James A. Spina
Vice President

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Constellation Energy
Generation Group

January 16, 2006

U. S. Nuclear Regulatory Commission
Washington, DC 20555

ATTENTION: Document Control Desk

SUBJECT: Calvert Cliffs Nuclear Power Plant
Unit No. 1; Docket No. 50-317
Class 1 Piping Operability Evaluation Submittal per Code Requirements

An anomaly was discovered in an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Class 1 Reactor Coolant System line (Field Weld No. 1 of spool piece 1-CC-14) during the review and digitization of original construction weld radiographs by Constellation Energy engineering staff prior to the 2006 Unit 1 refueling outage at Calvert Cliffs Nuclear Power Plant. The review of original construction radiographs was performed proactively to provide early review and identification of pre-existing flaws in Class 1 piping system components.

The initial construction radiograph of the Unit 1 shutdown cooling outlet nozzle safe-end-to-pipe weld (Field Weld No. 1 of spool piece 1-CC-14) identified a slag inclusion that exceeded the acceptance criteria of American National Standards Institute B31.7 (original construction code). A weld repair was performed during original construction and a follow-up radiographic test indicated that, although reduced in size, the inclusion remained outside of the B31.7 acceptance criteria. Prior to initial plant operations, a pre-service inspection ultrasonic test was performed on Field Weld No. 1 of spool piece 1-CC-14 in accordance with ASME B&PV Code Section XI standards. A second ultrasonic test was performed in 1994, in accordance with ASME B&PV Code Section XI, during an inservice inspection of the same weld. No indications were identified during either Section XI ultrasonic examination.

As a result of the discovery made during review and digitization of the original radiograph, an operability determination was initiated on November 9, 2005, in accordance with site procedures. The initial determination indicated the subject weld was operable but degraded and further evaluation would be required to adequately disposition the indication. A fatigue analysis completed by Structural Integrity Associates, Inc. on November 15, 2005, determined that flaw growth would be small enough to safely allow continued operation for at least two operating cycles.

Calvert Cliffs Nuclear Power Plant intends to perform additional non-destructive examinations of the subject weld during the 2006 Unit 1 refueling outage, which is planned to start in February 2006. Results of the additional non-destructive examinations are expected to allow more accurate characterization of the radiographic indication. This should allow further refinement of the fatigue analysis by reducing conservatisms included in the initial analysis, thus demonstrating the weld is adequate for the remaining life-of-the-plant. An alternative to leaving the weld in place, as-is, would be to remove and repair the

A047

weld during the 2006 or 2008 Unit 1 refueling outage. This option would be used only if additional characterization determines that the flaw growth is unacceptable for the remaining life-of-the-plant.

The attached operability determination is provided for Nuclear Regulatory Commission review and approval in accordance with ASME B&PV Code Section XI requirements contained in IWB-3640, "Evaluation Procedures and Acceptance Criteria for Austenitic Piping." Calvert Cliffs Nuclear Power Plant intends to provide Nuclear Regulatory Commission with additional evaluation results after completion of the Unit 1 2006 refueling outage, but in no case later than the start of the Unit 1 2008 refueling outage.

Should you have questions regarding this matter, please contact Mr. L. S. Larragoite at (410) 495-4922.

Very truly yours,

Joseph E. Pollock

for
James A. Spina
Vice President - Calvert Cliffs Nuclear Power Plant

GV/MJY/bjd

Attachment: (1) Calvert Cliffs Nuclear Power Plant Operability Determination No. 05-004R1

cc: P. D. Milano, NRC
S. J. Collins, NRC
Resident Inspector, NRC
R. I. McLean, DNR

ATTACHMENT (1)

CALVERT CLIFFS NUCLEAR POWER PLANT

OPERABILITY DETERMINATION NO. 05-004R1

Calvert Cliffs Nuclear Power Plant, Inc.
January 16, 2006

OD NO.:05-004R1 DATE/TIME INITIATED: 11/15/05/12:21 (Same OD number used on Attachment 7)

UNIT: 1 ISSUE REPORT #: IRE-009-389

EQUIPMENT/COMPONENT DESCRIPTION: (SYSTEM#/COMP#/UEI#/ETC.)052

OPERABILITY RECOMMENDATION CHECKLIST

CHECK ONE OF THE FOLLOWING:

1. ☒ The affected structure/system/component (SSC) should be declared OPERABLE as reasonable assurance exists which indicates that the degraded/non-conforming SSC WILL PERFORM its intended safety function(s) as required.

2. ☐ The affected structure/system/component (SSC) should be declared INOPERABLE as reasonable assurance of the SSC functionality DOES NOT exist and the degraded/non-conforming SSC WILL NOT PERFORM its intended safety function(s) when required. Terminate the use of this attachment and immediately inform the GS - NPO or Shift Manager of the inoperability.

DOCUMENTATION OF OPERABILITY RECOMMENDATION

1. Description of the issue/situation (that resulted in the need for the Functional Evaluation):

While performing 2006 refuleing outage preparations, original construction weld radiograph information was reviewed. During the review, it was determined that Field Weld # 1 in the class 1 Shut Down Cooling (SDC) line has a non metallic inclusion that should not have been accepted during construction. Construction code B31.7 Appendix B-1-140, 1969 Edition allows the acceptance of inclusions with an aggregate length no greater than the thickness of the pipe when there is a line of inclusions. In this case, there are a group of three small inclusions with a combined length of 1.5". Because the thickness of the pipe is 1.125", the inclusions should not have been accepted during the original analysis of the radiograph.

After acceptance of the construction weld, a Pre-Service Inspection (PSI) was completed on the weld. This was conducted in accordance with the requirements of ASME Section XI, 1970 Ed, Summer 1970 Addenda. This section directed the use of ASME Section III, 1970 Edition, Summer 1970 Addenda. The examination examined essentially the full volume of the weld, using calibration blocks with 1/4t, 1/2t, and 3/4t side drilled holes. This exam did not detect the presence of any recordable indications.

An ASME Section XI exam was satisfactorily completed in 1994. This exam was completed in accordance with ASME Section XI, 1983 Ed, Summer 1983 Addenda. The exam did not detect the presence of any weld inclusions. The exam volume of the ASME Section XI weld exam is the inner 1/3 thickness of the weld. This exam was conducted from both sides of the weld. While the Section XI exam concentrated on the inner 1/3 of the weld, the exam technique used an extended beam path that examined, essentially, the full volume of the weld. The inclusion could be outside of this volume or it could be aligned such that the angle beams used in the ASME Section XI UT were not able to detect the inclusion.

2. Impact on Nuclear safety and operation (Describe the potential or actual impact of the issue/situation on nuclear safety and operations):

Due to the characteristics of the indication, there is no immediate impact on Nuclear Safety. Because the inclusion was found during the construction weld RT, it is not a service induced flaw. There are three potential causes which could influence growth:

1. Pressurized Water Stress Corrosion Cracking (PWSCC)
2. Inter Granular Stress Corrosion Cracking (IGSCC)
3. Fatigue

Because the PSI and the 1994 ASME exams did not identify any indications at the ID of the pipe, PWSCC and IGSCC can be eliminated as potential phenomenon which would cause the flaw to grow. Without contact with the pumped fluid, conditions are not available to induce this type of flaw growth.

Structural Integrity Associates, Inc. (SI), performed fatigue analysis using conservative assumptions and determined that the flaw growth would be slow enough to allow continued operation for at least two operating cycles (until 2010 RFO) with margin. The analysis is based on the assumption that:

1. The inclusion is a 1.5" circumferential flaw
2. The flaw initiates at the pipe OD (a satisfactory PT examination was performed on the outside of the pipe during the 1994 volumetric examination)
3. The flaw extends 66% through the wall of the pipe (based on the satisfactory documented results of the inner 1/3 during the 1994 volumetric examination)

3. Regulatory requirements/commitments (Describe the potential or actual impact of the issue/situation on the Current License Basis):

Acceptance standards for welds are identified in IWB-3131, which references table IWB-3410-1, which states the acceptance standards of IWB 3514 are to be applied.

For the assumptions placed on this flaw, the acceptance criteria of table IWB-3410-1 are not met. IWB-3131 goes on to require the condition to be corrected under IWB-3132.2 or IWB-3132.3

IWB-3132.3 permits an analytical evaluation of the flaw as in IWB-3600.

IWB-3640 contains the evaluation procedures and acceptance criteria for austenitic piping. SI performed the analysis to these procedures and criteria and found the flaw to be acceptable for at least 4 more years of service, or 2 more operating cycles, beyond the current operating cycle. In accordance with IWB-3640, the evaluation procedures and acceptance criteria shall be the responsibility of the owner and shall be subject to approval by the regulatory authority having jurisdiction at the plant site. Based on this requirement, CCNPP must obtain NRC approval of the SI evaluation before restoring the system to full qualification and closing this NO-1-106.

Given the need for NRC approval of the SI evaluation, CCNPP shall remain in, TRM 15.4.3, Structural integrity of ASME Code class 1, 2 and 3 components shall be within the limits of the In-service Inspection Program, until the NRC approves the SI evaluation.

Tech Spec 3.4.13, RCS operational leakage shall be limited to no pressure boundary leakage. The system is in compliance with this Tech Spec. There has been no increase in RCS leakage and the bare metal visual inspection of a weld in close proximity during the 2004 RFO revealed no indication of leakage.

4. Structure/System/Component (SSC) safety function(s) (Fully describe the SSC safety functions, particularly those that are potentially impacted due to the issue/situation):

The safety function of the affected pipe is to provide a qualified pressure boundary connection between the Reactor Coolant System (RCS) and the SDC system and an RCS pressure boundary to the containment atmosphere while at power.

Revision Number:_____

Basis for the Revision:_____

5. Evaluation:

A. Scope of evaluation:

This evaluation will restore the system to full qualification through 2010.

B. Applicable specific events and scenarios (associated with the issue/situation):

This evaluation will apply to all modes of operation when the affected SDC line is required to be operable.

C. Givens/assumptions (information that supports the specific evaluation, including adverse impact):

For the purpose of this evaluation, the following assumptions were conservatively made.

1. The 1.5" long circumferential weld flaw is 66% through the wall thickness from the OD of the pipe.
   - Past inspections of this weld include a post construction PSI. This was essentially a full volume inspection of the weld that did not detect the presence of any recordable indications. More recently, the inner 1/3 thickness of the weld was examined in 1994. Again, the inspection did not detect the presence of any weld inclusions. Based on these two examinations, it was conservatively assumed that the weld flaw is 66% (0.7425") through the wall thickness from the OD of the pipe.

2. The flaw is connected to the OD of the pipe
   - During the 1994 inspection, a PT inspection of the weld was performed with no indications identified. Conservatively, this evaluation will assume that the flaw is connected to the OD of the pipe.

Until the issue with this weld is resolved, Operations will continue to monitor RCS leak rate for increasing trends and identify the source IAW the guidance in Operation's Standing Order 03-03 "RCS Leakage".

D. Specific evaluations (Document the results and long-term capabilities of the SSC):

SI has performed an analysis of the propagation of a fatigue flaw under the conservative assumptions outlined above. Their analysis (see attached) indicates that flaw growth would be relatively modest such that the flaw can be shown to meet ASME Section XI allowable flaw criteria for at least two more operating cycles (2010 RFO) with margin.

E. Method to restore SSC (e.g.: repair, Mod) (Document only the intended actions to restore the SSC to full qualification)

IAW IWB-3640, NRC approval of the SI evaluation will be required to restore the system to full qualification through 2010.

F. Estimated Completion Date (ECD) (For each action):

- Obtain NRC approval of the SI evaluation - IR200500308 ms 003 - TBD
- Develop the inspection plan to be performed during the 2006 RFO - IR200500308 ms 004 - 12/16/05
- Identify additional corrective actions following the 2006 RFO inspection -- IR200500308 ms 005 - 5/26/06
- Implement corrective actions following the inspection evaluation -- IR200500308 ms 006 - TBD

G. Expected plant configuration including the effect of Compensatory Actions (Document the safest plant configuration including the effect of any transitional actions).[1]

Based on the stated operability of the weld, it is acceptable to maintain normal plant operations. Operations will continue to monitor RCS leak rate for increasing trends and identify the source IAW the guidance in Operation's Standing Order 03-03 "RCS Leakage".

Because the unit is required to be shut down to perform inspections on the weld, the 2006 RFO will be utilized to perform inspections outlined in section 6.

---

1 Include any special methods or plant conditions needed to perform surveillance testing to maintain operability. [B0496]

6. Recommendations for further evaluation (Why should it be considered):

- Determine an evaluation plan to inspect the weld during the 2006 RFO. Based on the ASME Section XI inspections that have been performed to date and the SI analysis that conservatively confirms continued operability until the 2010 RFO, the weld is considered operable at this time. To substantiate this position, the weld in question will be interrogated closely during the 2006 RFO.

  This inspection will:
  1. Verify/locate the inclusion documented on the original construction radiograph inspection
  2. Validate the assumptions used in the SI analysis
  3. Determine if repairs are necessary

- Identify additional corrective actions required following the 2006 RFO interrogation of the subject weld. This will determine if the weld must be repaired or can be accepted as is.
- Implement corrective actions developed during the evaluation of the 2006 RFO inspection results

7. References (Supports the specific evaluation):
   1. SI analysis CA06657
   2. ASME Section XI, 1998 Edition, Section IWB-3112(a)
8. Attachments (Applicable items in Step 7):
   1. SI analysis CA06657
   2. IRE-009-389
9. Equipment is (Check One):

OPERABLE ☒ INOPERABLE ☐

	Signature	Date	Time
Prepared by:	[signature]	11/16/05	1535
Reviewed by:	[signature]	11/16/05	1640
Approved by GS-PES: (or designee)	[signature]	11/16/05	1650

Recommendation is (Check One):

ACCEPTED ☒ REJECTED ☐

	Signature	Date	Time
GS-NPO: (or designee)	[signature]	11/16/05	1700

If Recommendation is REJECTED, provide reasons below:_____

AIT No.: _____ IR No.: _____

10. Inactive Operability Determination

GS-PES (or designee): ____

GS-NPO (or designee): ____

Original To: Control Room's Active Functional Evaluation/Operability Determination Book.

Upon completion, if no action is to be taken, then process this Attachment per Section 7.0.

CR # IRE-009-389

CONDITION REPORT

MO #: 1200504256

PART A - INITIATOR

1. Do you have Personnel/Equipment Safety Concern? N
2. Do you have an Operability Concern? N
3. Do you have a Reportability Concern? N
4. Do you have a Potential Trip or Reactivity Concern? N
5. Should the area/equipment be Quarantined? N

Additional Information Attached? Y
ERO Related? N

6. Condition Descr.: WHILE REVIEWING AND DIGITIZING THE ORIGINAL CONSTRUCTION RADIOGRAPHIC FILM TO SUPPORT ISI, AND DISSIMILAR METAL WELD EXAMINATIONS DURING THE 2006 OUTAGE REVEALED AN INDICATION THAT DID NOT MEET CODE. BECHTEL WELD NUMBER 1 ON DRAWING 1-23-10, LINE CC-14 HAS A SLAG INCLUSION THAT EXCEEDS THE ACCEPTANCE CRITERIA IN ANSI B31.7 (ORIGINAL CONSTRUCTION CODE). THE INITIAL INDICATION WAS IDENTIFIED AS SLAG, AND THE WELD WAS REPAIRED. THE RE-SHOT OF THE REPAIR (R1) SHOWS A REDUCTION IN THE LENGTH OF THE INITIAL REJECTED CONDITION, BUT DID NOT REDUCE THE SIZE TO AN ACCEPTABLE LENGTH. THE RADIOGRAPHIC VIEW IS IDENTIFIED AS 9-16, WITH A LENGTH OF 1.78 INCHES.

7. Date/Time Discovered: 11/09/2005 0900
8. Activity In Progress when discovered: RADIOGRAPHIC FILM REVIEW AND DIGITIZATION
9. Immediate Action Taken: INFORMED SUPERVISION,GENERATED CR, DISCUSSED WITH CODE KNOWLEDGEABLE PERSONNEL
10. Is this a Recurring Condition? N
11. Apparent Cause: INTERPRETATION ERROR DURING REVIEW
12. Extent of Condition: UNKNOWN
13. Recommended Actions: DETERMINE IF CODE COMPLIANCE ISSUE EXIST. SEVERAL ISI EXAMINATIONS HAVE BEEN PERFORMED ON THIS WELD AND ACCEPTED.

HARDWARE INFORMATION
14. Unit #: 1
15. Eqp Loc:
16. Vendor/Mfg:
17. UEIs: 1SYS052 - SAFETY INJECTION SYSTEM 052

18. Tag(s) Placed? N Type: Tagnet#: Tag Location:
19. Equip status: OOS? N
If Equipment is still in service, is it Degraded? N

NON-HARDWARE INFORMATION
20. Related Documents:

21. Initiator's Name: REED, ALVIN S Ext: 495-2089 Group: 4A1003 Date: 11/09/2005 Time: 0410

PART B - REVIEWING SUPERVISOR

1. Is this an Immediate Personnel/Equipment Concern? N
2. Do you have an Operability concern in any Mode? Y
Do you have a Reportability Concern? N
4. Do you have a Potential Trip or Reactivity Concern? N
5. Do you have a Plant Tampering Concern? N

Y RECO attached
N Condition could, but does not affect operability of SSC
N Condition made SSC Inoperable, but operability restored
N Condition COULD NOT affect operability of an SSC

Operability Explanation: PIPE IN QUESTION (SDC) IS DESIGNED TO 94.5% OF THE ALLOWABLE VALUE FOR DBE LOADING. THE CODE PROVIDES A SF OF 4:1. ISI EXAMINED IN 1994 AND ACCEPTABLE FOR ASME SECTION XI. WILL NEED TO ACCEPT AS IS OR REMOVE THE FLAW PRIOR TO RESTART FROM 2006 RFO.

6. Recommended Category: II
7. MO Recommended? N
8. Should an outgoing OE be issued? N
9. Fitness for Duty Evaluation Considered? N
10. Compensatory Actions Taken:
11. Was Condition Corrected on the spot? N
12. Recommended Group to Resolve CR:
13. Discussed with:
14. Are further actions required? Y
15. Recommended Group to Resolve Programmatic CR: 4A1003 ENGINEERING PROGRAMS
16. Special Indicators Assigned:
17. Recommended Actions to Resolve Cond: CONDUCT ADDITIONAL NDE DURING 2006 RFO, EVALUATE FLAW, ACCEPT AS IS OR REPAIR TO MEET ACCEPTANCE CRITERIA.
18. CR Approved? Y Name: TIMOTHY LUPOLD Phone: 2283 Approved Date: 11/09/2005

PART C - SHIFT MANAGER REVIEW

1. Is this an Immediate Personnel/Equipment Safety Concern? N
2. Is this an Operability Concern in the current Mode? N
3. Is this a Reportability Concern? RM-1-101 report # N
2a. T.S. #: 15.4 3
4. Is this a Trip or Reactivity Concern? N
2b. This would be an Operability Concern in Mode:
5. Operability Determination Implemented per NO-1-106? N
6. Compensatory Actions Taken:
7. Comments: RESTORE PRIOR TO EXITING 2006 RFO.
8. Name: JAY GAINES Date/Time: 11/09/2005 1700 Phone: 4737

PART D - OMC REVIEW

MO Required? Y
2. Priority: 2
3. Work Type: C
4. Mode to Work: 1
5. RMG: NDE
6. Is CR Programmatic? Y
7. Mode Restraint: N
Mode Code:
8. Shift Manager approval Required prior to starting work? Y

ESP No.:	ES200500643	Supp No.	000	Rev. No.	0000	Page 1 of 12

FORM 19, CALCULATION COVER SHEET

A. INITIATION (Control Doc Type - DCALC)

Page 1 of 12

DCALC No.: CA06657 Revision No.: 0000

Vendor Calculation (Check one): ☒ Yes ☐ No

Responsible Group: Mechanical & Civil Engineering Unit

Responsible Engineer: Andre S. Drake

B. CALCULATION

ENGINEERING DISCIPLINE:
☐ Civil ☐ Instr & Controls ☐ Nuc Engrg
☐ Electrical ☒ Mechanical ☐ Nuc Fuel Mngmt
☐ Other: ☐ Reliability Engrg

Title: Prediction of potential crack growth of weld indication.

Unit ☒ 1 ☐ 2 ☐ COMMON

Proprietary or Safeguards Calculation ☐ YES ☒ NO

Comments: This calculation is for resolution of IRE-009-389 which identified a weld indication on the Shutdown Cooling Outlet Nozzle Safe-End-to-Pipe weld.

Vendor Calc No.: CCNP-06Q-301 REVISION NO.: 0

Vendor Name: Structural Integrity Associates

Safety Class (Check one): ☒ SR ☐ AQ ☐ NSR

There are assumptions that require Verification during walkdown: AIT #:

This calculation SUPERSEDES: N/A

C. REVIEW AND APPROVAL:

	Printed Name and Signature	Date
Responsible Engineer:	Structural Integrity Associates	11/15/05
Owner Acceptance	Andre S. Drake	11/15/05
Approval:	Jack J. McHale	11/15/05

IF the results or conclusions of this calculation or revision might affect a procedure or the basis of a procedure, a Change Notification Form (Form 14) shall be forwarded to the Procedure Development Unit with a summary of the calculation's purpose and results.

List of Effective Pages

Page No.	Revision
1	0
2	0

Appendix 1: Structural Integrity Calculation No. CCNP-06Q-301, "Prediction of Potential Crack Growth Rate of Weld Indication found in the Unit 1 Shutdown Cooling Outlet Nozzle Safe-End-to-Pipe Weld," Rev. 0.

Table of Contents

Reviewer Comments.

1. In Section 2.3 it is mentioned that radiographs were taken in 1994. It is clarified that the NDE method employed in 1994 was an ultrasonic examination. This does not impact the computations, methodology, or conclusions of this analysis.

Appendix 1 to CA06657 (10 Pages)

Structural Integrity Associates, Inc.	CALCULATION PACKAGE	File No.: CCNP-06Q-301 Project No.: CCNP-06Q

PROJECT NAME: Shutdown Cooling Outlet Nozzle Safe End-to-Pipe Weld Indication Evaluation

Contract No.: 416596

CLIENT: Constellation Energy	PLANT: Calvert Cliffs Unit 1

CALCULATION TITLE: Evaluation of the Shutdown Cooling Outlet Nozzle Safe End-to-Pipe Weld Indication at Calvert Cliffs Unit 1

Document Revision	Affected Pages	Revision Description	Project Mgr. Approval Signature & Date	Preparer(s) & Checker(s) Signatures & Date
0	1-10 Computer Files	Original Issue	[signature] 11/15/05	[signature] (P) 11/15/05 [signature] (P) 11/15/05 [signature] (C) 11/15/05 [signature] 11/15/05 (C)

SI Form F2001R2a

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1 INTRODUCTION

Based on information provided by Calvert Cliffs in References 1 and 2, during a recent review of the construction radiographs of the Unit 1 Shutdown Cooling System outlet nozzle safe end-to-pipe weld, a non-metallic inclusion was discovered. The inclusion consists of three closely spaced circumferential inclusions with a total length of 1.5 inches. Based on a 1994 ultrasonic examination of the weld, the inside third of the wall thickness was indication free. The end of the indication (nearer to the outside surface) could not be confirmed based on the available information. The insulation was removed from this weld during the 2004 refueling outage in support of a bare metal visual inspection of the dissimilar metal weld located in close proximity. This provided the opportunity to identify if the weld was leaking. There was no leakage identified when the dissimilar metal weld bare metal visual inspection occurred. Since a surface examination of the weld was not performed, there is no information available to determine if the inclusion observed on the radiograph is connected to the outside surface.

The indication was evaluated using the acceptance standards of the ASME Code [3]. It was concluded that the indication dimensions did not meet the acceptance standard in IWB-3500. The indication evaluation was therefore performed to the requirements of IWB-3600 of the ASME Code. Specifically, since the safe end, connected elbow and weld materials are stainless steels, the provisions of IWB-3640 of the ASME Code were used to perform the evaluation. The details of the evaluation and its conclusions are provided below. It should be noted that the indication is in stainless steel material and it is not connected to the inside surface of the pipe. Therefore, the only mechanism that requires consideration is fatigue.

2 TECHNICAL APPROACH OR METHODOLOGY

The acceptance of the indication in the as-is condition requires consideration of potential crack growth, applied stresses, and allowable flaw size. The allowable flaw sizes incorporate the required safety factors per ASME Code, Section XI, IWB-3640.

2.1 Indication Size

The depth of the indication was found to be 0.7425 inches, or 66% of the wall thickness from the outside surface, as discussed in Section 1. The length of the indication was determined to be 1.5 inches. The pipe thickness at the indication location is 1.125 inches and the inside diameter is 10.5 inches (12" Schedule 140) [4].

2.2 Allowable Flaw Size

ASME Code, Section XI provides acceptance criteria for flaws in austenitic piping. Tables IWB-3641-1 and IWB-3641-2 provide allowable end-of-evaluation period flaw depths for normal and emergency/faulted conditions, respectively. Table IWB-3641-1 of the ASME Code, Section XI gives the allowable depths as a function of stress ratios and the ratio of the flaw length to the pipe circumference. Section 4.1 provides the allowable flaw size calculations.

2.3 Crack Growth

Based on the information provided in References 1 and 2, it is believed that the indication is an original fabrication-related subsurface defect (non-service induced) that could potentially have broken the surface during operation. Potential crack growth mechanisms include stress corrosion cracking (SCC) and fatigue. SCC can be attributed to primary water stress corrosion cracking (PWSCC) or intergranular stress corrosion cracking (IGSCC). PWSCC is not a concern here because stainless steels have been shown to be resistant to PWSCC and the indication is not exposed to the coolant. IGSCC has typically been a problem for the boiling water reactors (BWRs) and has not been a concern for the PWRS due to reduced levels of oxygen in the primary loop. Since this location is subject to thermal cycling, however, crack growth from the time when the radiographs were taken (1994) must be considered.

The fatigue crack growth calculations are presented in Section 4.2.

3 ASSUMPTIONS / DESIGN INPUTS

3.1 Allowable Flaw Size Design Inputs/Assumptions

- Normal Operating Pressure: 2.235 ksi [4]
- Design Temperature: 650 °F [4]
- Moments at weld are provided in Reference 5
- Indication Size per Section 2.1
- Design Stress Intensity: 16.7 ksi (A376, Type 316) [6]
- SMAW or SAW field weld
- Indication is connected to outside surface

3.2 Crack Growth Analysis Design Inputs/Assumptions

- Indication is connected to the outside surface
- Fatigue crack growth is due to system thermal and pressure cycling

4 CALCULATIONS

4.1 Allowable Flaw Size

The applicable stress range formula for SMAW or SAW field welds to input to Table IWB-3641-1 and IWB-3641-2 for allowable flaw size for circumferential flaws is:

$$\text{Stress Ratio} = \frac{Z}{S_m}\left[P_m + P_b + \frac{P_e}{2.77}\right]$$

Where:

$Z = 1.15\ [1 + 0.013\ (D-4)]$ for SMAW

$= 1.30\ [1 + 0.010\ (D-4)]$ for SAW

P_m = primary longitudinal membrane stress (P*R/(2t)), ksi
P_b = primary bending stress (D/(2I)*Resultant moment at weld), ksi
S_m = Allowable design stress intensity
P_e = expansion stress resulting from restraint of free end displacement, ksi
D = nominal outside diameter of the pipe, in.
d = nominal inside diameter of the pipe, in.
P = operating pressure, ksi
R = nominal outside radius of the pipe, in.
I = moment of inertia $(\pi/64*(D^4-d^4))$, in.^{4}
t = nominal thickness, in.

P_b includes bending stresses due to dead weight plus operating basis earthquake (OBE) loads for normal/upset conditions and dead weight plus design basis earthquake (DBE) loads for emergency and faulted condition.

P_e includes bending stresses due to plant heat-up.

Substituting in the above equations yields a stress ratio of 0.89 for normal/upset conditions and 1.14 for emergency/faulted conditions for an SAW weld (SAW results in worst case Z). Note that primary bending stress for the emergency/faulted condition is conservatively assumed to be twice the stress for the normal/upset condition. For the observed indication, the length to pipe circumference ratio is less than 0.1. For these parameters, the allowable depth is 75% of the pipe wall. The actual depth from the pipe outside surface is 0.7425/1.125 = 66%.

The calculation details are provided in the project files.

4.2 Fatigue Crack Growth

Fatigue crack growth for two additional cycles was done on indication using pc-CRACK software [8], and TS-2 software [9]. TS-2 calculates the thermal stress at the local section due to thermal transients. Table 1 shows the transients considered for the fatigue crack growth analysis. Bending moments for pressure and dead weight (DW) during heat-up were extracted at the location of the indication per Reference 5. Using the bending moments for this location, bending stresses due to internal pressure and DW were calculated as follows:

Bending Stress due to pressure + DW = $\left|\sqrt{My^2 + Mz^2}\right|$ = 6.596 ksi

Since Pressure = 2250 psi at end of the heat-up transient

Hoop stress = $Pr_{mean}/2t$ = 5.812 ksi
Bending stress due to DW = 6.596 – 5.812 = 0.784 ksi

Hoop stresses during all other transients listed in Table 1 will be factored based on the operating pressure at each transient [7] and the above hoop stress calculation. Bending stress due to DW for all other transients listed in Table 1 will remain the same.

The axial stress distribution from OD to ID of the nozzle safe end at the indication location for several transient conditions was calculated using the TS-2 software [9]. The transients discussed below are the only significant transients with respect to this analysis and were used as the basis for calculating the stress response of all transients.

1. Heat-up: This transient was used to establish the baseline steady state case.
2. Loss of Secondary Pressure: Additional load case due to rapid temperature change. This load case was included in addition to the factored steady state load case during Loss of Secondary Pressure Transient stated above. The corresponding pressure stress variation was also included.
3. Reactor Trip/Loss of Reactor Coolant/Loss of Turbine Generator Load: Additional load cases due to rapid temperature change. These load cases were included in addition to the factored steady state load case during each transient stated above.
4. The steady states for all other thermal transient stresses were determined by applying a factor to the steady state of the heat-up transient.

All transients with the appropriate scaling factor at the beginning and the end of its transient state described above are shown in Table 2.

pc-CRACK software using ASME Code, Section XI elliptical surface crack in infinite plate model and the transient load cases described above as input calculated the fatigue crack growth for the next two operating cycles. Supporting calculations are contained in the project files.

5 RESULTS OF ANALYSIS

The observed indication was 66% of the wall thickness at the time of the 1994 UT examination and the ASME Code allowable flaw depth is 75% of the wall thickness. The crack depth, including fatigue crack growth for the period from 1994 to 2005 is 0.7428 inches. The crack depth, including fatigue crack growth, for the next two operating cycles, through April 2010 is 0.7429 inches. This indicates that there is 9% of wall available to accommodate any potential future crack growth after two more operating cycles. The growth of the indication length is on the same order as the crack depth.

As discussed above, the indication is believed to be associated with original fabrication and is likely subsurface. It is therefore not a serviced induced indication. Potential crack growth mechanisms discussed above indicates that the potential for crack growth is only due to fatigue. SI recommends that UT inspection of this weld be performed at the next outage or the next possible opportunity to characterize the indication more fully. If the indication is shown not to be surface connected to the outside surface, significant additional time could likely be demonstrated.

6 CONCLUSIONS AND DISCUSSIONS

Since the end-of-evaluation period flaw depth is well above that for the actual end-of-evaluation period indication, the required ASME Code, Section XI safety factors (2.77 for normal and upset, and 1.39 for emergency and faulted) are maintained throughout at least the next two operating cycles. Based on these results, it is concluded that operation for at least the next two operating cycles is justified with the observed indication left as-is.

Table 1: Plant Transient Condition [7]

Plant Transient Condition	40 Years Cycle Count
Heatup	500
Cooldown	500
Loading	15,000
Unloading	15,000
Step Load Increase	2000
Step Load decrease	2000
Reactor Trip	400
Hydrostatic Test	10
Leak Test	320
Normal Plant Variation	1000000
Loss of Reactor Coolant System	40
Loss of Turbine Generator	40
Loss of Secondary Pressure	5

Table 2: Bending Stresses due to Pressure and Gravity*

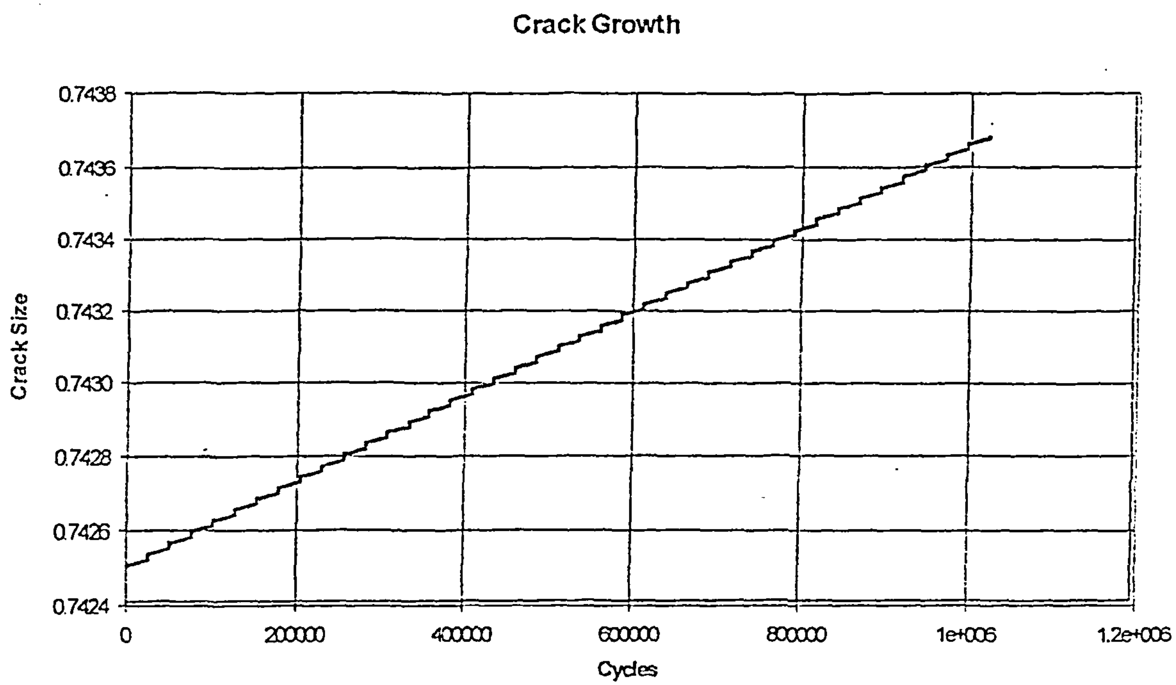
Plant Transient Condition	Load Case ID	Scale Factor at Beginning of the Transient	Scale Factor at End of the Transient	Reference Transient per Section 4.2
Heat-up	DW	1.000	1.0	Transient 1
	Pressure	1.000	N/A	Transient 1
	Steady State	0.8876	N/A	Transient 1
Cooldown	DW	1.000	1.0	Transient 1
	Pressure	1.000	N/A	Transient 1
	Steady State	0.8876	N/A	Transient 1
Loading	DW	1.000	1.000	Transient 1
	Pressure	0.666	1.044	Transient 1
	Steady State	0.8876	1.000	Transient 1
Unloading	DW	1.000	1.000	Transient 1
	Pressure	1.013	1.000	Transient 1
	Steady State	1.000	0.8876	Transient 1
Step Load Increase	DW	1.000	1.000	Transient 1
	Pressure	1.000	1.042	Transient 1
	Steady State	0.9831	1.000	Transient 1
Step Load decrease	DW	1.000	1.000	Transient 1
	Pressure	1.000	0.977	Transient 1
	Steady State	1.000	0.9850	Transient 1

Table 2 (Continued)

Plant Transient Condition	Load Case ID	Scale Factor at Beginning of the Transient	Scale Factor at End of the Transient	Reference Transient per Section 4.2
Reactor Trip	DW	1.000	1.000	Transient 1
	Pressure	1.000	0.800	Transient 1
	Steady State	1.000	1.000	Transient 1
	Trip	N/A	1.000	Transient 3
Hydrostatic Test	DW	1.000	1.000	Transient 1
	Pressure	N/A	1.389	Transient 1
	Steady State	0.617	0.617	Transient 1
Leak Test	DW	1.000	1.000	Transient 1
	Pressure	N/A	1.000	Transient 1
	Steady State	0.050	0.617	Transient 1
Normal Plant Variation	DW	1.000	1.000	Transient 1
	Pressure	1.000	1.089	Transient 1
	Steady State	1.000	1.000	Transient 1
Loss of Reactor Coolant System	DW	1.000	1.000	Transient 1
	Pressure	1.070	0.7640	Transient 1
	Steady State	1.000	1.000	Transient 1
	Trip	N/A	1.0	Transient 3
Loss of Turbine Generator	DW	1.000	1.000	Transient 1
	Pressure	1.070	0.7640	Transient 1
	Steady State	1.000	1.000	Transient 1
	Trip	N/A	1.0	Transient 3
Loss of Secondary Pressure	DW	1.000	1.000	Transient 1
	Pressure	1.000	0.090	Transient 1
	Steady State	0.850	0.852	Transient 1
	LOP	N/A	1.000	Transient 2

* Scale factor determined using transient pressure and temperature compared to one of the base transients.

Figure 1: Fatigue Crack Growth



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