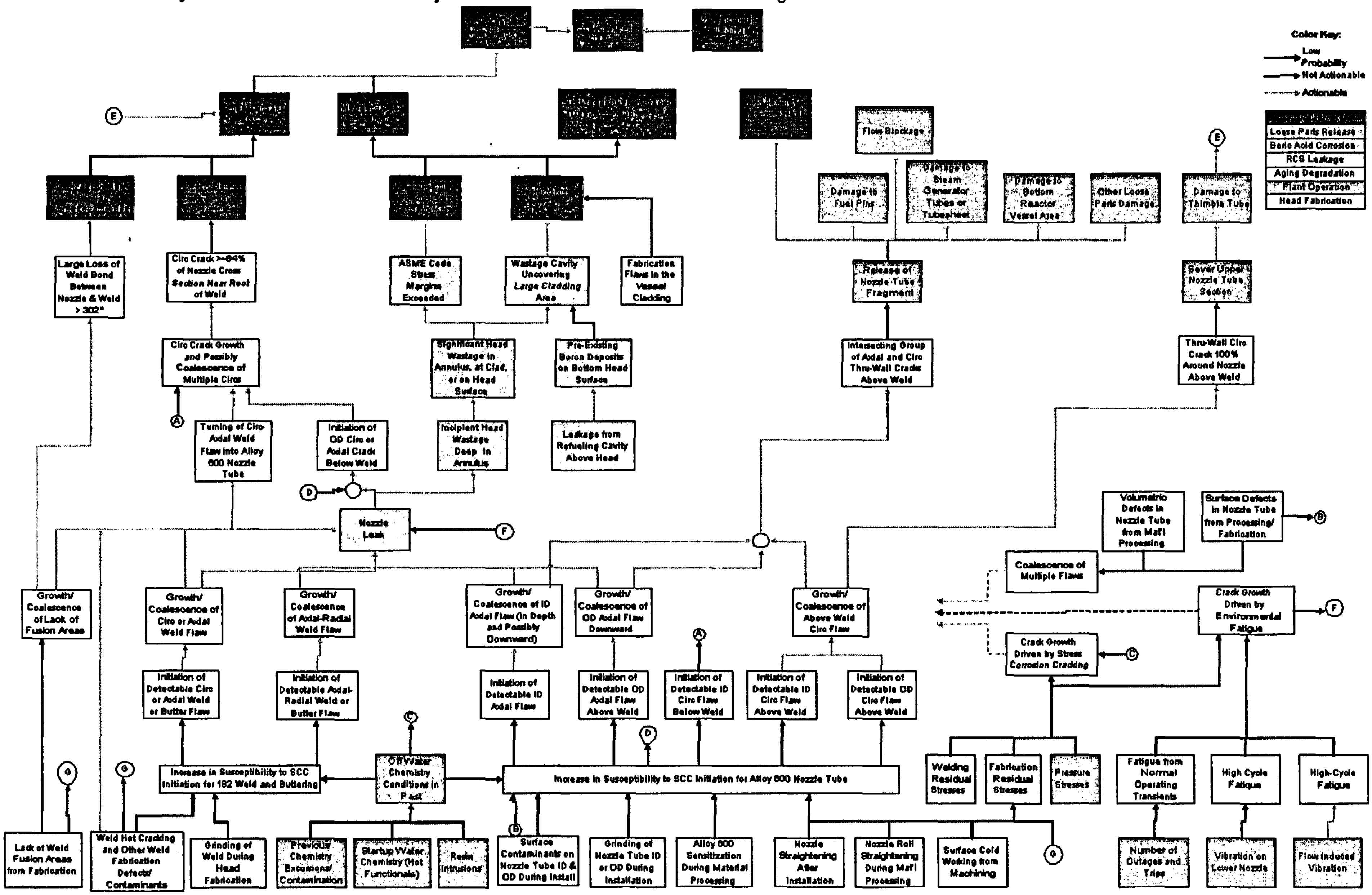
STP will perform ultrasonic examinations of the reactor pressure vessel base metal around the repaired penetrations at future selected Unit 1 refueling outages to confirm there are no indications of pressure vessel wastage from RCS water in the gap area of the repaired penetration. The details of these examinations will be described in ASME Code relief requests for the repairs.

X. Attachments

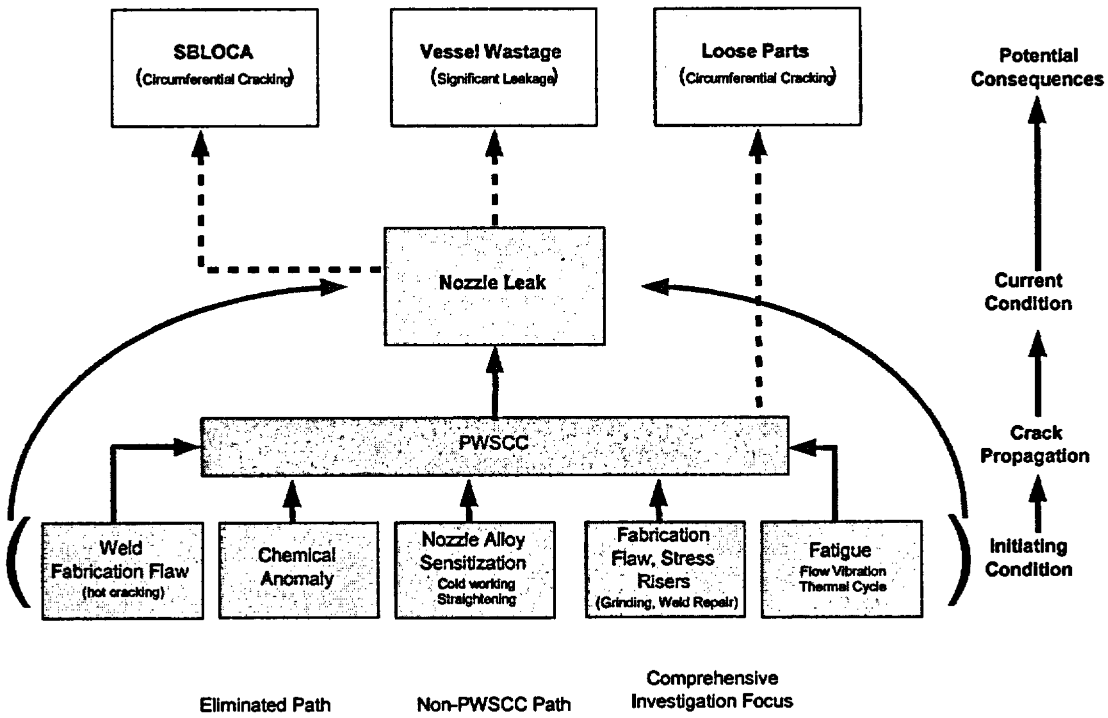
1. Failure Modes and Effects Analysis Chart
2. Streamlined Failure Modes and Effects Analysis Chart
3. Fault Tree Analysis
4. Refined Failure Modes and Effects Analysis Chart
5. Two Dimensional Representation of Cracks in Nozzle 1
6. Two Dimensional Representation of Cracks in Nozzle 46

South Texas Project Status Report on Resolving the Unit 1 BMI Issue – Attachment 1
Failure Modes and Effects Analysis Chart

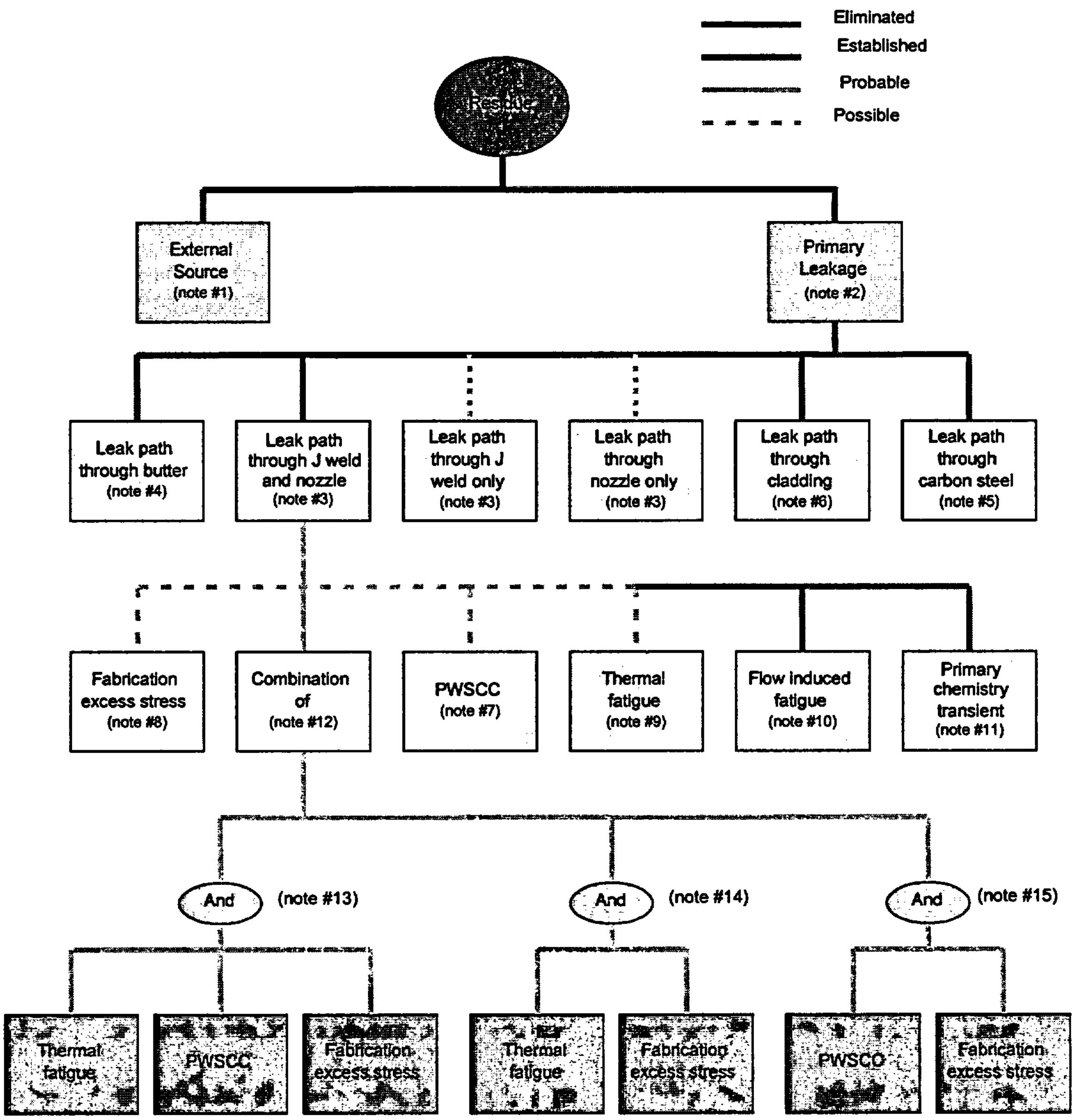
South Texas Project's Failure Modes and Effects Analysis for Bottom Mounted Instrument Nozzle Leakage July 11, 2003



South Texas Project Status Report on Resolving the Unit 1 BMI Issue – Attachment 2 Streamlined Failure Modes and Effects Analysis Chart



South Texas Project Status Report on Resolving the Unit 1 BMI Issue – Attachment 3 Fault Tree Analysis



Note #1: Swipes and visual inspection of the outside surface of the RPV do not support outside source for the boron residue.

Note #2: Boron, lithium, and cesium content confirm RCS coolant as source of residue.

Note #3: UT examination and the pressure test identified a leak through the weld and nozzle 1. UT examinations suggest the cracks in nozzles 1 and 46 originated at the nozzle weld interface. A crack originating at this location most likely involves both the nozzle and the weld. The investigation path for nozzle, weld and combined nozzle and weld is the same. The boat sample result will be the best opportunity to separate between the nozzle and the weld.

Note #4: Visual and ECT examination did not identify any surface breaking flaws.

Note #5: Visual examination did not identify any surface breaking flaws.

Note #6: Visual and ECT examination did not identify any surface breaking flaws.

Note #7:PWSCC requires three things; 1) susceptible metal, 2) residual stress, and 3) an aqueous environment. Only two of five nozzle cracks are confirmed to be in contact with primary water. Although these cracks appear to have their origin within the welded zone, it is possible, but unlikely, they could exist by PWSCC alone.

Note #8: Fabrication flaws could have been present below the weld surface during fabrication and migrated to the surface during the vessel hydro test.

Note #9: This is low cycle fatigue due to temperature and pressure changes. If fatigue alone were responsible for the flaws there should be more flaws in other nozzles. Fatigue is a possible source of the cracks but more likely in combination with pre-existing fabrication defects.

Note #10: High-cycle vibration such as flow induced vibration is not considered practical since analysis and measurements indicate that the natural frequency of the nozzles and the vibrations imparted through RCS circulation are two orders of magnitude apart.

Note #11: There are no records of significant RCS chemical transients. STP experience with the Alloy 600 steam generator tubes, which are much thinner, have not indicated any degradation due to RCS chemical contaminats.

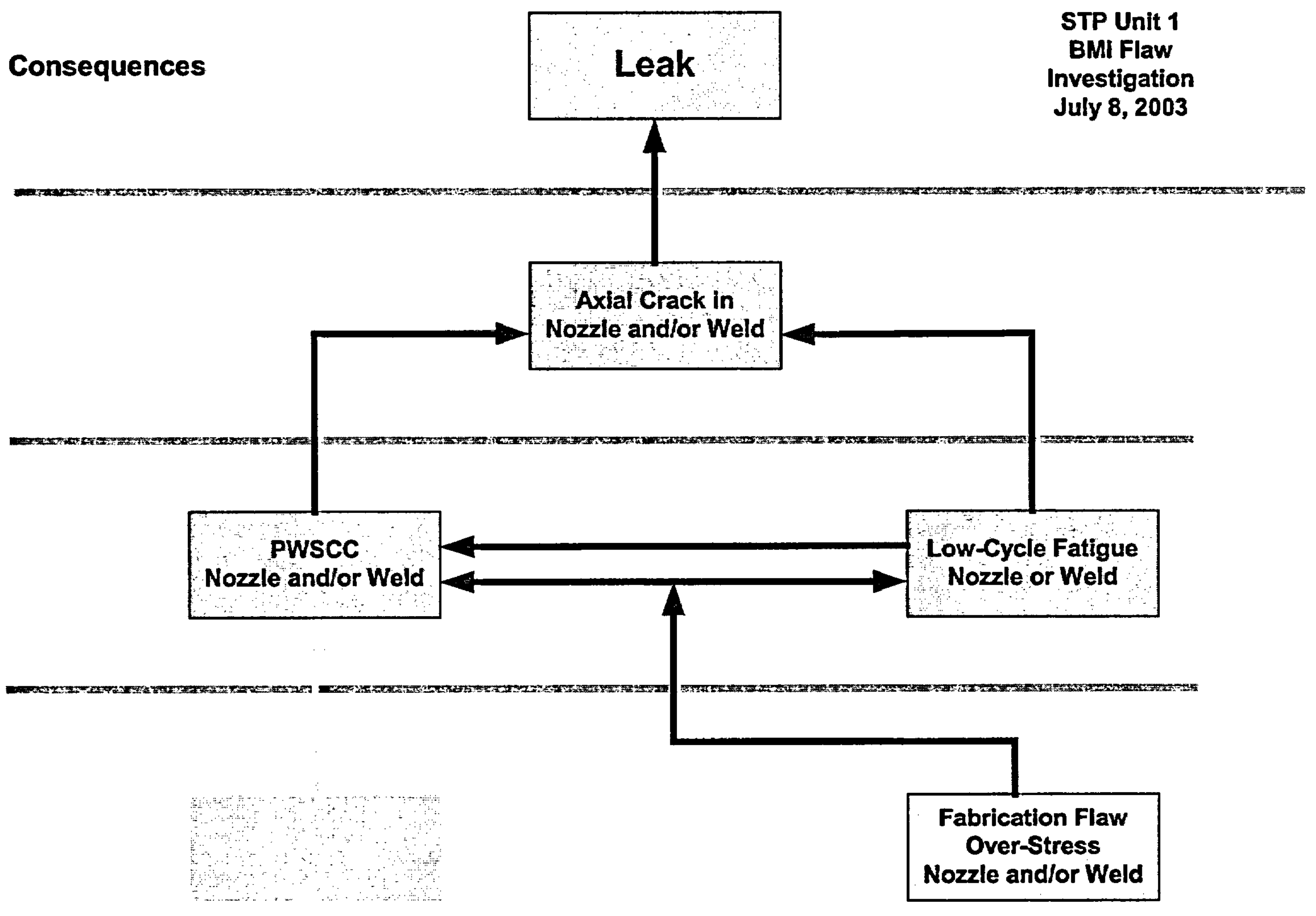
Note #12: Cracks that do not extend beyond the weld zone, (i.e., not in contact with primary water) Are most likely the result of a combination of fabrication flaws

and low-cycle fatigue. However, cracks that are in contact with primary water may be a combination of PWSCC, fabrication flaws, and low-cycle fatigue.

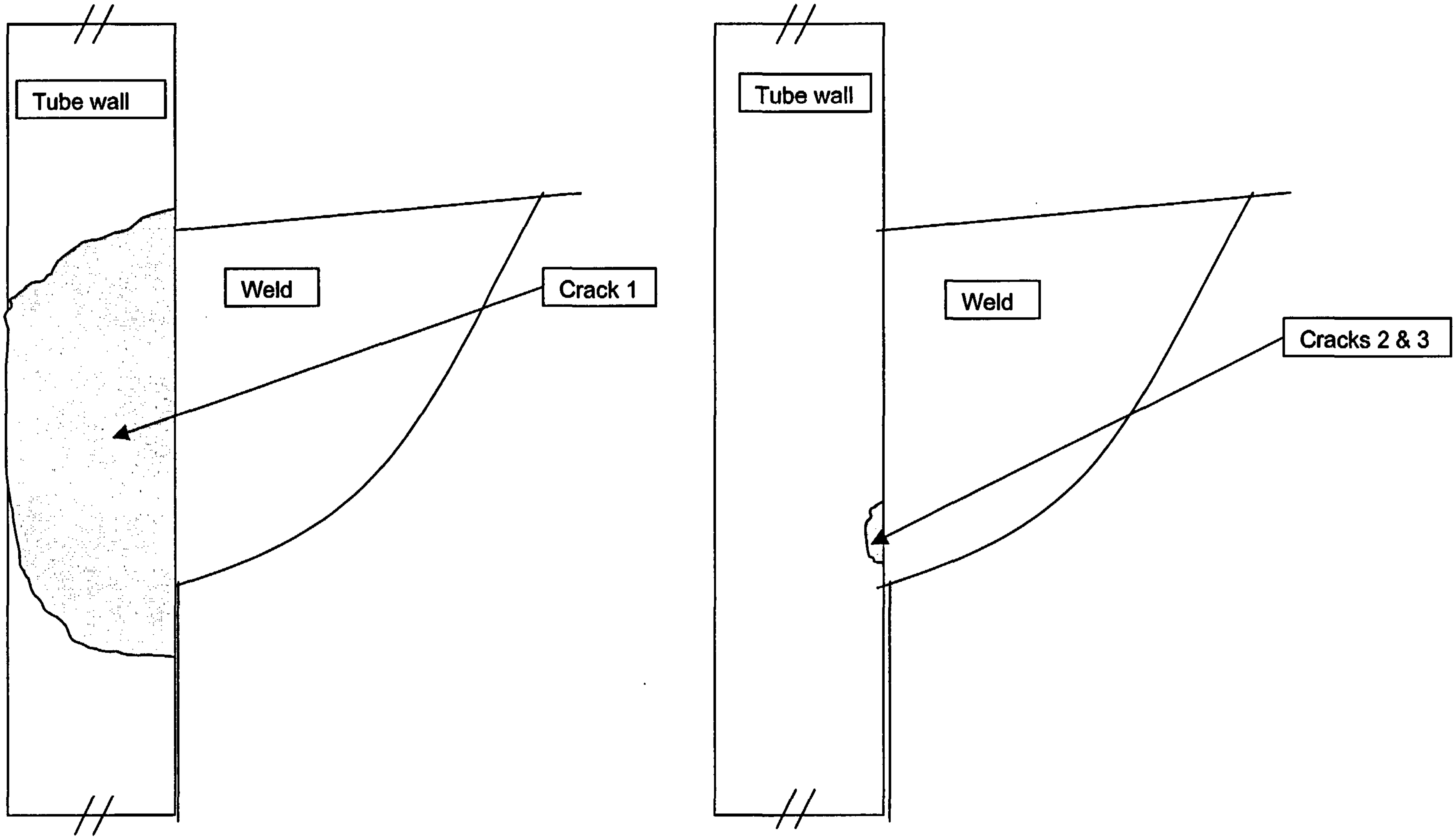
Note #13: Possible for cracks that appear to have their origin within the weld and extend to the point of contact with primary water.

Note #14: Most likely source of cracks totally imbedded in the weld zone but could also be possible for cracks that extend beyond the weld.

Note #15: This postulates a fabrication flaw that was missed by the final PT examination or propagated to the surface during the initial hydro vessel testing and later grew through the weld zone by PWSCC.


Consequences
Leak
STP Unit 1
BMI Flaw
Investigation
July 8, 2003
Axial Crack in
Nozzle and/or Weld
PWSCC
Nozzle and/or Weld
Low-Cycle Fatigue
Nozzle or Weld
Fabrication Flaw
Over-Stress
Nozzle and/or Weld

South Texas Project Status Report on Resolving the Unit 1 BMI Issue – Attachment 5 Two Dimensional Representation of Cracks in Nozzle 1



South Texas Project Status Report on Resolving the Unit 1 BMI Issue – Attachment 6 Two Dimensional Representation of Cracks in Nozzle 46