OCT 23 2013

FPL.

L-2013-296
10 CFR 50.55a

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555-0001

Re: Turkey Point Unit 3
Docket Nos. 50-250
Relief Request No. 13

Florida Power & Light Company (FPL) has identified a through-wall flaw on the bonnet of the manual suction isolation valve (3-844A) to the Turkey Point Unit 3 Containment Spray Pump "A." Pursuant to 10 CFR 50.55a(a)(3)(ii), FPL requests relief from the applicable code requirements to delay the repair/replacement activity until the next scheduled refueling outage or outage of sufficient duration requiring entry into Mode 5.

The relief is requested on the basis that hardship or unusual difficulty exists without compensating increase in level of quality and safety. As discussed in the attached Relief Request, the use of the proposed alternative provides reasonable assurance of structural integrity of the subject valve.

This relief request contains a new commitment listed in Page 2 of this letter.

If you have any questions or require additional information, please contact Robert Tomonto, Licensing Manager, at (305) 246-7327.

Very truly yours,

Michael Kiley
Site Vice President
Turkey Point Nuclear Plant

Enclosure
Attachments

cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, USNRC, Turkey Point Plant

A047
NRR

New Commitment:

The projected final flaw size at time of repair is projected to be the present 5/16 inch x 1/16 inch measured. If the monthly measurement increases from the present by 1/16 inch in either direction (allowing 1/16 inch for measurement uncertainty), then the growth rate will be re-examined to verify the structural analysis conclusions and predicted growth rate.

L-2013-296

Enclosure

TURKEY POINT UNIT 3

RELIEF REQUEST No. 13

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Proposed Alternative In Accordance with 10 CFR 50.55a(a)(3)(ii)

Hardship or Unusual Difficulty Without Compensating
Increase in Level of Quality and Safety

1.0 ASME Code Component(s) Affected

The component associated with the relief request is valve 3-844A, which is the Turkey Point Unit 3 suction isolation valve for the "A" Containment Spray Pump (3P214A) from the common containment spray pump suction supply header. The valve passively maintains the Containment Spray System suction piping integrity and provides maintenance isolation for the "A" Containment Spray Pump. The valve is normally maintained in the back-seated, locked-open position during Modes 1-4.

Valve 3-844A is a Quality Group B, ASME Class 2, an 8 inch manually-operated gate valve manufactured by Anchor-Darling, with Series 150 welding ends, and constructed of ASTM A-351 grade CF-8 cast stainless steel material. The valve and pump are located outside containment in the Auxiliary Building Unit 3 Containment Spray Pump room.

2.0 Applicable Code Edition

The Code of Record for the Turkey Point Unit 3 Fourth 10-year inservice inspection interval is the 1998 Edition through 2000 Addenda of the American Society of Mechanical Engineers (ASME) Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components" subject to the limitations and modifications in 10 CFR 50.55a(b).

3.0 Applicable Code Requirements

ASME Section XI Code, subsection IWC, "Requirements for Class 2 Components of Light-Water Cooled Power Plants," subparagraph IWC-3122.2, "Acceptance by Repair/Replacement Activity." ASME Section XI, Subsection IWC, subparagraph IWC-3122.2 states in part that a component with flaws that exceed the acceptance standards of Table IWC-3410-1, is unacceptable for continued service until the component is corrected by a repair/replacement activity.

4.0 Reason for Request

Florida Power & Light Company (FPL) has identified a through-wall flaw on the bonnet of valve 3-844A that exceeds the acceptance criteria of Table IWC-3410-1, Acceptance Standards. While subparagraph IWC-3122.3 allows for acceptance by analytical evaluation per IWC-3600, it does not provide acceptance criteria for austenitic components. IWC-3600 states that the criteria of IWB-3600 may be applied, wherein Subarticle IWB-3640, "Evaluation Procedures and Acceptance Criteria for Austenitic Piping," it is stated that the evaluation procedures and acceptance criteria shall be the responsibility of the Owner and shall be subject to approval of the regulatory authority.

NRC Inspection Manual 9900: Technical Guidance, "Operability Determinations & Functionality Assessments for Resolution of Degraded or Nonconforming Conditions Adverse to Quality of Safety" (dated April 16, 2008), Appendix C, "Specific Operability Issues," Item C.11, "Flaw Evaluation," addresses evaluations of ASME Class 2 and Class 3 system components with through-wall flaws. When ASME Class 2 or Class 3 components do not meet ASME Code acceptance standards, the requirements of a NRC-endorsed ASME Code Case, or NRC approved alternative, then a determination of whether the degraded or nonconforming condition results in a Technical Specification required system, structure, or component being inoperable, is required. This section of the manual also states that whenever a flaw does not meet ASME Code or construction code acceptance standards or the requirements of an NRC endorsed ASME code case, a relief request needs to be submitted in a timely manner after completing the operability determination process documentation.

The through-wall flaw is located on the bonnet near the top of the packing gland of valve 3-844A. This ASME Class 2 valve is in the 8 inch suction line to the "A" Containment Spray Pump. Valve 3-844A can not be isolated from the upstream common supply header of the Containment Spray System without taking both trains of the Containment Spray System out of service. With both trains inoperable, Technical Specifications (TS) Limiting Condition for Operation 3.6.2.1, Action b., would require to restore at least one train to operable status within 1 hour, or to be in at least Hot Standby within the next 6 hours. The Prompt Operability Determination (POD) evaluation concluded that the valve continues to be capable of performing its required safety functions and is not susceptible to sudden or catastrophic failure. Performing a Code repair/replacement activity now to correct the flaw would create a hardship based on the potential risks associated with unit shutdown, thermal stress cycling of plant components, and emergent equipment issues incurred during shutdown and startup evolutions, with no compensating increase in the level of quality and safety gained by immediate repair of the flaw.

Accordingly, FPL requests relief from the applicable code requirements to delay the repair/replacement activity until the next scheduled refueling outage or outage of sufficient duration requiring entry into Mode 5.

5.0 Proposed Alternative and Basis for Use

The 10 CFR Section 50.55a(g)(4) specifies that ASME Code Class 1, 2, and 3 components must meet the requirements except for the design and access provisions and the pre-service examination requirements, set forth in the ASME Code Section XI to the extent practical with the limitation of design, geometry and materials of construction of the components.

Paragraph 50.55a(a)(3) of 10 CFR Part 50 states in part that alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the NRC if the licensee demonstrates (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

FPL is requesting authorization of an alternative to the requirements of the ASME Code Section XI, IWC-3122.2 pursuant to 10 CFR 50.55a(a)(3)(ii).

FPL proposes to temporarily accept the as-found condition (i.e. through-wall flaw) to allow continued service operation until the next scheduled refueling outage or outage of sufficient duration requiring entry into Mode 5, instead of performing immediate flaw correction by a repair/replacement activity described in Code subparagraph IWC-3122.2, "Acceptance by Repair/Replacement Activity". Performing a Code repair/replacement activity to correct the flaw would create a hardship based on the potential risks associated with unit shutdown, thermal stress cycling of plant components, and emergent equipment issues incurred during shutdown and startup evolutions, with no compensating increase in the level of quality and safety gained by immediate repair of the flaw.

Although the provisions of Code Case N-513-3, "Evaluation Criteria for Temporary Acceptance of Flaws in Class 2 & 3 Piping, Section XI, Division I," do not apply directly to valves, the guidance of this Code Case was followed since it provides criteria for analytical evaluation, and rules for temporary acceptance of flaws in piping.

Specifically, Code Case N-513-3 Procedure (paragraph 2.0) Methodology was applied:

(a) Flaw size characterization is by visual examination and direct measurement: 5/16 inch in length and 1/16 inch in width.

(b) Flaw is characterized as through-wall coplanar flaw.

(c) The flaw is a single flaw.

(d) Flaw evaluation was performed and is attached to the POD (Attachment 2).

(e) FPL will perform a monthly PT examination of the area of manual valve 3-844A with the identified through-wall flaw to validate the analysis supporting the POD.

(f) FPL will perform a daily visual walkdown of manual valve 3-844A to confirm analysis from PT examinations remains valid, i.e. no new significant leakage.

(g) FPL will repair or replace manual valve 3-844A if the predicted flaw size from either periodic inspection or by flaw growth analysis exceeds the acceptance criteria (allowable and critical flaw lengths) of 5.63 inches and 23.4 inches in the circumferential and axial directions, respectively.

    The structural analysis performed predicts negligible flaw growth for the remainder of Unit 3's present operating cycle (Cycle 27). The next outage is currently scheduled for March 2014. Therefore, the projected final flaw size at time of repair is projected to be the present 5/16 inch x 1/16 inch measured. If the monthly measurement increases from the present by 1/16 inch in either direction (allowing 1/16 inch for measurement uncertainty), then the growth rate will be re-examined to verify the structural analysis conclusions and predicted growth rate.

(h) FPL will repair or replace manual valve 3-844A during the next scheduled Turkey Point Unit 3 Refueling Outage currently planned to begin in March 2014, or forced outage of sufficient duration requiring entry into Mode 5, whichever occurs earlier.

<u>Augmented Inspection</u>

FPL performed an extent of condition visual examination (method that found original flaw) at five of the most susceptible and accessible locations. The five locations examined were manual valves. Three susceptible locations are identical valves in the same service. The remaining two valves examined are 6 inch stainless steel manual valves in similar service (same fluid and service conditions). The inspections did not identify any evidence of unacceptable defects (no relevant indications, leakage or Dry Boric Acid) at the valves examined.

Flaw Evaluation

The flaw evaluation is documented in the attached POD (Attachment 2), which is the basis for considering the valve as operable, but degraded/non-conforming with compensatory measures.

The flaw was conservatively evaluated using a linear elastic fracture mechanics evaluation using the methods and acceptance criteria of ASME Section XI non-mandatory Appendix H, "Evaluation of Flaws in Ferritic Piping." Applied loads for pressure, deadweight, thermal and seismic conditions were included. Overall, these loads are relatively minor due to the low pressure and temperature conditions, the plant configuration location adjacent to the pump suction nozzle anchor point, and the low seismic accelerations. Stress intensity factors and safety factors were applied per ASME Section XI, Appendix H. The structural analysis evaluation calculated an allowable critical flaw size of 5.63 inch and 23.4 inch in the circumferential and axial directions, respectively. The measured flaw length is 5/16 inch circumferentially. Since the measured flaw size is significantly less than the allowable critical flaw sizes, there is substantial margin which ensures that structural integrity is maintained.

Additionally, a flaw growth evaluation was performed that considered potential effects of environmentally-assisted cracking, limited number of operating cycles and low resultant stresses and it concluded that flaw growth is not expected for the valve during the remainder of the current operating cycle. As such, valve 3-844A will retain its structural integrity until the flaw is removed via repair/replacement activity during the next refueling outage or forced outage of sufficient duration requiring entry into Mode 5.

Based on the flaw evaluation, it was determined that the through-wall flaw, is stable and the valve will not fail catastrophically under design loading or accident conditions.

6.0 Duration of Proposed Alternative

The requested temporary Code relief will apply until Code repair/replacement activities are performed on the valve body either during the next scheduled refueling outage (or forced outage of sufficient duration requiring entry into Mode 5) or when the predicted flaw size exceeds acceptance criteria. The next scheduled Turkey Point Unit 3 refueling outage is planned to begin in March 2014.

7.0 Precedent

This relief request is similar to the relief granted McGuire Nuclear Station, Unit 1, March 26, 2008, Accession Number ML080580577, which involved a through-wall flaw in an austenitic stainless steel ASME Class 2 valve, evaluated using the guidance of Code Case N-513-3.

8.0 Attachments

1. Partial drawing and picture of valve 3-844A and through-wall flaw.
2. Condition Report AR1904263, Prompt Operability Determination.

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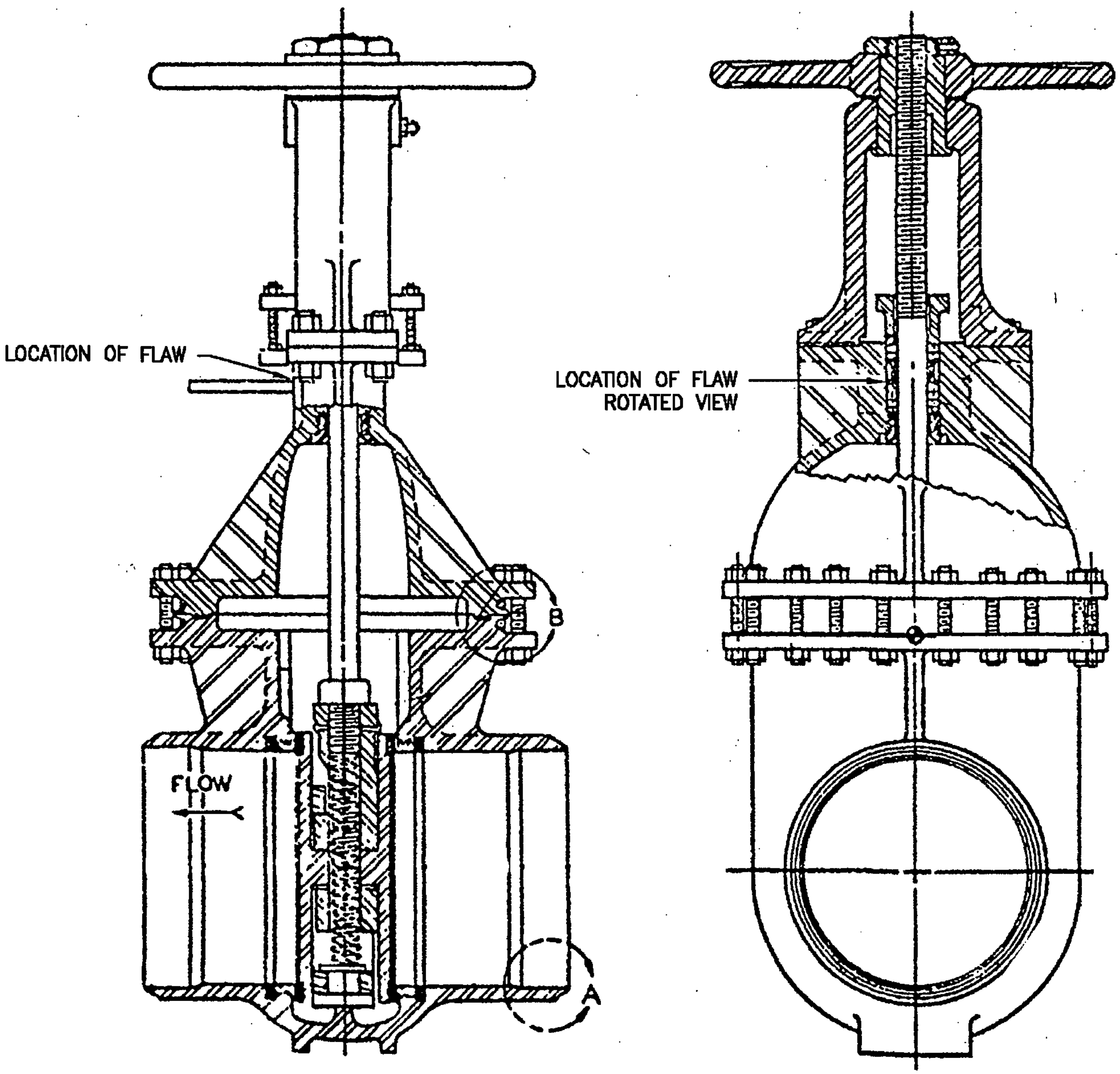
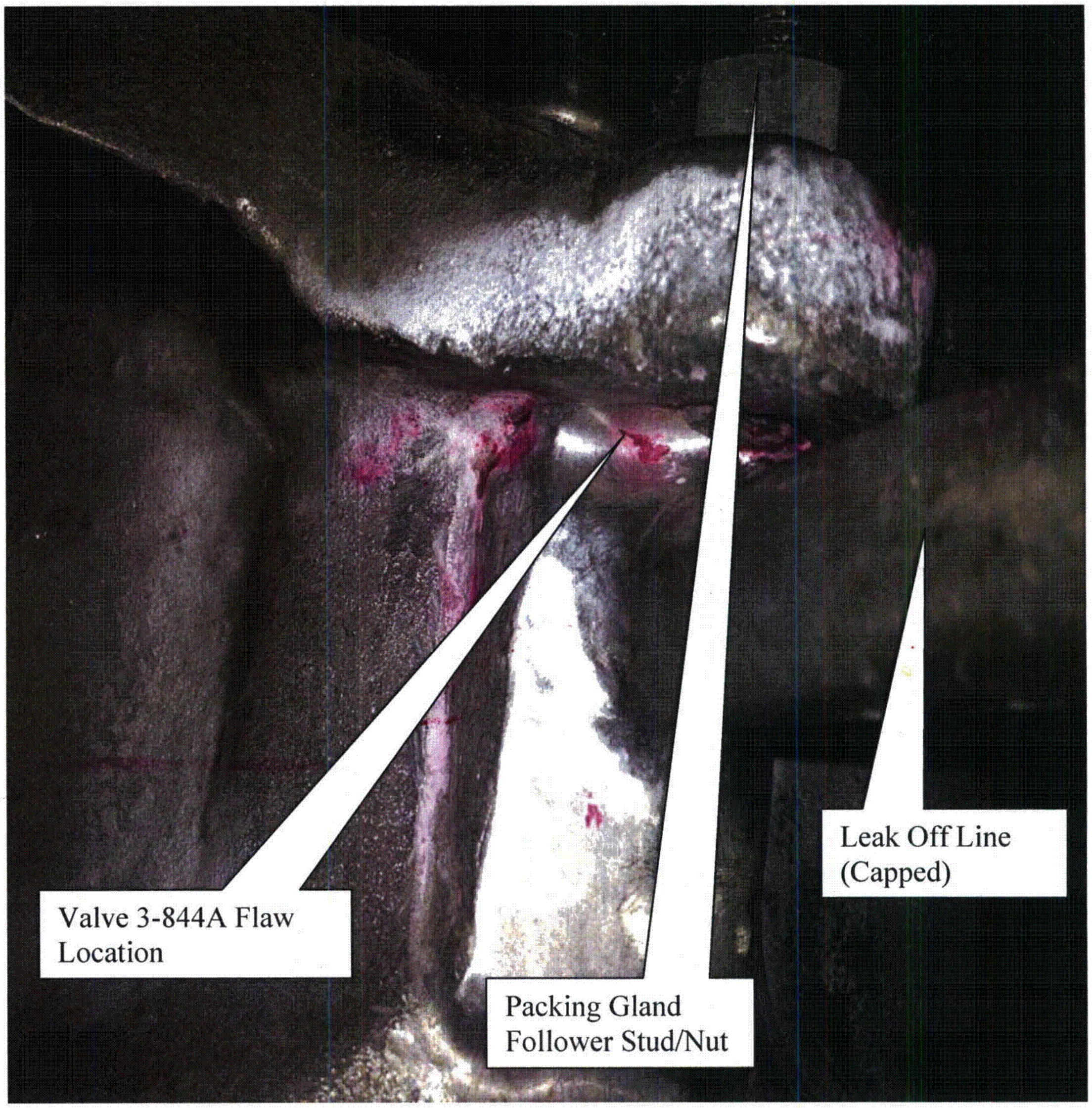
Attachment 1

ATTACHMENT 1
PARTIAL DRAWING OF VALVE 3-844A AND THROUGH-WALL FLAW





Valve 3-844A (Partial Picture)

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Attachment 2

AR: 01904263 **AR Assignment Number: 03**

CR Title: FLAW IN VALVE 3-844A BONNET

NOTE: To ensure a complete POD, each of the following items shall be addressed to a level of detail commensurate with the affected SSC safety significance. Use instructions in EN-AA-203-1001 section 4.4 and Attachment 5 to complete this form.

1. Describe affected SSCs (System #/ Comp #, etc.), considering the extent of the condition:

   System: Containment Spray / System # 068
   Affected Components: 3-844A, CTMT SPRAY PMP A SUCT ISO VLV
   Safety Classification: Safety Related

2. Describe degraded or nonconforming condition:

   A thru-wall leak was discovered in the area of the packing leak off line in valve 3-844A upper bonnet. As part of the Evaluation for AR 01904263, a Prompt Operability Determination (POD) was requested to determine the operability of the 3-844A valve, along with the "A" train. Since this valve is the first valve off the common supply header, both the "A" train and "B" train of the Containment Spray System are affected in regards to pressure boundary and structural integrity.

3. Identify Current Licensing Basis function(s) and performance requirements, including Technical Specifications, FSAR, NRC Commitments, or other appropriate information:

   UFSAR Design Basis

   As documented in Section 6.4.1 of the UFSAR , the Containment Spray System is designed to:

   > The primary purpose of the Containment Spray System is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident. Operation of the Containment Spray System and the Emergency Containment Cooling System will ensure that containment pressure does not exceed its design value which is 55 psig at 283 °F (100% R.H.).

   Table 6.2-12 of the UFSAR lists the maximum potential leak rates for various components in the ECCS recirculation loop. The total maximum potential leakage resulting from all sources is listed in Table 6.2-12 as 2325 cc/hr.

   Technical Specifications (U3 Amendment #257):

   Technical Specification (TS) 3/4.6.2.1 states that "Two independent Containment Spray Systems shall be OPERABLE with each Spray System capable of taking suction from the

RWST and manually transferring suction to the containment sump via the RHR System". Action 'a' of this section states that "With one Containment Spray System inoperable restore the inoperable Spray System to OPERABLE status within 72 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

Action 'b' of this section states that "With two Containment Spray Systems inoperable restore at least one Spray System to OPERABLE status within 1 hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours. Restore both Spray Systems to OPERABLE status within 72 hours of initial loss or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours."

10CFR50.55a and Tech Spec 4.0.5a requires that structural integrity must be made in conformance with ASME Code Section XI. Per RIS 2005-20 and NRC Inspection Manual Part 9900, structural integrity of Class 2 structures, systems & components shall conform to the requirements of the original construction code (USAS B31.1-1955), ASME Section XI, or an NRC-endorsed code case or NRC approved alternative, or a relief request needs to be submitted in a timely manner after completing the operability determination process documentation.

Design Basis Document: (5610-068-DB-002)

DBD Pge 20-21:
FSAR Section 6.2 includes description of the following intent for valve design:
"All valves, except those which provide a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter (some specs say pipe diameter), assuming no credit taken for valve packing. [MOVs] that are normally operated and in containment are not backseated. Other normally open valves are backseated. Normally closed globe valves are installed with recirculation flow under the seat to prevent leakage of recirculated water through the valve stem packing. Relief valves are totally enclosed. Control and motor operated valves, 2.5" and above, which are exposed to recirculation flow, are provided with double-packed stuffing boxes and stem leakoff connections which are piped to the Waste Disposal System." FSAR Section 6.2 also provides generally acceptable valve leakage rates at 2 or 3 cc/hr/in of nominal pipe diameter. This criteria was established for the equipment specification and hydrostatic test requirements, and was not related explicitly to any CSS functional requirements such as containment leakage isolation.

4. Identify the established minimum design basis values necessary to satisfy the SSC specified safety function(s):

Valve 3-844A shall passively maintain the CS system pressure boundary integrity.

The safety functions affected by the identified nonconforming/degraded condition are:

a) Structural integrity of the valve to maintain the CS system pressure boundary integrity.
b) Leak tightness of the post-accident containment sump recirculation flow path components outside containment

5. Evaluate effects of condition, including potential failure modes, on the ability of the SSC to perform its specified safety function(s) and support function(s), if any. The following items shall be covered in the Evaluation:

A. Identify the Mode or other specified conditions of Operability when the specified TS function(s) for the affected SSCs are required;

TS 3.6.2.1 requires that the CS System be operable in Modes 1, 2, 3 and 4.

B. Identify assumptions used;
1. During the last performance of 3-OSP-202.4 (3/1/12), Recirculation Piping Verification, there was no indication of leakage at 3-844A. During this test the Containment Spray suction piping is pressurized, with 3-844A open and backseated, to 120-140 psig. Valves that are provided with backseats (including 3-844A) are theoretically capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter. This equates to 1.25 cc/hr at the packing area of 3-844A. If the 3-844A valve were to be closed during a LOCA event, the leakage would be seat leakage theoretically at the UFSAR described 3 cc/hr/in of valve size (8") Therefore, leakage from the thru wall defect is estimated at 24 cc/hr based on the valve being either back seated or closed based on leakage discussed in the UFSAR.
2. Drawing 5610-M-600-20, Rev.3 latest in NAMS, indicates the bore of the packing area in the region of the flaw is 2" and field caliper measurements indicate an OD in this area of 3-7/8". Therefore the wall thickness in the region of the flaw is assumed to be 0.937".
3. The 3-844A bonnet flaw is hydraulically equivalent to an orifice with a similar flow rate.

C. Discuss why the degraded or nonconforming condition does or does not prevent the SSC from performing its specified safety function(s) or support function(s). (Include known information that supports the specific evaluation, any adverse impact about the condition, or related analysis);

Leak Tightness:

FSAR Section 6.2 includes description of the following intent for valve design:
"All valves, except those which provide a control function, are provided with backseats which are capable of limiting leakage to less than 1.0 cc per hour per inch of stem diameter (some specs say pipe diameter), assuming no credit taken for valve packing. The stem of the valve 3-844A is 1-1/4" diameter and this valve is normally on its backseat during standby and accident conditions. This equates to 1-1/4 cc/hr leakage. The leakage thru the defect in the packing area can not be measured directly and just barely shows an indictation of leakage after 24 hours observation, therefore it is conservatively estimated at 3cc/hr at atmospheric pressure.

UFSAR Chapter 6.2.3 describes that for the recirculation phase of a LOCA the reactor coolant water which eventually drains to the containment recirculation sump is recirculated through the sump line from the containment to the suction of the RHR pump through two independent and redundant recirculation lines.

Table 6.2-12 summarizes the maximum expected leak rate from components in the containment sump recirculation loop that could potentially be a leak source. A potential leak rate of 10 drops per minute was assumed for each flange connection in the recirculation even though each flange would be adjusted to essentially zero leakage during plant operation. The total maximum potential leakage resulting from all sources is listed in Table 6.2-12 as 2325 cc/hr.

The leakage of fluid from post-accident containment sump recirculation loop components outside containment would constitute a release of (unfiltered) radioactive iodine to the outside environment. This radioactive iodine release would primarily result in increased thyroid doses for personnel located in the control room, the site exclusion boundary, and the low population zone. The new (post EPU) UFSAR LOCA dose analysis post EPU assumes 4650 cc/hr leakage (2325 cc/hr X 2) from the components in the post-accident containment sump recirculation loop outside containment. The minimum established values necessary to satisfy the specified safety function are those regulatory dose acceptance criteria in 10 CFR 100 and 10 CFR 50, Appendix A.

Note that Table 6.2-12 of the UFSAR lists the maximum potential leak rates for various components in the ECCS recirculation loop. The sum total of the tabulated leakage rate to atmosphere from these sources is 2325 cc/hr. This total leakage value has historically been used as the operability limit for ECCS recirculation loop leakage and has been institutionalized in many plant procedures and engineering evaluations over the years as it was presumed to be an integral assumption of the UFSAR safety analysis. Since the current UFSAR Table 6.2-12 ECCS recirculation loop leakage value of 2325 cc/hr has been determined to be half of the UFSAR accident analysis assumption, it now provides an upper limit for ECCS operability. Based on the current UFSAR LOCA analysis, the ECCS would be considered "not fully qualified" (i.e., not meeting all aspects of the current licensing basis) if it has any measurable external leakage greater than 2325 cc/hr. Continued operability under these conditions is based on the provisions of NRC RIS 2005-20, such that the new upper leakage limit for operability can be tied to the regulatory dose acceptance criteria.

During PT3-26 Operations and Engineering performed, 3-OSP-202.4, Recirculation piping verification. The main purpose of this procedure is to verify that no leakage exists in any of the piping/components of the RHR, SI and CS systems. During this test the Containment Spray suction piping, including the cyclone separator loop and seal water heat exchanger loop is pressurized to 120-140 psig. This test gives reasonable assurance that no through-wall leak exists in any of the components that are used during post-LOCA recirculation. During the last performance of this procedure, with the valve in the open position, there was no indication that the leak on the 3-844A valve, if present at the time of the test, was affecting the ability of the system to maintain the pressure boundary. The test was completed satisfactorily.

As indicated in the response above, the minimum established values for recirculation loop leakage are established under UFSAR as 2325 cc/hr for "full qualification" and 4650 cc/hr for "operable but degraded" classification. However, the absolute limit necessary to satisfy the leak tightness safety function are those regulatory dose acceptance criteria in 10 CFR 100 and 10 CFR 50, Appendix A. As such, they must meet the guidelines of NRC Regulatory Guide 1.183 and 10CFR 50.47. This means that the dose criteria for control room personnel following the accident is 5 Rem TEDE. In keeping with this, the dose contribution for control room personnel is actually 3.64 Rem total based on UFSAR Table 14.3.5-6.

A review of UFSAR Table 14.3.5-6 also indicates that the new 30-day control room (CR) dose post-EPU is limiting with respect to additional radioiodine release. The latest CR dose contribution reported in the UFSAR is 0.5 Rem TEDE based on ECCS recirculation loop leakage outside containment dose to the CR operators.

As mentioned above, the dose contribution from 4650 cc/hr of ECCS leakage during post-accident containment sump recirculation is presented in UFSAR Table 14.3.5-6 as 0.5 Rem TEDE. The total CR dose is 3.64 Rem. The amount of leakage that could be tolerated without exceeding the 5 Rem TEDE regulatory limit can be determined using the following relationship (where margin is 5-3.64=1.36 Rem):

$$\frac{0.5 \cdot \mathrm{rem}_{\,(30\mathrm{day})}}{4650 \cdot \dfrac{\mathrm{cc}}{\mathrm{hr}}} \equiv \frac{1.36 \cdot \mathrm{rem}_{\,\mathrm{dose\,margin}(30\mathrm{day})}}{X \dfrac{\mathrm{cc}}{\mathrm{hr}}}$$

The resulting leak rate is 12,645 cc/hr. For the purpose of this operability determination, the maximum ECCS post-LOCA recirculation loop leakage (all sources) that can be tolerated will be maintained at the UFSAR limit of 2325 cc/hr (even though the above discussion shows that a much greater leakage value can be accepted). Although the leakage is not active at RWST suction pressure conditions (17 psig), a visible indication of weepage was detected. During Post-LOCA alignment, the pressure at the referenced valve will be close to RHR suction pressure (44 psig from the Recirculation Sumps) plus RHR discharge pressure(=180 psig as referenced in the DBD). Compensating for ECCS Recirculation conditions at the referenced valve utilizing standard orifice scaling, the expected value is 78.1 cc/hr at 180 psig CS Pump suction pressure. This makes the new ECCS Recirculation Loop Leakage total 1356.83cc/hr. Subtracting the known ECCS recirculation loop leakage currently being tracked for Unit 3 (1278.35 cc/hr) gives a maximum allowable leak rate margin for ECCS Systems (including 3-844A bonnet) of 968.17 cc/hr. This leak rate would apply to RHR/CS system operation during post-LOCA operating conditions (44 psig and 283 °F in containment).

At a leak rate of 1356.83 cc/hr, the total amount of leakage over the course of 30 days would be approximately 258 gallons. This diversion of flow is negligible compared to the volume of water in the containment sump and it would not result in a loss of net positive suction head for the operating RHR or CS pumps.

This volume of diverted containment sump fluid is also considered insignificant for flooding concerns assuming the 3-844A leakage migrates to the RHR pump rooms. The current limit for flooding affecting room safety equipment is 3,000 gallons (Reference 7).

The credited 78.1 cc/hr leakage is equivalent to .0004 gpm. The reduction in flow due to the current 3-844A bonnet leak will not prevent adequate delivery of ECCS cooling flow during normal or post-accident service conditions. Furthermore, the leakage is also below the 1 gpm unidentified and 10 gpm identified maximum allowed RCS leak rate. As such, it is not considered a significant contribution.

The above analysis demonstrates that the referenced leakage does not pose an operability/functionality challenge to the RHR or CS System.

Structural Integrity:

Per the NDE report (in EDMS of AR 19042630, the flaw (5/16" long x 1/16" wide) meets the ANSI B16.34-1981 Annex D (D2) acceptance standards for Castings based on the thickness in the area of the flaw. However, the flaw must be evaluated for structural integrity since 10CFR50.55a and Tech Spec 4.0.5a require that structural integrity must be in conformance with ASME Code Section XI.

In order to support structural integrity determination of the valve, fracture mechanics analyses were performed consistent with the ASME B&PV Code, Section XI [1 to SIA Report] to determine the maximum through-wall flaw sizes associated with structural stability. It is understood that Section XI (incl Code Case 513-2) does not specifically address valve through-wall flaws. The evaluation results (Attachment 1) are based on verified analyses which utilize many conservative assumptions. This evaluation addresses the structural integrity of the valve bonnet. Also included in Attachment 1 is a flaw growth evaluation for determination of the time for the flaw to grow to maximum allowable size. This time exceeds the time to the next Unit 3 outage.

The evaluation is performed using the procedures of IWC-3600 in Section XI of the ASME Code [1 of SIA Report] for the through-wall flaw modeling the bonnet region of the valve as straight pipe. The 3-844A valve is a Class 2 component design per ASA B31.I [ref.2 of Att.1]. Following Section XI flaw evaluation procedures, the indications are evaluated as two independent through-wall planar flaws, one in the axial direction and the other oriented in the circumferential direction. Performing separate flaw evaluations in each orthogonal direction will provide bounding results, regardless of flaw orientation.

Critical and allowable flaw sizes are determined using the linear elastic fracture mechanics (LEFM) approach as described in Reference [1 of SIA Report] for postulated through-wall axial and circumferential flaws in the valve bonnet to assess the structural integrity of the valve.

The ASME Section XI based allowable and critical (no safety margin) circumferential through-wall flaw lengths are both determined to be 5.63 inches (due to the fracture mechanics model applicability limits reached). Similarly, the allowable and critical axial through-wall flaw lengths are both determined to be 23.4 inches (also due to the fracture mechanics model applicability limits reached). The flaw growth analysis included in Attachment 1 indicates the flaw will remain acceptable for at least the remainder of the present refueling cycle.

Therefore, the existing flaw is determined to be structurally acceptable and the above limits are applicable to the periodic monitoring for maintenance of structural integrity (within the time evaluated for the flaw to increase to maximum allowable size).

D. Describe (for SSC not fully capable of performing its specified safety function(s)) compensatory actions (e.g., procedure changes, facility changes, or substitution of manual actions for automatic functions) taken to address the condition (compensatory actions must be reviewed under 10 CFR 50.59):

No compensatory actions are required. CS system is capable of performing its specified TS functions.

E. Evaluate continued operation should the degraded condition degrade further and describe the method used to monitor the degraded condition until corrected (e.g., operator rounds, system health trending/walkdowns, CAP monitoring action) or provide justification why monitoring is not required. (The POD must be forward looking to assess conditions that may impact the SSC during the period of operation until the condition is corrected, especially for PODs that rely on equipment performance information):

The performance of 3-OSP-202.4 during PT3-26 (performed 3/1/12) verified 3-844A locked open and backseated. Based on the SAT performance of this leak inspection, which included 3-844A, there is reasonable assurance that the identified leak on the 3-844A bonnet is not active (due to being isolated by the backseat in the valve's normal position) and that it is not adversely degrading. Even though this condition is not expected to degrade, periodic flaw monitoring (daily observation) per assignment AR 01904263-04 will identify any further degradation of the leak. A Assignment AR 01904263-05 is for determination of any extent of condition or follow up examinations required. An NRC relief request regarding the TS Section XI requirements in regard to this condition is tracked by AR 01904263-06. Assignment AR 01904263-07 will address the OBD determination in regard to repair/replacement of the degraded component.

F. Assess (for SSC not fully capable of performing its specified safety function(s)) the impact of relevant Engineering Changes (e.g., modifications) scheduled for implementation over the expected duration of the final corrective actions and applicable open operability and functionality issues (listed in Cognos Report AT-01-28 and their cumulative impact on this POD (including a review of any related compensatory actions in place as a result of an open issues):

There are currently three PODs that cumulatively impact the Unit 3 ECCS systems or this evaluation: AR 01902914 for the 3A RHR pump threaded gland seal leak, AR 1694007 identifies dry boric acid on the 3A CS Pump cyclone separator line, and AR 445213 identifies dry boric acid on the 3A RHR pump quench line threaded connection. However, these identified leakages are already accounted for under the 1356.83 cc/hr analyzed herein.

G. Conclusion:

Based on the above discussion there is reasonable assurance that the CS valve 3-844A is capable of performing its design function and that it will meet its 30 day post LOCA mission time.

6. References:

1) UFSAR, Chapter 06, latest electronic version.
2) Technical Specifications 3.6.2.1, latest electronic version.
3) 5610-068-DB-002, Design Basis document Containment Spray System
4) 5613-M-3068 Sht 1
5) AR 1694007
6) AR 1902914
7) AR 0460757

7. Attachments:

1) Structural Integrity Associates Report # Report No. 1301208.401.R0 Dated 9/19/13

8. MODE Restrictions (APPLICABILITY Restrictions for ISFSI Conditions):

N/A

9. Operability Recommendation

CHECK ONE	PROMPT OPERABILITY DETERMINATION
	Affected SSC should be considered operable, since it is fully qualified, meeting all CLB and design requirements.
	Affected SSC should be considered operable and above full qualification but with reduced margin below some design requirement. There is a high degree of confidence that the degraded SSC meets full qualification as described in the Current Licensing Basis.
	Affected SSC should be considered operable but non-conforming, and below Full Qualification. Continued Operability is based on the provisions of RIS 2005-20.
X	Affected SSC should be considered operable but degraded, and below full qualification. Continued operability is based on the provisions of RIS 2005-20.
	Affected SSC should be considered inoperable.

Is a past operability review required?	NO	If yes, ensure POR type AR assignment is initiated.

AR: 01904263 **AR Assignment Number: 03**

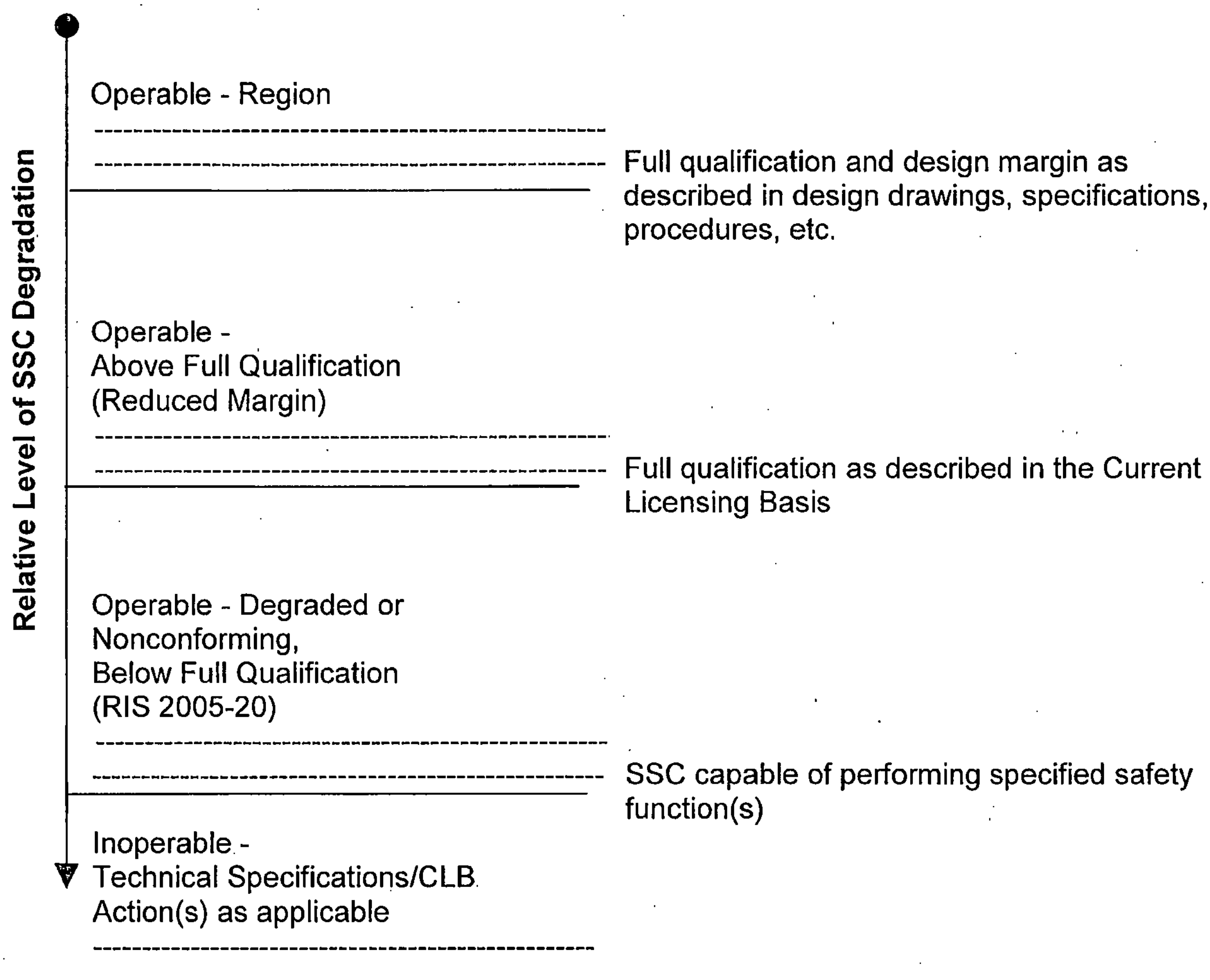
Prepared By: Marke Toner / Marke Toner Print/Sign Date 9/19/13

Reviewed By: C. Melchen / [illegible] Print/Sign Date: 9-19-13

SM Approval: Juergens E.K. Juergens Print/Sign Date/Time: 9/19/13 2200

After SM approval, preparer shall ensure the appropriate OBN, OBD, or ONOT type AR assignments are initiated to track final corrective actions, and COMP type AR assignments are initiated for compensatory actions, in accordance with EN-AA-203-1001 section 4.9. If operable and above full qualification but with reduced design margin, preparer shall initiate an RWT type AR assignment to the System Engineer as notification for potential System Health Report discussion.

10. Nuclear Station Operability Condition Model



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September 19, 2013
Report No. 1301208.401.R0
Quality Program: ☒ Nuclear ☐ Commercial

Mr. Philip R. Barnes
Florida Power & Light Company
Turkey Point 3 & 4 Nuclear Plant
9760 SW 344 Street
Florida City, Florida 33035

Subject: Evaluation of 3-844A Valve Bonnet Leak

References:

1. ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition through 2000 Addenda.
2. ASA B31.1-1955, Code for Pressure Piping.
3. Email attachment from Mark Toner (FPL) to Robert McGill (SI), "3-844A ISOs.pdf," 9/17/2013, SI File Number 1301208.201 (reproduced in Attachment 1).
4. Email attachment from Mark Toner (FPL) to Robert McGill (SI), "3-844A Stress1.pdf," 9/17/2013, SI File Number 1301208.201 (reproduced in Attachment 2).
5. Email attachment from Mark Toner (FPL) to Robert McGill (SI), "3-844A Seismic Design.pdf," 9/18/2013, SI File Number 1301208.201 (reproduced in Attachment 3).
6. NUREG-0800, Rev.4, March 2010, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants."
7. Email from Mark Toner (FPL) to Robert McGill (SI), "RE: Valve Body Leak – Design Input Request," 9/18/2013, SI File Number 1301208.201 (reproduced in Attachment 4).
8. **pc-CRACK™**, Version 4.0.1.0, Structural Integrity Associates, December 14, 2011.
9. NUREG/CR-4513, Rev. 1, May 1994, "Estimation of Fracture Toughness of Cast Stainless Steels During Thermal Aging in the LWR Systems."
10. Email attachment from Philip R. Barnes (FPL) to Robert McGill (SI), "5610-068-D8-002.pdf," 9/19/2013, SI File Number 1301208.201 (reproduced in Attachment 5).
11. Fatigue Crack Growth Computer Program, NASA/FLAGRO Version 2.0, National Aeronautics and Space Administration, April 1992.
12. Takahashi, Y., "Evaluation of leak-before-break assessment methodology for pipes with a circumferential through-wall crack. Part I: stress intensity factor and limit load solutions," International Journal of Pressure Vessels and Piping, Vol. 79, pp. 385-392, 2002.
13. Structural Integrity Report No. 1301208.402.R0, "Corrosion Evaluation of Valve 3-844A at Turkey Point Unit 3."

Dear Philip:

This summary report documents the flaw evaluation of the leaking 3-844A valve bonnet at the Turkey Point Unit 3 Nuclear Plant to determine the allowable through-wall flaws that would meet ASME B&PV Code, Section XI stability requirements. It is understood that Section XI does not specifically address through-wall flaws. The evaluation results summarized herein are based on verified analyses which utilize many conservative assumptions. This evaluation addresses the structural integrity of the valve bonnet and serves as a reference for the NDE inspection performed in parallel.

1.0 INTRODUCTION

A through-wall leak was recently discovered in the 3-844A valve bonnet of the Containment Spray System at the Turkey Point Unit 3 Nuclear Power Plant. NDE inspection performed on 9/17/13 after cleaning off the boric acid buildup (as shown in Figure 1) identified a through-wall flaw in the valve bonnet above the packing leakoff line. No observable leakage was present. The flaw appears to be due to an original casting defect based on its void like appearance. The specific location of the flaw is in the valve bonnet above the packing as shown in Figure 2. The flaw noted herein appears localized to the area of the bonnet just under the bonnet to yoke flange.

Since the valve cannot be isolated, the primary concern is that the valve be able to perform its function in the degraded condition until scheduled repair/replacement. Therefore, the objective of this calculation is to perform fracture mechanics analyses consistent with the ASME B&PV Code, Section XI [1] to determine the maximum through-wall flaw sizes associated with structural stability.

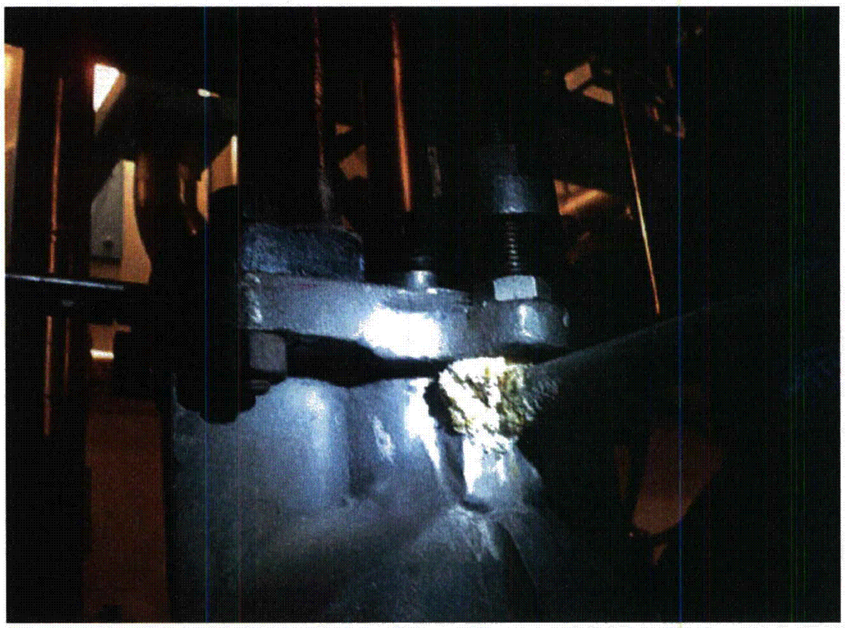


Figure 1: Boric Acid Buildup Identified on 3-844A Valve Bonnet

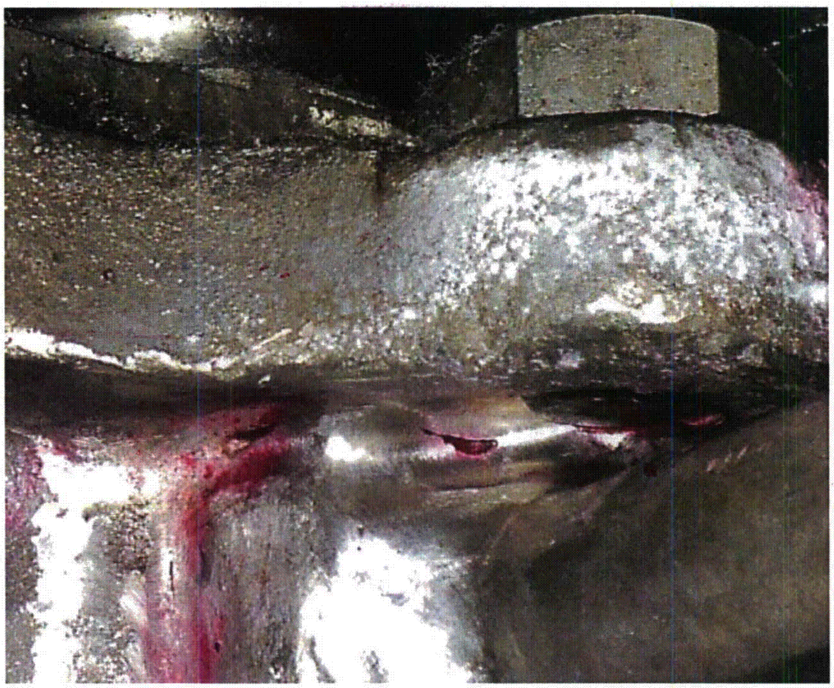


Figure 2: Photograph of Indication Locations at 3-844A Valve

2.0 TECHNICAL APPROACH

The evaluation is performed using the procedures in Section XI of the ASME Code [1] for the through-wall flaw modeling the bonnet region of the valve as straight pipe. The 3-844A valve is a Class 2 component design per ASA B31.l [2]. Following Section XI flaw evaluation procedures, the indication is evaluated as two independent through-wall planar flaws, one in the axial direction and the other oriented in the circumferential direction. Performing separate flaw evaluations in each orthogonal direction will provide bounding results, regardless of flaw orientation.

ASME Section XI Article IWC-3600 [1] does not provide acceptance criteria for austenitic components. For ferritic components, it is stated that the criteria of the IWB-3600 may be applied. In Subarticle IWB-3640, 'Evaluation Procedures and Acceptance Criteria for Austenitic Piping,' it is stated that the evaluation procedures and acceptance criteria shall be the responsibility of the Owner and shall be subject to approval of the regulatory authority.

For conservatism, a linear elastic fracture mechanics (LEFM) evaluation is used since this is conservative compared to a limit load or elastic plastic fracture mechanics approach. In addition, since acceptance criteria for austenitic piping using LEFM is not available in Reference 1, the acceptance criteria for ferritic piping in Appendix H are used.

Critical and allowable flaw sizes are determined using the Appendix H, LEFM approach as described in Reference 1 for postulated through-wall axial and circumferential flaws in the valve bonnet to assess the structural integrity of the valve.

3.0 FLAW EVALUATION

3.1 Component Dimensions

The 3-844A valve is connected to 8" piping [3]. The leak location is in the area of the bonnet just under the bonnet to yoke flange. Therefore, the following dimensions of the valve bonnet are used in this evaluation:

- Valve bonnet inside diameter: 2" [3]
- Valve bonnet outside diameter, near leak: 3-7/8" [7]
- Valve bonnet thickness, near leak: (3-7/8" - 2")/2 = 0.9375"

3.2 Design Conditions

The maximum operating conditions of the piping containing the 3-844A valve are listed below and used in this evaluation.

Maximum Operating Conditions:

- Pressure = 180 psig [10]
- Temperature = 205°F [3]

3.3 Materials and Material Properties

The material of the 3-844A valve bonnet is specified as ASTM A-351 Grade CF8 from Reference 3.

The material properties of the valve component are obtained from Reference 2 for the valve bonnet. The Modulus of Elasticity for A-351 Grade CF8 (18Cr-8Ni) E is 27,100 ksi at 205°F.

3.4 Applied Loads

The stress report [4] provided forces and moments for the 3P214A pump inlet nozzle (Node 540), which is the anchor point just downstream of the 3-844A valve. The stress report included Pressure (P), Deadweight (DW), Thermal, and SSE loading conditions [3]. The loads need to be transferred from the pump inlet nozzle to the valve bonnet.

Since the valve bonnet has a free end, any thermal expansion in the valve bonnet will be free of constraint. Consequently, stress due to thermal expansion at the yoke wheel is insignificant for the valve bonnet.

Similarly, due to the free end effect, moments from pump inlet nozzle will not be transferred to and do not affect the valve bonnet. Only the forces from the pump inlet nozzle that resolve in the axial direction of the valve may induce axial stress in the valve bonnet. Per piping isometric drawing [3], global Y-direction or vertical is aligned with the valve bonnet axial direction, and therefore FY from pump inlet nozzle stress report will be used for valve bonnet axial stress calculation.

The forces and moments from the stress report are presented in Table 1. The load combinations corresponding to each Service Level are as follows:

Level A: P + DW

Level B: P + DW + OBE

Level C/D: P + DW + SSE

3.5 Stresses Calculation

3.5.1 *Pressure Stress*

The axial stress due to pressure is calculated as

$$\sigma_m = P R_i^2 / (R_o^2 - R_i^2) \quad (1)$$

where

P = design pressure (psi)
R = radius of the pipe (in)
t = thickness of the pipe (in)

3.5.2 *Deadweight Stress*

The axial stress due to deadweight is calculated as

$$\sigma_m = F/A \quad (2)$$

where

F = axial force
A = area of the pipe cross section (in^2)

3.5.3 *SSE and OBE Stresses*

The valve is modeled as a cantilever beam with a rigid support condition at the bottom, i.e. at the pipe connection as shown in Figure 3. The stress at the valve bonnet is caused by the portion of the valve from the leak location to the top of the valve hand wheel.

The seismic acceleration is *3.0g* in the horizontal direction and *2.0g* in the vertical direction per seismic design criteria in the valve equipment specification [5]. These accelerations are assumed to be for SSE. Also, a multi-mode factor of 1.5 is included to account for response at different frequency per NUREG-0800 [6, page 3.7.2-9].

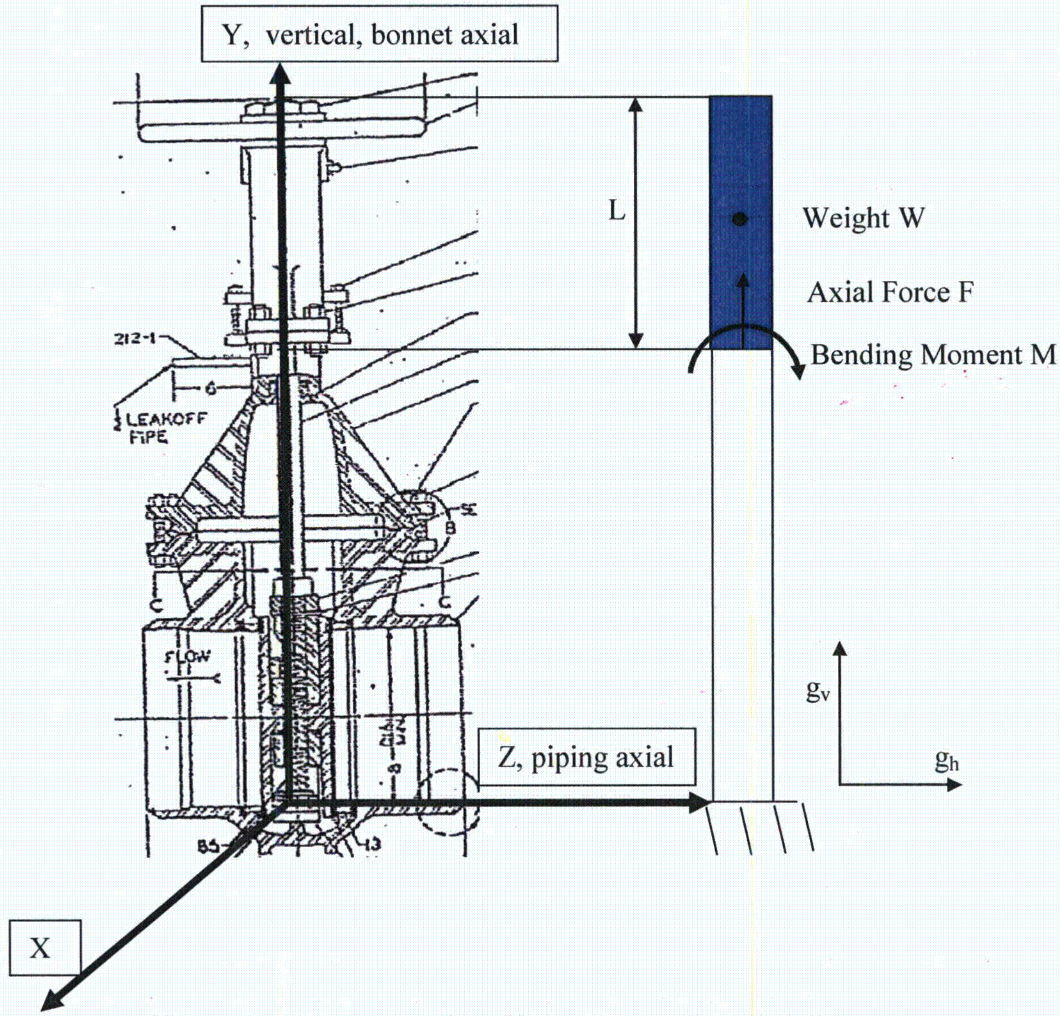


Figure 3: Schematic of the Valve Bonnet Load Model

The membrane and bending stresses are calculated as:

$$\sigma_m = MM * g_v * W / A \qquad (3)$$

where

- MM = multi-mode factor, 1.5
- g_v = vertical seismic acceleration, 2.0g
- W = the weight of portion of the valve from leak location to the top of the valve hand wheel (lb), taken as 35 lb [7]

The bending moments are calculated using distributed loads:

$$M = MM * g_h * W * L / 2 \qquad (4)$$

where

- MM = multi-mode factor, 1.5
- g_h = horizontal seismic acceleration, 3.0g

M = moment at pipe connection (in-lb)
L = length of beam (in), distance from the leak location to the top of the valve hand wheel, 13.5" [7]

The bending stresses are calculated as:

$$\sigma_b = M / Z \tag{5}$$

where

Z = section modulus of pipe (in^3)

The resultant axial stress and bending stresses due to various loads are shown in Table 1. OBE loads are conservatively taken to be half of the SSE loads.

The torque loading from an operator closing the valve causes axial stresses in the yoke at the piping contact region, and does not affect valve bonnet. The eccentric axial force that may be induced by an operator closing the valve introduces a bending moment on the valve bonnet. This bending moment is judged to be insignificant compared to other loads on the valve bonnet such as OBE.

Table 1: Loads and Resultant Stresses

Load	P	Anchor pt	Seismic		Resultant Stress	
		FY	FY	M	σ_m	σ_b
	(psig)	(lb)	(lb)	(in-lb)	(ksi)	(ksi)
Pressure	180.0	--	--	--	0.065	
DW	--	301.0			0.035	
OBE			52.5	531.6	0.006	0.100
SSE			105.0	1063.1	0.012	0.200

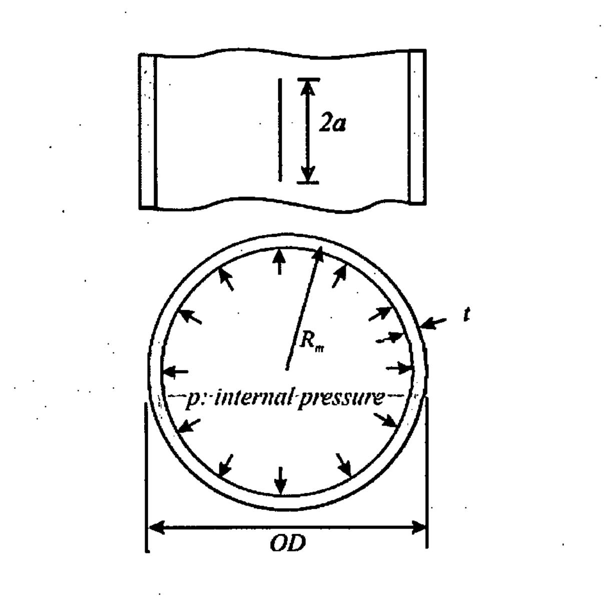
Note: OBE loads are conservatively taken as half of the SSE loads.

3.6 Stress Intensity Factors

Stress intensity factors are calculated for the postulated axial and circumferential through-wall flaws using fracture mechanics crack models of an axial or circumferential flaw in a pressurized cylinder. The stress intensity factors are determined using the **pc-CRACK™** [8] quality assured fracture mechanics software. The flaw models are shown in Figure 4 for the axial through-wall flaw and in Figure 5 for the circumferential through-wall flaw. The stress results derived in Section 3.5 are software inputs to determine the stress intensity factors for each of the postulated flaws.

The **pc-CRACK™** fracture mechanics model for the axial flaw internally calculates the hoop stress based on the pressure input for evaluation.

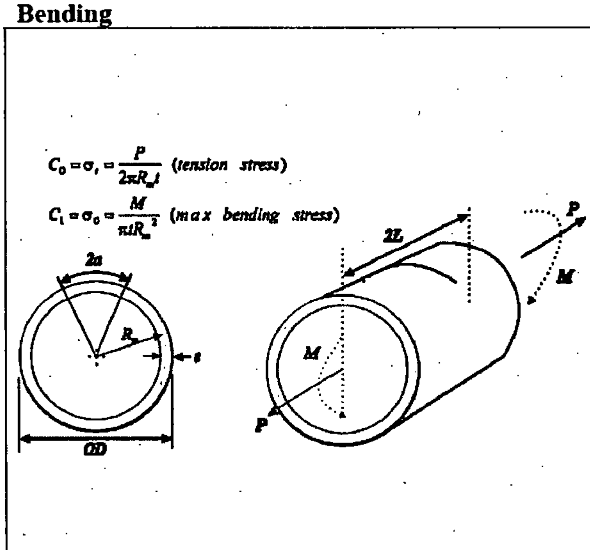
Crack Model: 310 - Through-Wall Axial Crack in Pressurized Cylinder



Stress/Load Input	
Stress Coefficients	✓
Coeffs. from Stress Table	×
Stress Table	×
Stress Intensity Factors (1D)	×
Stress Intensity Factors (2D)	×
C_0 = Internal pressure	
Crack Dimensions: a Component Dimensions: t R_m	
Range: $a/(R_m t)^{½} \leq 10$ Source: [14]	

Figure 4: Fracture Mechanics Model for Axial Through-Wall Flaw [11]

Crack Model: 311 - Through-Wall Circumferential Crack in Cylinder Under Tension And Bending



Stress/Load Input	
Stress Coefficients	✓
Coeffs. from Stress Table	×
Stress Table	×
Stress Intensity Factors (1D)	×
Stress Intensity Factors (2D)	×
C_0 = Membrane Stress C_1 = Max. Bending Stress	
Crack Dimensions: a Component Dimensions: t R_m	
Range: $1.5 < R_m/t \leq 100$ $a/(\pi R_m) < 0.611$ Source: [13]	

Figure 5: Fracture Mechanics Model for Circumferential Through-Wall Flaw [12]

3.7 Fracture Toughness

The material toughness, J_{Ic}, values for the CF-8 CASS material is obtained from NUREG/CR-4513 [9, Figure 25]. As illustrated in Figure 6 (reproduced from [9]), the J_{IC} for aged CASS material shows a lower bound fracture toughness value of $J_{IC} = 200$ kJ/m^2 at room temperature and $J_{IC} = 150$ kJ/m^2 at 290°C. The J_{IC} value is interpolated between the two temperatures and calculated as 186 kJ/m^2 at 205°F. The value is converted to $J_{IC} = 1062$ in-lb/in^2 to calculate the critical stress intensity factor. The aged CASS material is conservatively used, recognizing that J_{IC} for aged CASS is lower than the unaged material acknowledging that the valve bonnet material may not be thermally aged at the maximum operating temperature of 205°F.

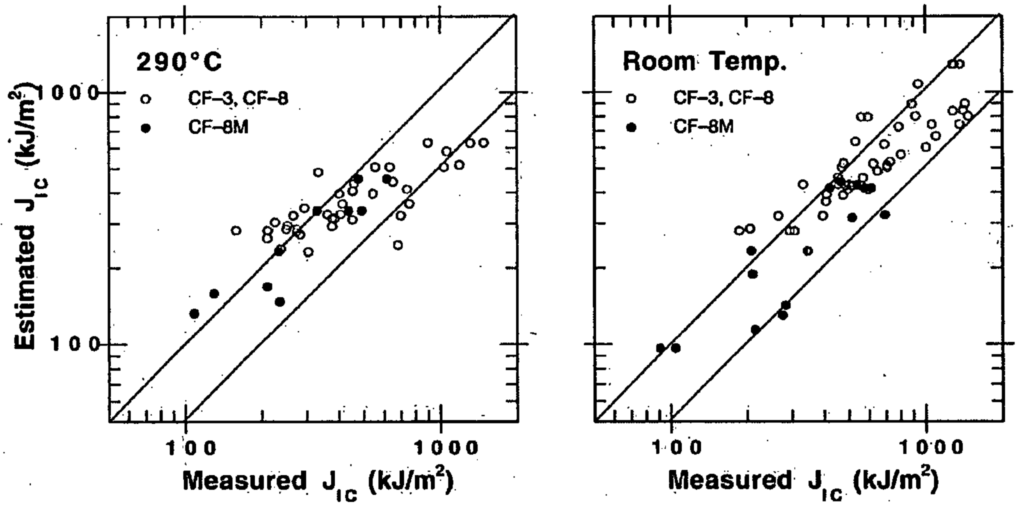


Figure 6: Experimental and estimated value of J_{IC} for aged cast stainless steel [9]

Thus, using the fracture toughness, the critical stress intensity factor is calculated as:

$$K_{Ic} = (J_{IC}E'/1000)^{0.5} = 177.8 \text{ ksi-}\sqrt{\text{in}} \quad [1, \text{H-7200}]$$

where

$E' = E/(1-\nu^2)$
E = Young's modulus = 27,100 ksi
ν = Poisson ratio = 0.3

Critical and allowable flaw sizes are determined using the LEFM approach as described Section 2.0 for postulated through-wall axial and circumferential flaws in the valve bonnet. Article H-7000 of Section XI [1], provides a Safety Factor (SF) to various applied stress intensity factors at each Service Level. The modified applied stress intensity factor is then compared to the critical material fracture toughness, K_{IC}, for allowable flaw size determination. Alternatively, applying the appropriate Safety Factors to K_{IC} for the different Service Levels gives the allowable stress intensity factor, K_{allow}, for each Service Level.

Safety Factors for circumferential flaws are provided in ASME Section XI, Appendix H [1]:

Service Level	Safety Factor, SF
A/B	2.77
C/D	1.39

Hence, the allowable stress intensity factors for circumferential flaws at each Service Level are:

Service Level	K_{allow} (ksi-√in)
A/B	64.2
C/D	128.0

Safety Factors for axial flaws are provided in ASME Section XI, Appendix H [1]:

Service Level	Safety Factor, SF
A/B	3.0
C/D	1.5

Hence, the allowable stress intensity factors for axial flaws at each Service Level are:

Service Level	K_{allow} (ksi-√in)
A/B	59.3
C/D	118.6

4.0 RESULTS

4.1 Allowable and Critical Flaw Lengths

The allowable and critical flaw sizes are determined by comparing the calculated stress intensity factors to the valve material allowable stress intensity factor, K_{allow}, and fracture toughness, K_{Ic}, respectively. The calculated stress intensities for circumferential flaws and axial flaws are presented in Figure 7 and Figure 8, respectively. Note that Figure 8 has only one curve for all Service Levels because only the pressure based hoop stress is considered for the axial flaw evaluation; as a result, all the Service Levels have the same load.

For the circumferential flaw, an upper limit of crack length exists in the fracture mechanics crack model illustrated in Figure 5 ($a / \pi R_m < 0.611$). As a result, the half crack length a in Figure 7 is limited to 2.815". Comparing the calculated stress intensity factors in Figure 7 to the circumferential K_{allow} values given in Section 3.0, it is concluded that K_{allow} is not exceeded at all circumferential flaw Service Levels and the allowable circumferential through-wall flaw length is at the model limit of 5.63" (61.1% of the circumference). This conclusion is the same for the critical circumferential crack length since K_{IC} is greater than K_{allow}. These results are summarized in Table 2.

Similarly, an upper limit exists for the axial flaw fracture mechanics model as shown in Figure 4 ($a \leq 10 \sqrt{R_m t}$). As a result, the half crack length a in Figure 8 is limited to 11.7". Comparing the calculated stress intensity factors in Figure 8 to the axial K_{allow} values given in Section 3.0, it is concluded that K_{allow} is not exceeded at all axial flaw Service Levels and the allowable axial through-wall flaw length is at the model limit of 23.4" (2 x 11.7"). This conclusion is the same for the critical circumferential crack length since K_{IC} is greater than K_{allow}. These results are summarized in Table 2.

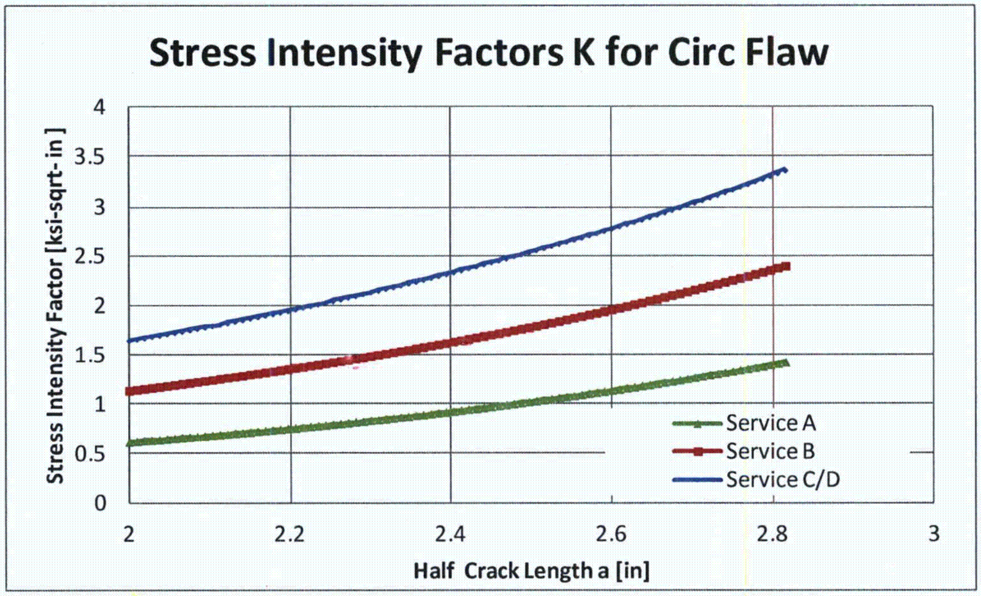


Figure 7: Stress Intensity Factors for Circumferential Flaws at Various Service Levels

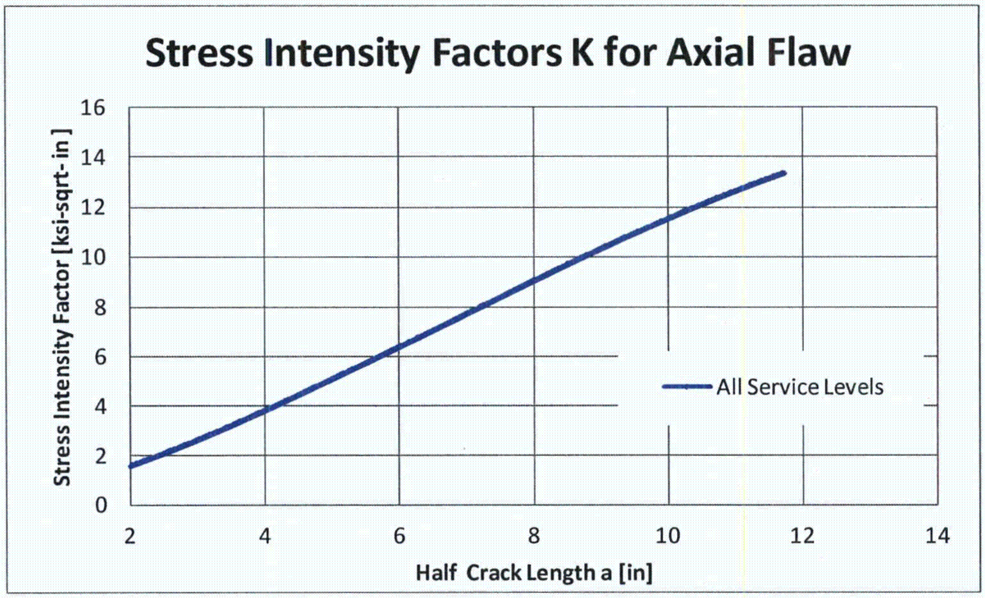


Figure 8: Stress Intensity Factors for Axial Flaws

Table 2: Allowable and Critical Flaw Lengths

Circumferential Flaw		Axial Flaw	
Allowable	Critical	Allowable	Critical
5.63"	5.63"	23.4"	23.4"

Note: The allowable and critical flaw lengths are the maximum permitted by the applicability range of each fracture mechanics model.

4.2 Flaw Growth

Fatigue crack growth is not expected due to the low resultant stresses summarized in Table 1 and the limited number cycles applied to this valve per operating cycle [7]. SI Report 1301208.402 [13] addresses the potential for flaw growth due to corrosion mechanisms (originated as part of this same project) including stress corrosion cracking. Flaw growth due to corrosion is not expected for the 3-844A valve.

5.0 CONCLUSIONS

Fracture mechanics analyses were performed for the Turkey Point Unit 3 Nuclear Plant to evaluate the through-wall flaw discovered in the 3-844A valve bonnet. Postulated allowable through-wall flaw lengths in the circumferential and axial directions were determined to show structural stability following ASME Section XI guidance understanding that Section XI does not specifically address through-wall flaws.

The ASME Section XI based allowable and critical (no safety margin) circumferential through-wall flaw lengths are both determined to be 5.63 inches (due to the fracture mechanics model applicability limits being reached). Similarly, the allowable and critical axial through-wall flaw lengths are both determined to be 23.4 inches (also due to the fracture mechanics model applicability limits being reached).

Fatigue and corrosion crack growth are evaluated for the valve and determined to be negligible.

Please contact us if you have any questions. Thank you.

Prepared by:		Verified by:	
Peihua Jing Senior Engineer	9/19/2013 Date	Shu S. Tang Associate	9/19/2013 Date

Approved by:

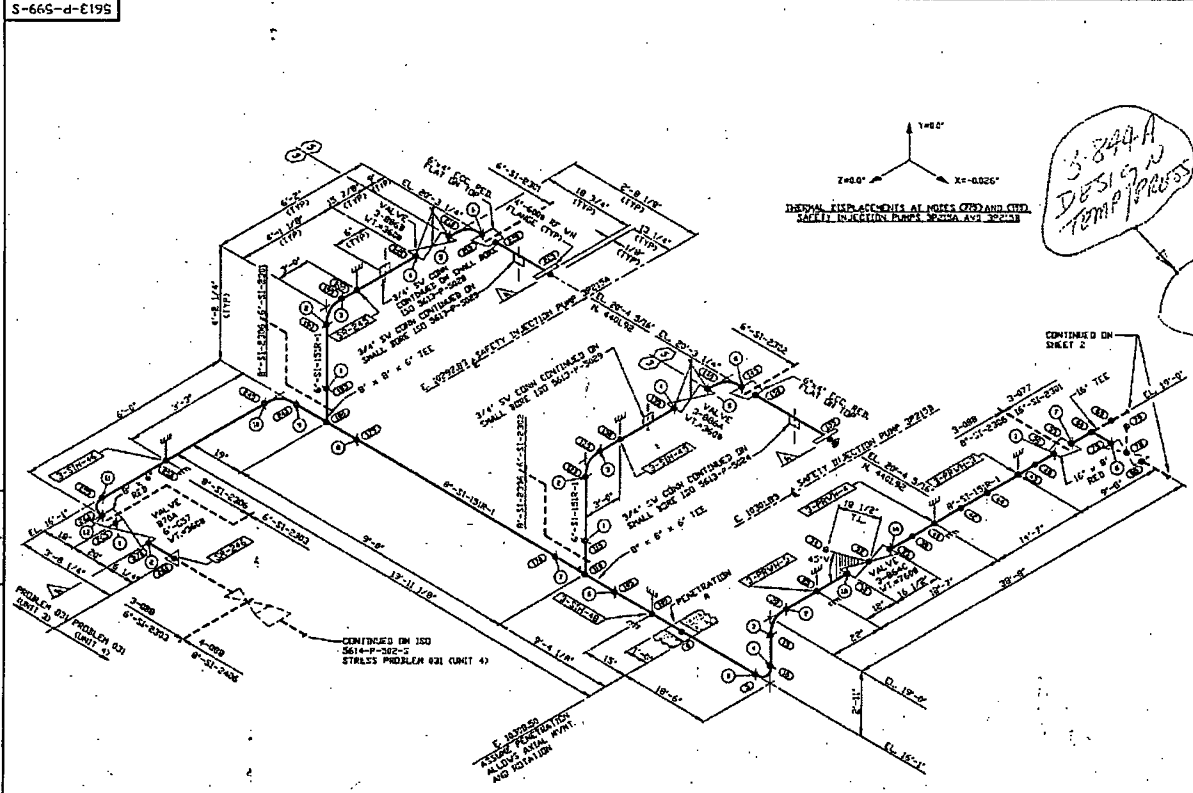
Robert O. McGill, P.E. Senior Associate	9/19/2013 Date

Attachments (5)

Attachment 1

Piping Isometric Drawings (Reference 3)

5613-P-599-S



RESTRAINTS	SINGLE ACTING IN DIRECTION OF STEM	
SNUBBER	DOUBLE ACTING	SPRING HANGER
ANCHOR	GROUTED ANCHOR	WELDED STANCHION
SUPPORT MARK NO.	ANALYSIS NODE NO.	

GLOBAL COORDINATE SYSTEM

REFERENCE P&ID NO. 5613-M-3062 SHEET 1
5613-M-3068 SHEET 1
REFERENCE OPERATING DIAGRAM NO. 5613-M-3062 SHEET 1
5613-M-3068 SHEET 1

HIGH HEAD SAFETY INJECTION PIPING FROM THE RWST TO THE SI PUMPS 3P215A & B AND TO THE CONTAINMENT SPRAY PUMPS 3P214A & B THRU VALVE 3-870B

THIS DRAWING MADE FROM 79-14 WALKDOWN INDEX FOR STRESS PROBLEM 031
REFERENCE DOCUMENTS: 5177-102-SK-P-358 REV. E
5177-102-SK-P-359 REV. B
5177-102-SK-P-360 REV. B

SEISMICALLY ANALYZED PIPING

REGION	HYDRO	DESIGN		MAX OPERATING	
	PRESS. PSIG	TEMP °F	PRESS. PSIG	TEMP °F	PRESS. PSIG
A	250	300	250	205	160

SYSTEM TEMPERATURE AND PRESSURE OPERATING MODES
SEE TABLE "A"

ISI WELD NUMBER — WHIP RESTRAINT — WELD SYMBOL

PIPE MATERIAL	LINE SPEC	LINE SIZE	INSULATION
A358 & A312 TP 304 WLD	16"-SI-151R-2	16" SCH 10	NONE
A312 TP 304 WLD	10"-SI-151R-3	10" SCH 10S	NONE
A312 TP 304 SMLS OR WLD	8"-SI-151R-1	8" SCH 10S	NONE
A312 TP 304 SMLS OR WLD	4"-SI-151R-1	4" SCH 10S	NONE
A312 TP 304 SMLS OR WLD	6"-SI-151R-1	6" SCH 10S	NONE
A358 TP 304 SMLS OR WLD	8"-SI-151R-2	8" SCH 10S	NONE
A358 TP 304 SMLS OR WLD	14"-SI-151R-P	14" SCH 10	NONE

NOTES:
1) ALL ELBOWS ASSUMED LONG RADIUS UNLESS OTHERWISE SPECIFIED
2) FOR PIPE SUPPORT DETAILS SEE DRAWING SERIES 5613-H-599
3) FOR SUPPORT ISI INFORMATION SEE STRESS PROBLEM 033
4) NODE POINTS WITH MORE THAN ONE NUMBER ARE INDICATED AS STRESS PROBLEM 031/STRESS PROBLEM 033
5) FLOODWALL PENETRATION ALLOWS FOR PIPING MOVEMENT. FOR COMPLETE DETAILS SEE DRAWING 5610-C-397, SHEET 4.
6) SUPPORT STIFFNESSES MAY APPLY. CHECK ANALYSIS OF RECORD FOR STIFFNESSES.
7) SEE CALCULATION 8N00-102-073 FOR EQUIPMENT MOTIONS, VALVE WEIGHTS, ETC.
8) AT LOCATION SHOWN, A 2 1/2" DIA. X 4" LG TRUNNION IS WELDED TO THE PIPE WITH A 3/16" FILLET WELD ALL AROUND.
9) ALL ISI INFORMATION, WELD NUMBERS & LOCATIONS, ZONES, LINE NUMBERS & WHIP RESTRAINT LOCATIONS ARE FOR ISI REFERENCE ONLY.

ZONE NUMBER	LINE NUMBER
3-077	16"-SI-2301
3-088	6"-SI-2301
3-088	6"-SI-2302
3-088	6"-SI-2303
3-088	8"-SI-2306

TABLE "A"

MODES OF OPERATION	1	2	3	4	5	6
EMERGENCY INJECTION	100° 25 PSIG	100° 25 PSIG	100° 25 PSIG	100° 25 PSIG	100° 25 PSIG	100° 25 PSIG
RECIRCULATION MODE 1	205° 156 PSIG	205° 156 PSIG	100° 156 PSIG	100° 25 PSIG	100° ATM	205° 156 PSIG
RECIRCULATION MODE 2	100° 156 PSIG	205° 156 PSIG	205° 156 PSIG	205° 156 PSIG	205° 156 PSIG	100° ATM

NO	DATE	REVISION
5		REVISED PER BCR-TPN-93-415, ADDED ISI WELD NUMBERS, ZONE NUMBERS AND WELD SYMBOLS + REVISION CLOUD OMITTED
4		AS BUILT PER BCR-TPN-88-550
3		AS BUILT PER BCR-TPN-86-449
2		ISSUED AS BUILT FOR PC/M 79-014
1		TELEDYNE ES REDRAW PER PCN 82-217

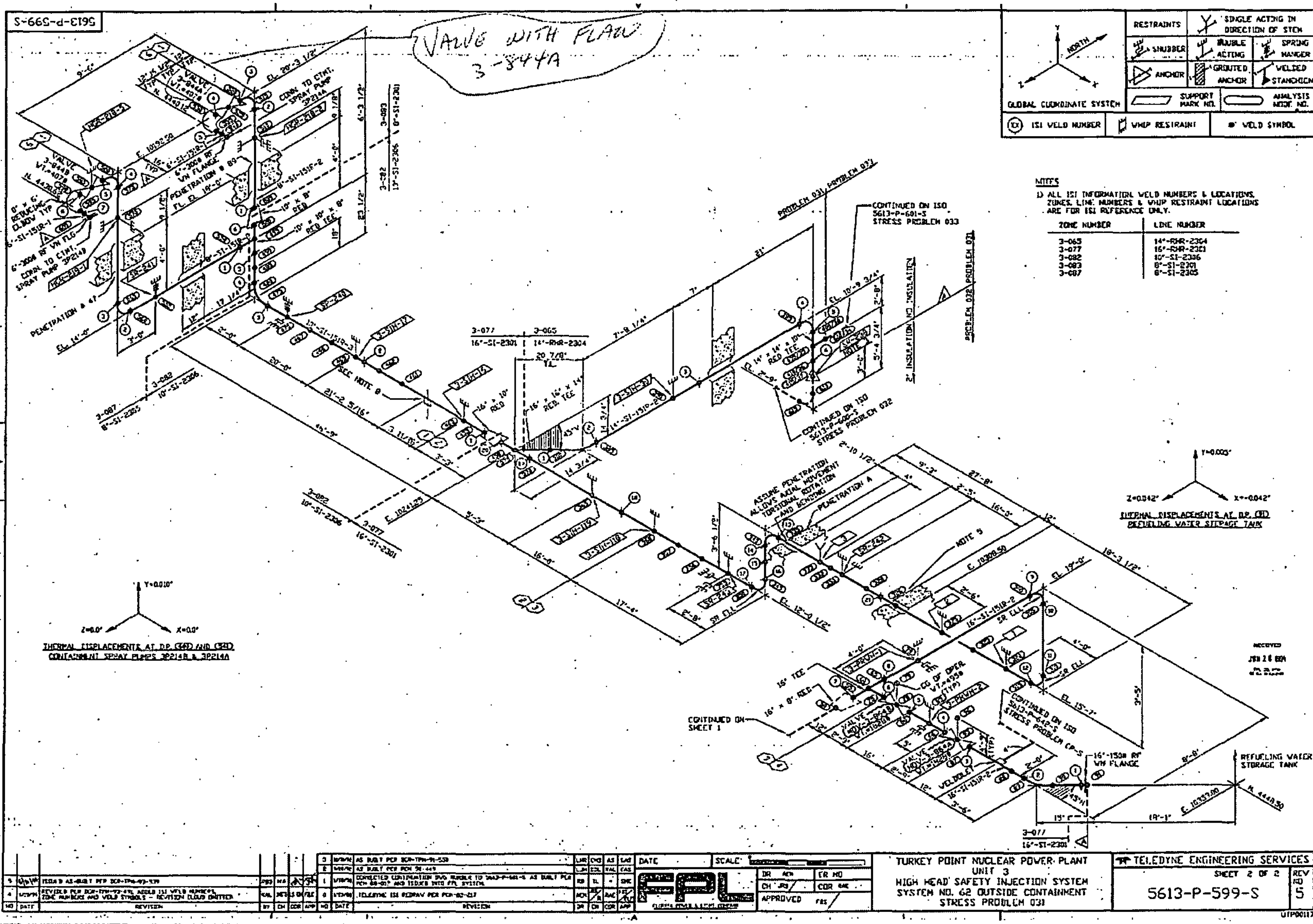
FPL

DATE | SCALE | DR | CR NO | CH | CCR | APPROVED

TURKEY POINT NUCLEAR POWER PLANT
UNIT 3
HIGH HEAD SAFETY INJECTION SYSTEM
SYSTEM NO. 62 OUTSIDE CONTAINMENT
STRESS PROBLEM 031

TELEDYNE ENGINEERING SERVICES
SHEET 1 OF 2
5613-P-599-S
REV NO 5

UTP00343



Structural Integrity Associates, Inc.®

Attachment 2

Containment Spray Pump Inlet Nozzle Stress Report (Reference 4)

BY W.A. DATE 5/15/86
CHKD. BY JBD DATE 5/16/86

F.P.& L. TURKEY POINT UNIT 3 NRC I & E BULLETIN 79-14 ANALYSIS

SHEET NO. ____ OF ____
PROJ. NO. 5322/5875

NOZZLE EVALUATION PROB 031 UNIT 3
CONTAINMENT SPRAY PUMPS - SUCTION SIDE
D.P. 540 & 600 Ref ISO 5177-102-SF-P359 R/B TES P/O ANALYSIS

(UNITS FT-LBS)

CASE	Fx	Fy	Fz	Mx	My	Mz
D.P. 540						
D.W.	30	301	-27	-270	0	-149
THERMAL 1	-115	-29	-251	906	-642	-53
" 2	-399	-85	-654	2399	-1731	-157
" 3	-181	-122	-750	2709	-1683	-60
SSE ±	83	238	124	194	84	505
D.P. 600						
D.W.	-34	68	1	-71	-39	18
THERMAL 1	-80	442	194	-192	158	-624
" 2	-380	539	631	109	429	-1014
" 3	-150	860	558	-125	546	-1168
SSE ±	22	321	64	142	193	492

D.P. 540

$$MAX\ F_{SRSS} = \left[(30-399-83)^2 + (301+238)^2 + (-27-750-124)^2\right]^{1/2}$$
$$= (452^2 + 539^2 + 901^2)^{1/2} = 1143\ \#$$

$$MAX\ M_{SRSS} = \left[(-270+2709+194)^2 + (-1731-84)^2 + (-149-157-505)^2\right]^{1/2}$$
$$= (2633^2 + 1815^2 + 811)^2 = 3299\ FT\text{-}\#$$

THERMAL 1 ≡ EMER. INJECTION ; THERMAL 2 ≡ RECIRC MODE 1
THERMAL 3 ≡ RECIRC MODE 2

Attachment 3

Seismic Specification of the 3-844A Valve (Reference 5)

EQUIPMENT SPECIFICATION

4.0 REQUIREMENTS

4.1 Arrangement

4.1.1 Each isolation valve shall include a stuffing box with leak-off (if required), yoke, diaphragm motor operator, and limit switches for remote indication of the full open and/or full closed positions. See valve Specification Sheet.

4.1.2 Each modulating valve shall include a stuffing box with leak-off (if required), yoke, diaphragm motor operator, positioner, and limit switches for remote indication of the full open and/or full closed positions. See valve Specification Sheet.

4.1.3 The valve specification sheets will indicate all particular and special requirements in addition to the general requirements as delineated. Any exceptions to this specification will be covered by additions or Addendums to the specification sheets.

4.2 Design Conditions - General

4.2.1 The design of each valve shall satisfy all conditions and requirements set forth on the specification sheet including the water chemistry and thermal transient effects. Abbreviations and symbols relate to a description of the various water chemistries and thermal transients appearing in the Special Requirements addendum, Section 8, of this specification.

4.2.2 The ambient temperature is 120°F, ambient pressure is 8 to 15 psia; 100% relative humidity.

4.2.3 The design life of each valve shall be 40 years.

4.2.4 The number of open-shut cycles shall be 500 per year for the design life.

4.2.5 All valve assemblies shall be designed to withstand seismic loading equivalent to 3.0g in the horizontal direction and 2.0g in the vertical direction. When exposed to the above loadings, the valve shall be capable of performing all functions intended within this specification.

4.2.6 The maximum operating air pressure to main seat and/or backseat these valves shall not exceed 85 psig.

WESTINGHOUSE ELECTRIC CORPORATION
ATOMIC POWER DIVISIONS

Westinghouse Form 56062

Revision No. 1

E-Spec. 676270

Page 6 of 19 Pages

Attachment 4

Relevant Design Input (Reference 7)

From: Toner, Mark [Mark.Toner@fpl.com]
Sent: Wednesday, September 18, 2013 6:56 AM
To: McGill, Bob
Cc: Jing, Peihua; Tang, Stan; Barnes, Philip R
Subject: RE: Valve Body Leak - Design Input Request
Attachments: 3-844A Chem-Cycles.pdf

Bob,

1) Attached please find the present Chemistry information for the process fluid (borated water). During accident conditions this water is potentially containing small radioactive particles, Sodium Pentaborate and other chemicals found in the post LOCA containment sump water (it'll be recirculated by the Containment Spray System (CSP) to the containment for steam suppression.
2) The history of cycles for this manual valve over the past year is on the same sheet. It indicates 4 open/closed cycles during a year with a long outage. Using 10 cycles per year would be conservative for this manual maintenance isolation valve.
3) Distance from the leak location to the top of the valve is 13-1/2"
4) Estimated weight of the yoke + handwheel is 25-35#.
5) Ambient temperature in the room is ~95F. Accident temperature will be confirmed later but for now assume it's the max pipe operating temperature of 205F.
6)The OD of the packing gland area at the leak measured 3-7/8" with a caliper. This agrees will scaling the drawing. Will try to confirm with UT later but we're not having much success with UT on the curved casting surface.

Regards,

Mark Toner

From: McGill, Bob [mailto:Rmcgill@Structint.com]
Sent: Tuesday, September 17, 2013 11:08 PM
To: Barnes, Philip R
Cc: Toner, Mark; Jing, Peihua; Tang, Stan
Subject: RE: Valve Body Leak - Design Input Request

Phil and Mark,

Using the bounding nozzle loading is really too conservative and not entirely correct to apply at the flaw location. Thus, it would be helpful if you could provide the following additional design inputs:

1. What is the distance from the leak location to the top of the valve hand wheel?
2. Do you have an estimate of the weight of this portion of the valve?
3. Do you have a conservative number of g's we should consider for the seismic load? We were thinking 10g would certainly bound, but that seems very high.

Thanks.

Bob McGill, P.E.
Senior Associate
***Structural Integrity** Associates, Inc.*
Experts in the prevention and control of structural and mechanical failures
5215 Hellyer Avenue, Suite 210

Attachment 5

Design Input for Valve Maximum Operating Pressure (Reference 10)

psig). The cumulative pressure in this line could thereforeapproach 300 psig near the pump discharge and drop with the associated line losses down to the containment ambient pressure (in this case 14 psig).

200 Psig
CS Pump suction lines from MOV-*-844 A/B: The maximum service conditions would occur after isolation of contaminated sump water in this line during a Post-LOCA recovery lineup. Radioactive decay of the enclosed fluids would cause thermal expansion and potential overpressure conditions. Appropriate protection is provided by RV-*-871 at 200 psig. This setpoint was selected to be high enough to avoid challenging this relief valve when the RHR pumps are supplying the HHSI and/or CS Pumps during post-accident recirculation. (The discharge pressure of the RHR pYumps during post-accident recirculation is approximately equal to the containment pressure (0 to 60 psig) plus the RHR pumps' total developed head (100 to 130 psig). Therefore, the suction pressure to the containment spray pumps and high-head SI pumps can range from 130 psig to 180 psig). (References 7, 8, 9, 18).

Note that CS Pump IST does not challenge this relief setpoint either. Although the CS Pump discharge pressure approaches pump shutoff conditions (265 psig), the pressure drop of the flow indicating orifice and restricting orifice, however, keep suction line pressures much below 200 psig.

Design Temperatures

200 °F
Components and piping in the Post-LOCA Recirculation flowpath, must be designed to withstand, without loss of function, the maximum temperature expected to occur during the post-accident recirculation mode of operation. 200°F is the maximum expected temperature based on a conservatively high 300°F sump temperature minus 100°F temperature drop across the RHR heat exchanger.

For the Thermal Uprate project, a revised maximum temperature of 205°F was defined for LOCA recirculation operation. (Reference 58)

2.3.4 Seismic Requirements

The CS System is classified in UFSAR Appendix 5A as a Class 1 system. The system is Class 1 because the system is an engineered safety feature. Class 1 systems must meet Turkey Point General Design Criterion 2, Performance Standards, including design to withstand earthquakes without loss of function.

The original Turkey Point plant design did not include specific seismic design requirements, because South Florida is historically a seismically dormant region of the country (UBC Zone 0). However, adoption of the 1967 AEC draft General Design Criteria (specifically GDC 2) forced FPL to impose a nominal seismic requirement on Class 1 systems.

Report No. 1301208.402
Revision 0
Project No. 1301208
September 2013

Corrosion Evaluation of Valve 3-844A at Turkey Point Unit 3

Prepared for:

Florida Power and Light
Turkey Point Nuclear Plant
Florida City, Florida
Contract Number 2321491

Prepared by:

Structural Integrity Associates, Inc.
San Jose, California

Prepared by:	Barry M. Gordon, P.E.	Date:	9/19/2013
Reviewed by:	Anthony J. Giannuzzi, Ph.D., P.E	Date:	9/19/2013
Approved by:	Robert O. McGill, P.E.	Date:	9/19/2013

Structural Integrity Associates, Inc.®

REVISION CONTROL SHEET

Document Number: **1301208.402**

Title: Corrosion Evaluation of Valve 3-844A at Turkey Point Unit 3

Client: Florida Power & Light

SI Project Number: **1301208** Quality Program: ☒ Nuclear ☐ Commercial

Section	Pages	Revision	Date	Comments
1.0	1-1 – 1-6	0	9/19/2013	Initial Issue
2.0	2-1 – 2-2			
3.0	3-1 – 3-8			
4.0	4-1			
5.0	5-1			
App. A	A-1 – A-2			
App. B	B-1 – B-2			

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1.0 INTRODUCTION

A non-destructive examination (NDE) inspection performed on September 17, 2013 after cleaning off the boric acid buildup identified a void in the ASTM-351 Grade CF8 valve bonnet above the packing leak-off line at Turkey Point Unit 3, Figure 1-1 [1]. The staff at Turkey Point Unit 3 noted that there was no observable leakage and that the "void" appeared to be an original casting defect. The two primary concerns for this indication are fluid inventory control and structural integrity of the flaw.

Valve 3-844A is presently exposed to static head of the refueling water storage tank (RWST) and is pressurized to post-accident piggy-back pressures every outage via the emergency core cooling system (ECCS) suction hydrostatic leak test via 3-OSP-202.4 [1]. The specific location of the flaw is in the valve bonnet above the packing and is additionally isolated from system pressure via the valve backseat per 3-NOP-068. As a probable original casting defect, this flaw was present during the most recent test near the beginning of PT3-26 RFO in spring 2012 per 3-OSP-202.4 with satisfactory results. Therefore, the staff at Turkey Point Unit 3 considers that it is reasonable to assume that any potential leakage in the post-accident piggy-back alignment is within UFSAR limits for ECCS leakage outside of containment.

The casting void noted herein appears localized to the area of the bonnet just under the bonnet to yoke flange [1]. Typically, the stresses are low at this location for manual valves that remain locked open. Previous flaw evaluations of 4-inch stainless steel pipes have allowed more than half of the circumference of a pipe (Ref AR 570552 POD Rev 2). Photographs and NDE inspection show that the current flaw is significantly less than half of the bonnet circumference in this location. As such, there is reasonable assurance that the immediate condition is operable.

This report is designed to perform a corrosion engineering evaluation of the possibility and propensity for environmentally-assisted cracking (EAC) of a cast ASTM-351 CF8 (0.08% max C, 18 to 21% Cr, 8 to 11% Ni, balance Fe) stainless steel valve located on a RWST that is nominally exposed to nominally 100°F (38°C) borated water, i.e., water containing boric acid (H_3BO_3) at the Turkey Point Unit 3. This evaluation will consist of a historical review of field and laboratory data on EAC with special emphasis on stress corrosion cracking (SCC) of stainless steels in pressurized water reactor (PWR) type borated water environments rather than environmentally-assisted fatigue (EAF) due to the lack of assumed cyclic stress.

1.1 Early Experience with SCC in PWR Borated Systems in Wrought Stainless Steels

A U. S. Nuclear Regulatory Commission (NRC) Office of Inspection and Enforcement information notice issued in 1979 noted that a number of cracking incidents were experienced from November 1974 to February 1977 in safety-related stainless steel piping systems and portions of systems that contained oxygenated stagnant or essentially stagnant borated water [2]. Subsequent metallurgical destructive examinations of the components revealed these cracks initiated on ID surfaces of the weld heat affected zones (HAZs) of 8-inch to 10-inch Type 304 stainless steel (Schedule 10 and 40) piping. Crack propagation occurred in either an intergranular or transgranular mode. Analysis indicated the probable important contributing environmental factors were chloride and dissolved oxygen contamination in the affected systems. The PWRs affected up to that time were Arkansas Nuclear Unit 1, R. E. Ginna, H. B. Robinson Unit 2, Crystal River Unit 3, San Onofre Unit 1 and Surry Units 1 and 2. The NRC issued Circular 76-06 in view of the apparent generic nature of the problem [3].

No further borated system related cracking was identified until March 1979 during the refueling outage of Three Mile Island Unit 1 when visual inspections identified five through-wall cracks at welds in the spent fuel cooling system piping and one crack at a weld in the decay heat removal system. These cracks were found as a result of local boric acid buildup and later confirmed by liquid penetrant tests. This initial identification of cracking was reported to the NRC in a Licensee Event Report (LER) dated May 16, 1979. A preliminary metallurgical analysis was performed by the licensee on a section of a cracked and leaking weld joint from the spent fuel cooling system. The conclusion of this analysis was that cracking was due to intergranular stress corrosion cracking (IGSCC) originating on the pipe ID. The cracking was localized to the HAZ where the Type 304 stainless steel is weld sensitized, i.e., chromium carbides precipitated at the austenite grain boundaries. In addition to the main through-wall crack, incipient cracks were observed at several locations in the weld HAZ including the weld root fusion area where a miniscule lack of fusion defect existed. The stresses responsible for cracking were believed to be primarily residual welding stresses in as much as the calculated applied stresses were found to be less than code design limits. The investigators indicated that there was no conclusive evidence at that time of this report to identify those aggressive chemical species that promoted the IGSCC.

An Electric Power Research Institute (EPRI) review of PWR cracking incidents indicated that there was no generic SCC problem in PWR auxiliary piping containing borated water [4]. Any unresolved incidents of SCC (e.g., Arkansas Nuclear Unit 1 and Three Mile Island Unit 1 [5]) were believed to have been due to high carbon content and high weld heat input weld sensitized Type 304 stainless steel exposed to an environment containing aggressive impurities such as chloride, thiosulfate and dissolved oxygen in borated water.

1.2 Laboratory SCC Test Results in Borated Environments on Wrought Stainless Steels

The SCC propensities of Types 304 and 316 stainless steels in simulated pressure-suppression and fission-product absorption sprays were investigated over forty years ago [6]. The test solutions contained 0.28M H_3BO_3 (3000 ppm B) with pH values ranging between 4.5 and 7.5 and chloride concentrations from 5 to 200 ppm. All SCC test specimens such as U-bends, double U-bends and C-rings were exposed for only one day at 286°F (141°C) and then seven days at 212°F (100°C) in a recirculating spray loop and were then transferred to a tank for an additional exposure of two months at 180°F (82°C).

The results showed that both Types 304 and 316 stainless steels suffered SCC in all test solutions [6]. The tendency to crack was greater in the lower pH environment and in the higher chloride concentration solutions. Type 316 stainless steel was somewhat more resistant to SCC than Type 304 stainless steel and both alloys were more susceptible when furnace sensitized (1250°F [677°C] for one hour) than when exposed in the annealed condition. Specimens that were sensitized in air and covered with thin oxide films cracked more frequently than similarly treated specimens that were cleaned by pickling after the sensitization heat treatment. The cracks were branched and transgranular with only one exception. In one case, when the solution had a pH of

6.5 and contained 20 ppm chloride and 2 ppm iodide, wide short cracks were found only in the vicinity of welds; no other localized attack was found on any other specimen.

Another experimental investigation was conducted of the susceptibility of Type 304 stainless steel to SCC in borated water at 130°F (55°C) [4]. Most importantly, SCC data from this EPRI sponsored program indicated that uncontaminated boric acid solutions did not support SCC. In fact, other data from this program showed that contaminates in the solutions were responsible for SCC of the stainless steel and the order of increasing aggressiveness of contaminating species was fluoride (least detrimental), chloride and thiosulfate (most detrimental); combinations of these species are synergistically more aggressive. Furthermore, as would be anticipated, as the concentration of dissolved oxygen or oxidizing species increases, i.e., as the cathodic reactant increases, the extent of cracking increased. The experimental program also verified that the degree of alloy sensitization and, thus, cracking susceptibility, depend on base-metal carbon content and welding heat input.

The IGSCC susceptibility of various sensitized austenitic stainless steels (e.g., Types 304, 304L, 316, 316L and 316LN stainless steels) was investigated in oxygenated high-purity deionized water (e.g., chloride <0.05 ppm) and borated water (e.g., 2,100 ppm as B) at temperatures ranging from 86°F (30°C) to 464°F (240°C) by using different SCC test specimen configurations such as uniaxial constant load, creviced bent beam and double U-bend specimens [7]. The results indicated that there was no significant difference in IGSCC susceptibility between the two environments. In fact, a statistical analysis of SCC test results at much higher temperatures than would be experienced at Turkey Point Unit 3, i.e., 100°F (38°C) vs. 464°F (240°C), revealed that the median (50% of cumulative probability) time-to-failure shows less IGSCC susceptibility for sensitized Type 304 stainless steel (0.06% C) uniaxial constant load specimens tested at an applied tensile stress of 49.8 ksi (35 kg/mm^2) in borated water (e.g., 132 hours) compared with that in high-purity to oxygenated (8 ppm O_2) water (e.g., 55 hours), Figure 1-2.

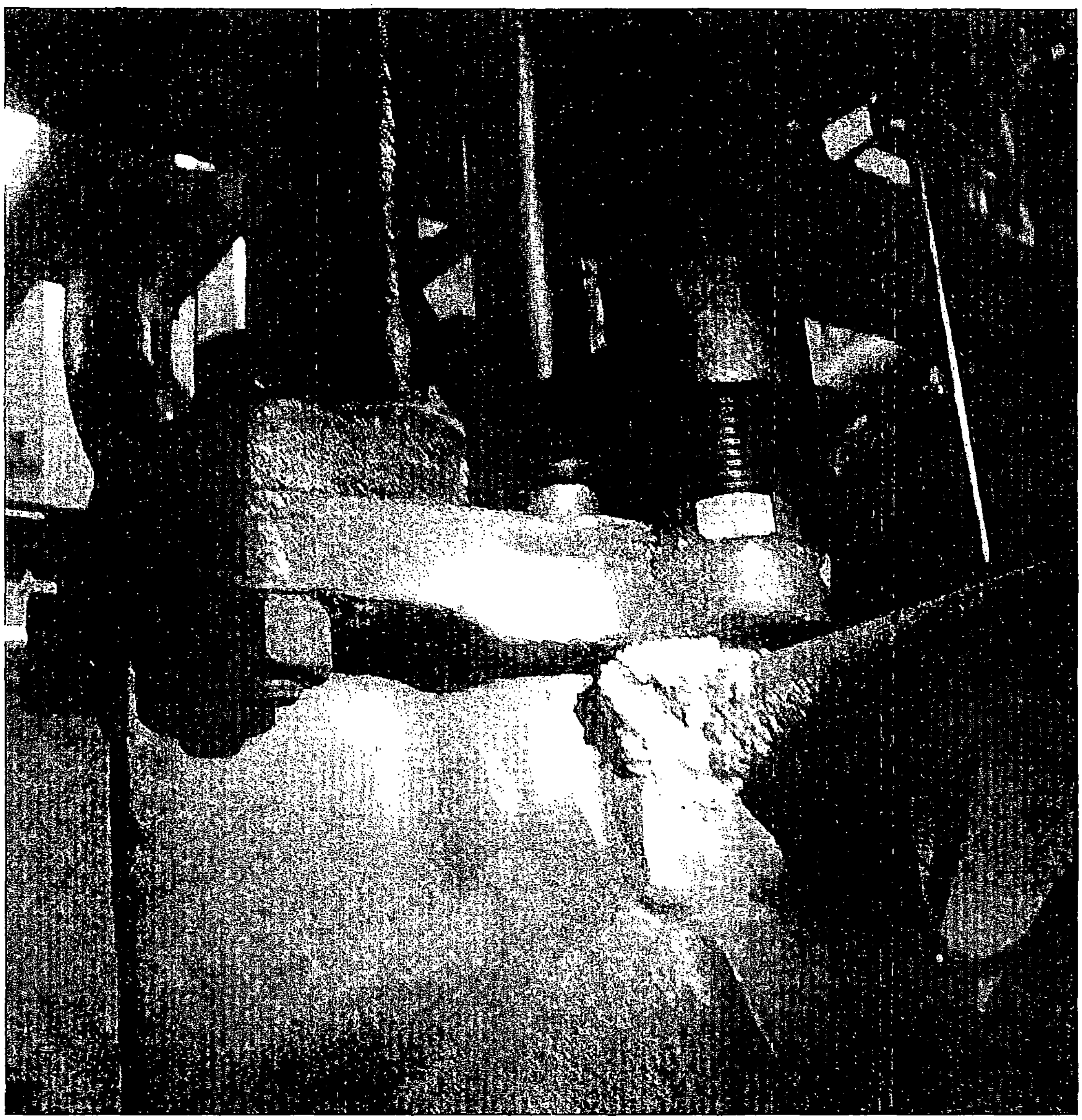

Figure 1-1. Boric Acid Deposits from Leaking CF8 Valve at Turkey Point Unit 3

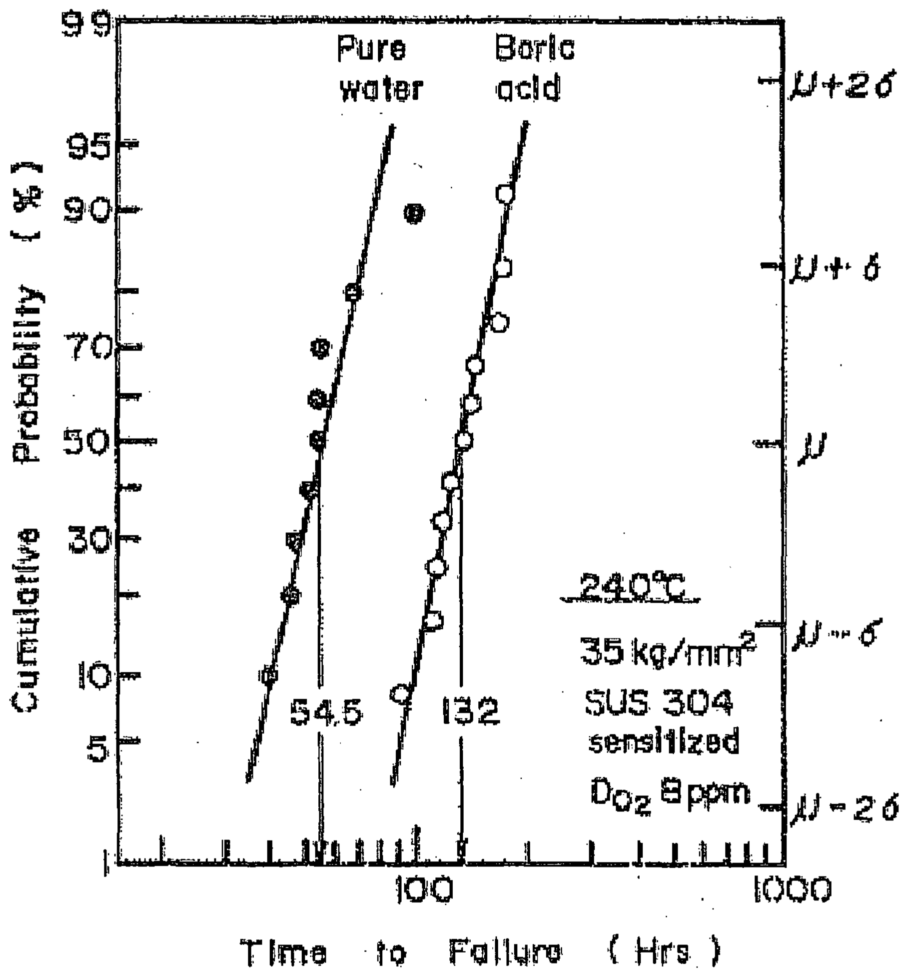

Figure 1-2. Comparison of Time-to-Failure in Oxygenated High-Purity Water and Borated Water for Sensitized Type 304 Stainless Steel [7]

2.0 TURKEY POINT UNIT 3 RSWT ENVIRONMENT AND CORROSION

Table 2-1 presents the recent RWST water chemistry for the Turkey Point Unit 3 [8]. Since the tank is vented to atmosphere, the dissolved oxygen concentration near the water surface is calculated to be approximately 6.75 ppm at 100°F (38°C) [9]. The presence of dissolved oxygen means that there are two possible cathodic reactions in borated PWR water to facilitate any corrosion reaction:

$$O_2 + 2H_2O + 4e^- \rightarrow 4(OH)^- \text{ (reduction of dissolved oxygen to form hydroxyl)}$$
$$2H^+ + 2e^- \rightarrow H_2 \text{ (reduction of the hydrogen ion from the boric acid to form hydrogen gas)}$$

While these two cathodic reduction reactions can support the anodic dissolution of Fe, Cr and Ni from the CF8 stainless steel components, these cathodic reactions are probably not sufficient to support the initiation of SCC at the low operating temperatures of the Turkey Point Unit 3 RWSTs. Impurities in the water such as chloride, fluoride and sulfate affect the anodic factors of corrosion and degrade the cast stainless steel passive film. While the corrosion reaction does require sufficient cathodic influence, the cathodic reaction alone cannot provide the driving force for corrosion alone.

In fact, the entire operating experience and corrosion test data presented in Section 1.0 suggests that boric acid, per se, is not responsible for the SCC identified in various stainless steel alloys, but it is the presence of the impurities such as chloride, fluoride and sulfate that facilitate SCC in a borated environment. Since the impurity level in the water chemistries of the RWST is so low, i.e., orders of magnitude lower than that identified in the PWRs with SCC in borated systems or used in the corrosion tests described in Section 1.0, it is highly unlikely that the Turkey Point Unit 3 RWST environment would support SCC initiation in cast CF8 unless the tensile stress/strain state were extremely high. These two factors along with relevant low temperature SCC studies will be discussed in greater detail in the next section of this report.

Table 2-1. Water Chemistry Data for Turkey Point Unit 3 RWST [8]

Conductivity, μS/cm	**pH**	**B, ppm**	**Calculated H_3BO_3, %**	**Cl, ppb**	**F, ppb**	**SO_4, ppb**	**Mg, ppb**	**Ca, ppb**
8.68	4.22	2469	1.41%	12.97	2.77	1	1	1

3.0 SCC EVALUATION

3.1 Initiation of SCC at Low Temperatures in Wrought Stainless Steel

As presented in Figure 3-1, the initiation of SCC in high purity environments is difficult below approximately 212°F (100°C) where crack initiation requires significantly higher strains at lower temperatures compared to that observed at higher temperatures for Type 304 stainless steel [10]. (Note that the dissolved oxygen concentration increases from 0.2 ppm for constant extension rate tests [CERTs] at 392 and 550°F [200 and 288°C] to 1.8 ppm for the 257°F [125°C] tests to simulate higher dissolved oxygen at lower temperatures and to facilitate SCC.) Even higher strains to initiate SCC would be required at the low Turkey Point Unit 3 RWST's low temperatures.

The strain required to initiate SCC is also affected by the impurity present in the environment. This effect is illustrated in Figure 3-2 where the amount of strain to initiate SCC as generated in a CERT study on welded plus low temperature sensitized Type 304 stainless steel is plotted for various impurities at 550°F (288°C) as a function of pH [11]. Unfortunately, neither low temperature data nor the specific effect of boric acid on the strain to initiate SCC is available in this plot. While the strains required to initiate SCC at low temperature in high purity environments are indeed very high, such high strains can be the result of welding as shown in Figure 3-3 [12].

3.2 SCC Propagation Rates at Low Temperatures in Wrought Stainless Steel

If a crack can initiate due to very high weld residual strains or if a pre-existing flaw is present in the component, the subsequent crack growth rate can be significant. Figure 3-4 presents the reversing DC potential drop (DCPD) technique crack growth rates of furnace sensitized Type 304 stainless steel in high purity water environments as a function of temperature. The overall plot reveals that the crack growth rate of Type 304 stainless steel reaches a maximum crack growth rate at approximately 300 to 390°F (150 to 200°C) in high purity water [13]. The observed maximum in IGSCC growth is attributed to two competing effects: the increase in growth rate vs. temperature from increasing kinetics of mass transport and the decrease in growth rate vs. temperature from the decrease in corrosion potential due to decreasing dissolved oxygen content.

While Table 3-1 summarized all the low temperature crack growth rate data from this study [13], the crack growth rate of greatest relevance for the Turkey Point Unit 3 RWST evaluation is the data point obtained on a furnace sensitized Type 304 stainless steel fracture mechanics compact tension specimen tested at a stress intensity of 30 ksi√in (33 MPa√m) and exposed to an environment at close the Turkey Point Unit 3 RWST's operating temperature of 100°F (38°C) with a calculated dissolved oxygen content of 6.75 ppm. The 0.270 µS/cm crack growth rate is considered conservative because it was produced by strong acid anion sulfate that is stable in highly reducing crack tip environments. As illustrated in Figure 3-4, the crack growth rate under these environmental and stress conditions on wrought sensitized austenitic stainless steel is significant at 1 x 10^{-3} mm/h or 345 mpy.

3.3 Effect of Stainless Steel Microstructure on SCC

It is very important to note that all of the above testing results and discussion have been based on studies performed on in wrought stainless steels and not stainless steels with a duplex microstructure (e.g., austenite and delta ferrite) that is present in cast stainless steels. Duplex stainless steels such as casting and weld metals are dramatically more resistant to SCC than wrought stainless steels [14].

For example, Figure 3-5 presents the effect of chloride and temperature on the SCC propensities of wrought Types 304 and 316 austenitic stainless steels compared to a duplex austenitic and ferritic stainless steel SAF 2304 (0.03% max C, 22.5% Cr, 4.5% Ni nominal) [14]. The duplex microstructure of cast stainless steel such as Turkey Point Unit 3's CF8 would require higher chloride content and temperature to produce SCC. In this particular example, the duplex stainless steel cracks at an aggressive combination of 0.001% or 10 ppm or 10,000 ppb chloride (compared to an average of 12.97 ppb chloride at Turkey Point Unit 3) and at 302°F (compared to the approximately 100°F operating temperature at Turkey Point Unit 3).

Table 3-1. Summary of Low Temperature Crack Growth Rate Data on Type 304 Stainless Steel [13]

Type 304 Heat	Temp., °F (°C)	Dissolved Oxygen, ppm	Impurity, μM	Conductivity, μS/cm	Crack Growth Rate, mm/h	Crack Growth Rate, mpy
AJ9139	77 (25)	0.2	0.3 SO_4	0.270	7.7×10^{-4}	266
2P4932	77 (25)	0.2	0.3 SO_4	0.270	2.0×10^{-3}	690
71635	77 (25)	0.2	0.3 SO_4	0.270	1.2×10^{-3}	421
71635	77 (25)	8.8	0.3 CO_3	0.580	1.0×10^{-3}	345

SO_4 is from the strong acid H_2SO_4

CO_3 is from the weak acid H_2CO_3 from dissolved CO_2 in air saturated water

$K = 30$ ksi√in (30 MPa√m)

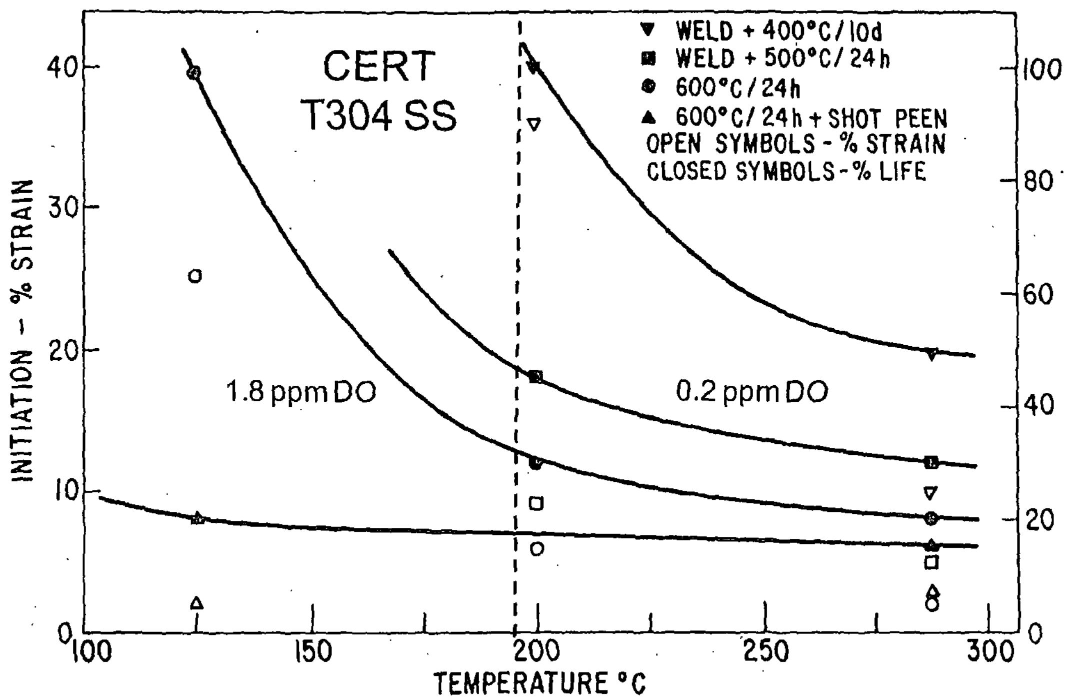

Figure 3-1. Effect of Temperature on the Strain Required for Crack Initiation for Welded plus Low Temperature Sensitized and Furnace Sensitized Type 304 Stainless Steels [10]

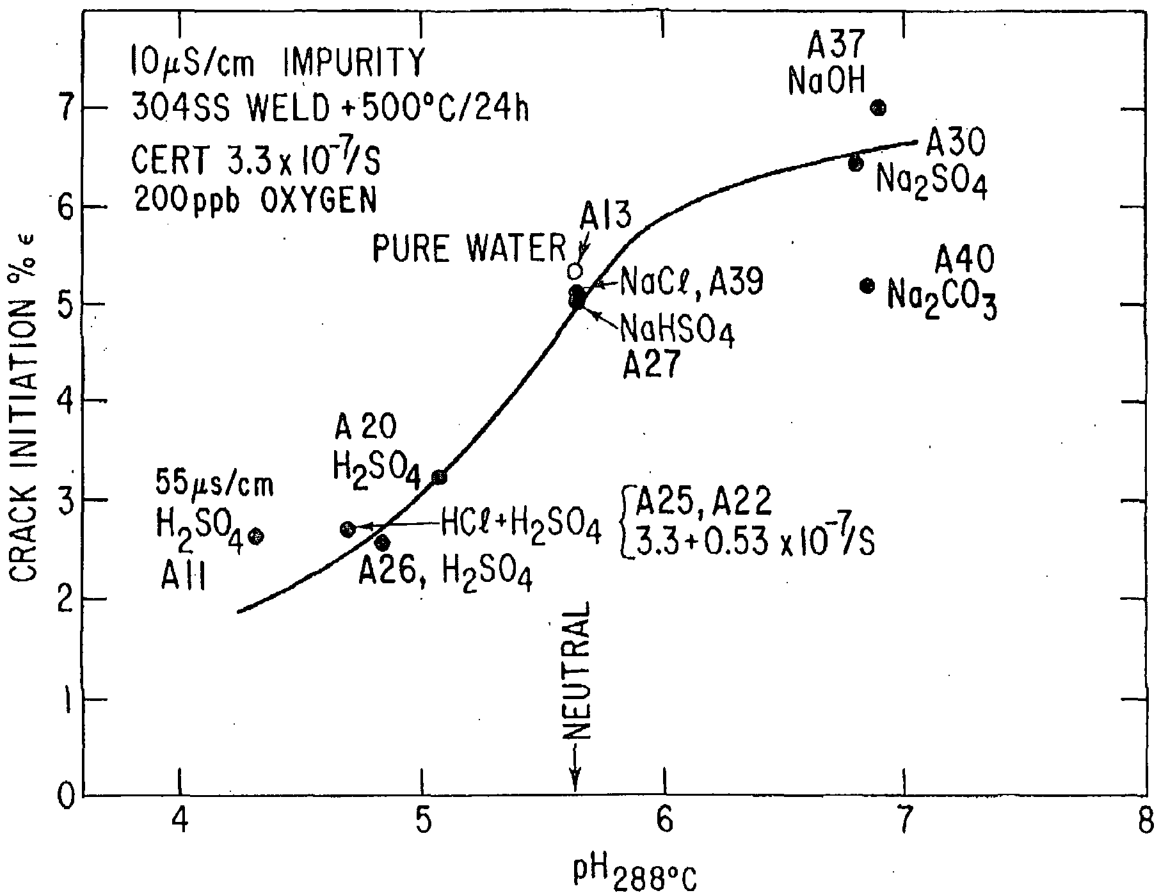

Figure 3-2. Effect of Impurity on the Strain for Crack Initiation for Welded plus Low Temperature Sensitized Type 304 Stainless Steels at 550°F (288°C) [11]

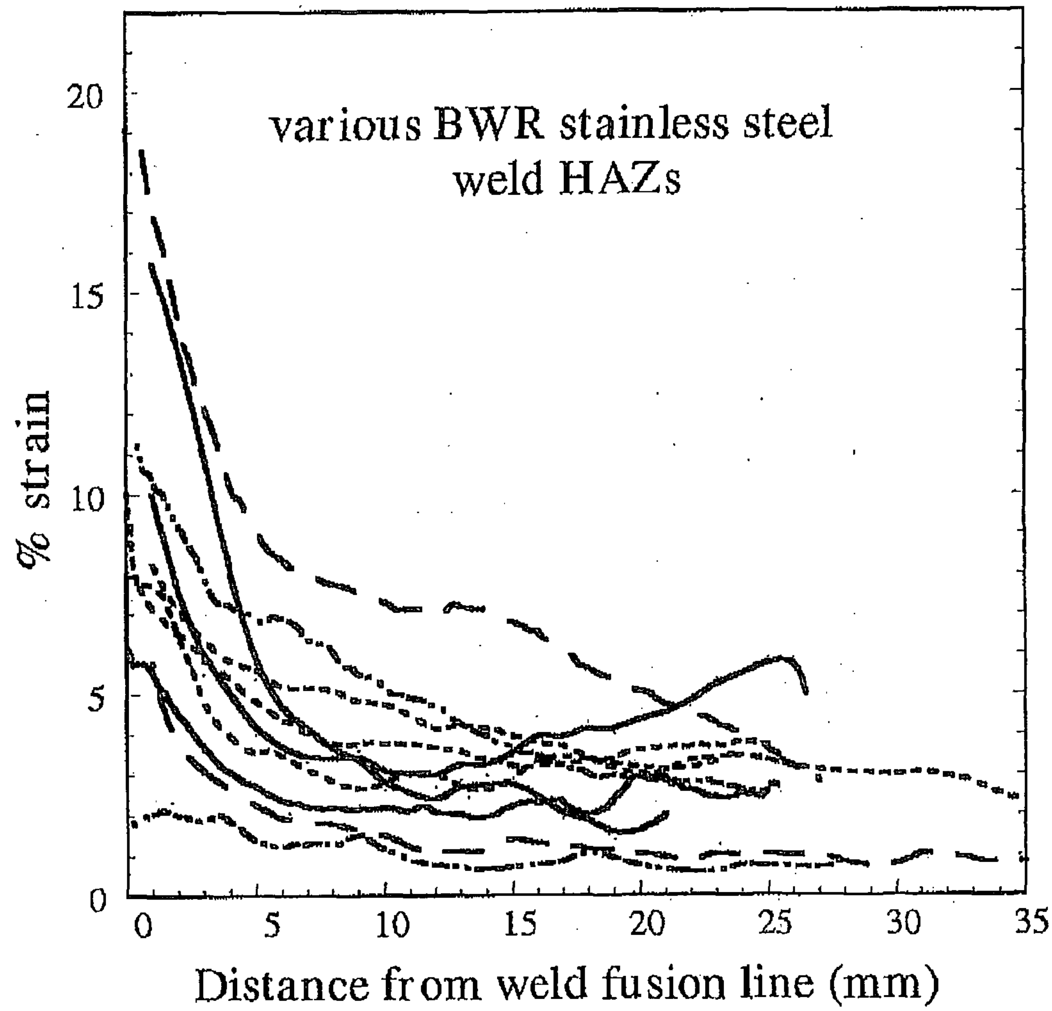

Figure 3-3. Residual Strain from Weld Shrinkage Strain in the HAZ of Stainless Steels Deformation is expressed in Terms of Equivalent Tensile Strain at Room Temperature, and Results from Shrinkage Strains during Welding [12]

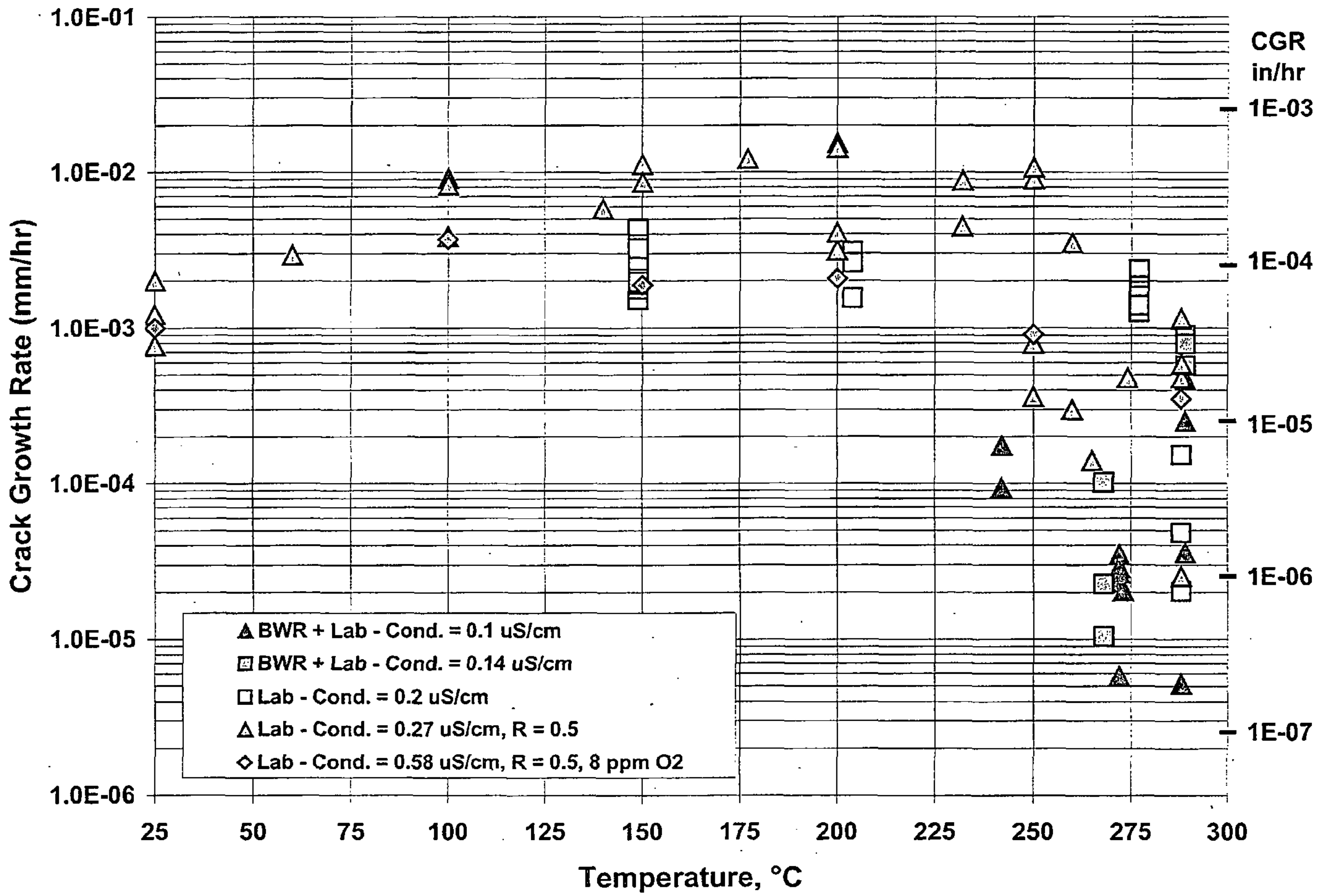

Figure 3-4. Effect of Temperature on Crack Growth Rate for Type 304 Stainless Steel in Impure Water Environments [13]
(BWR + Lab = BWR and lab crack growth rate data at the indicated conductivity)

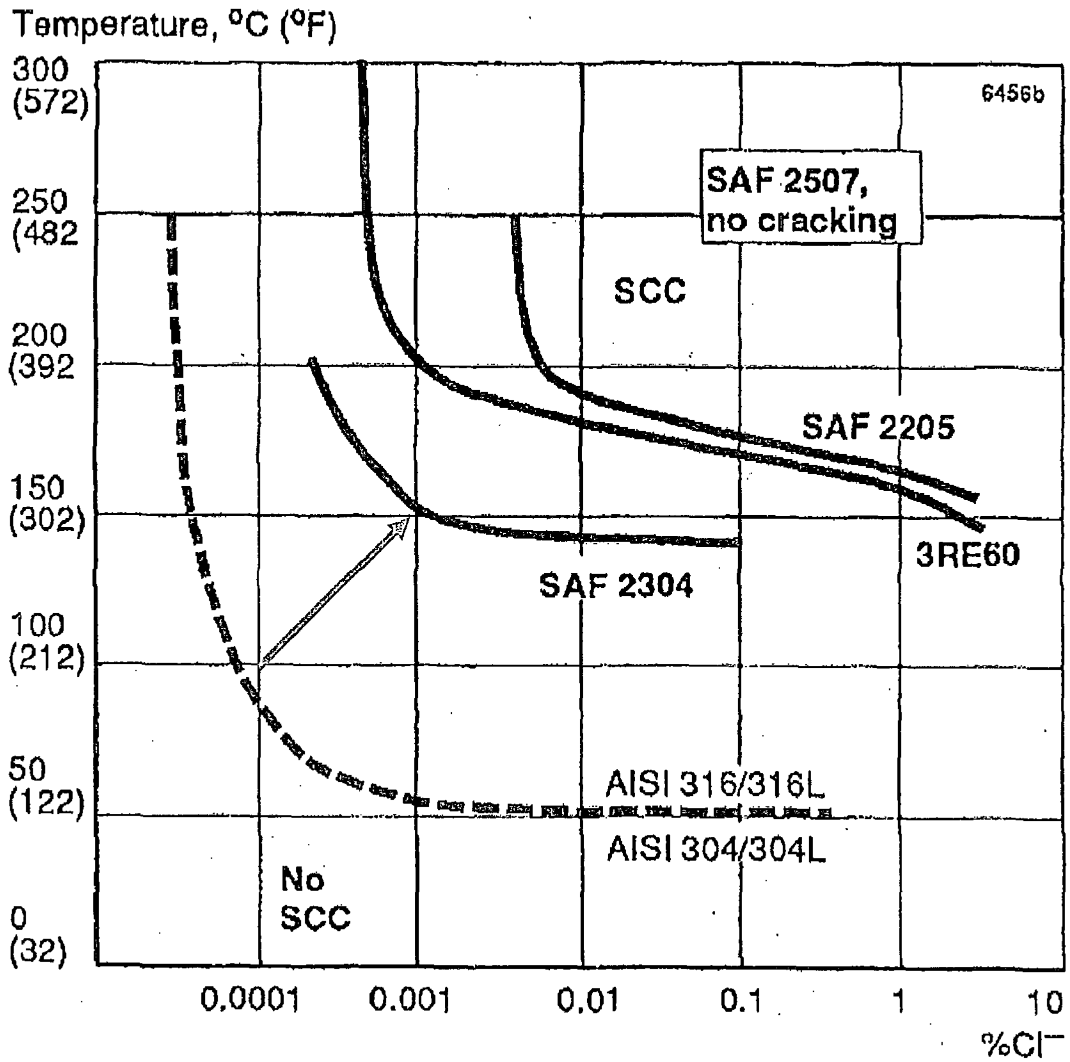

Figure 3-5. Example of Improvement in SCC Resistance between Cast Duplex SAF 2304 and Wrought Type 304 and 316 Stainless Steel [14]

4.0 SUMMARY AND CONCLUSIONS

The PWR operating experience and laboratory studies suggest that it is the presence of impurities such as fluoride, chloride and sulfate in the water that facilitate SCC of wrought Type 304 stainless steel in borated water and may have a greater detrimental affect than the presence of boric acid itself. However, the above discussion and present results also demonstrate that while SCC initiation in sensitized Type 304 stainless steel is difficult at low temperatures, SCC propagation can clearly occur in low temperature oxygenated environments at stress intensities that reflect those that are calculated for typical reactor components, Table 3-1. The key difference here is that the Turkey Point Unit 3 valve 3-844A is fabricated from highly SCC resistant cast CF8 stainless steel that would require a significantly more aggressive environment to support the SCC mechanism. Therefore, it is highly unlikely that the flaw would advance at a rate of any engineering significance over the next fuel cycle by any corrosion related mechanism.

It is considered prudent to evaluate only non-EAC factors for the leak of the 3-844A valve (e.g., manufacturing defect, porosity). Only the planned destructive examination of the valve can be used to determine the mechanism responsible for the flaw and subsequent leak in this component. Regardless, it is not anticipated that the flaw will propagate by any corrosion mechanism and the current leakage rate associated with this flaw will remain constant. This conclusion includes the possibility of the temperature of the valve reaching its maximum anticipated accident temperature of 205°F (96°C) since corrosion is not considered as a viable degradation mechanism of this component.

5.0 REFERENCES

1. M. Toner e-mail attachment to R. McGill, et al., "AR 1904263 Input for IOD.docx," September 17, 2013, SI File Number 1301208.201 (reproduced in Appendix A).
2. U. S. Nuclear Regulatory Commission, "Pipe Cracks in Stagnant Borated Water Systems at PWR Plants," IE Information Notice No. 79-19, July 17, 1979.
3. U. S. Nuclear Regulatory Commission, "Stress Corrosion Cracks in Stagnant, Low Pressure Stainless Piping Containing Boric Acid Solutions at PWRs," IE Circular No. 76-06, November 26, 1976.
4. "Pipe Cracking in Pressurized Water Reactors with Low-Pressure Borated Water Systems," EPRI NP-3320, Palo Alto, CA, December 1983.
5. F. S. Giacobbe, "Examination, Evaluation and Repairs of Stress Corrosion Cracking in a PWR Borated Water Piping System," paper #164 presented at Corrosion 81, Toronto, Ontario, April 6-10, 1981.
6. J. C. Griess and G. E. Creek, "The Stress Corrosion Cracking of Types 304 and 316 Stainless Steel in Boric Acid Solutions," ORNL-TM-2412, May 1971.
7. T. Tsuruta and S. Okamoto, "Stress Corrosion Cracking of Sensitized Austenitic Stainless Steels in High Temperature Water," Corrosion, Vol. 48, No. 6, NACE, Houston, TX, June 1992, p. 518.
8. M. Toner e-mail attachment to R. McGill, et al., "3-844a Chem-Cycles.pdf," September 18, 2013, SI File Number 1301208.201 (reproduced in Appendix B).
9. http://antoine.frostburg.edu/chem/senese/101/solutions/faq/predicting-DO.shtml
10. P. L. Andresen, "Crack Initiation in CERT Tests on Type 304 Stainless Steel in Pure Water," Corrosion, Vol. 38, No. 1, January 1982, p. 53.
11. P. L. Andresen, "The Effects of Aqueous Impurities on Intergranular Stress Corrosion Cracking of Sensitized Type 304 Stainless Steel," EPRI NP-3394, Palo Alto, CA, November 1983.
12. "Advanced Testing Techniques to Measure the PWSCC Resistance of Alloy 690 and its Weld Metals," EPRI, Palo Alto, CA and U.S. Department of Energy, Washington, DC: 2004. 101 1202.
13. "BWRVIP-186: Effect of Water Chemistry and Temperature Transients on the IGSCC Growth Rates in BWR Components," EPRI, Palo Alto, CA 1016485. 2008.
14. S. Bernhardsson, "The Corrosion Resistance of Duplex Stainless Steels," paper presented at the conference Duplex Stainless Steels '91, Beaune, France, October 1991.

APPENDIX A

ENGINEERING SUPPORT STATEMENT FOR IMMEDIATE OPERABILITY DETERMINATION (REFERENCE 1)

Engineering Support Statement for Immediate Operability determination of AR 1904263

NDE inspection performed on 9/17/13 after cleaning off the boric acid buildup identified a void in the valve bonnet above the packing leakoff line (see photo attached in EDMS). No observable leakage was present. The void appears to be an original casting defect. The two primary concerns for this indication are fluid inventory control and structural integrity of the flaw. Valve 3-844A is presently exposed to static head of the RWST, and is pressurized to post-accident piggy-back pressures every outage via the ECCS suction hyrdrostatic leak test via 3-OSP-202.4. The specific location of the flaw is in the valve bonnet above the packing, and is additionally isolated from system pressure via the valve backseat per 3-NOP-068. As an original casting defect, this flaw was present during the most recent test near the beginning of PT3-26 RFO in spring 2012 per 3-OSP-202.4 with satisfactory results. Therefore, it is reasonable to assume that any potential leakage in the post-accident piggy-back alignment is within UFSAR limits for ECCS leakage outside of containment.

The casting void noted herein appears localized to the area of the bonnet just under the bonnet to yoke flange. Typically, the stresses are low at this location for manual valves that remain locked open. Previous flaw evaluations of 4" stainless steel pipes have allowed more than half of the circumference of a pipe (Ref AR 570552 POD Rev 2). Per the attached photographs and NDE inspection, the current flaw is significantly less than half of the bonnet circumference in this location. As such, there is reasonable assurance that the immediate condition is operable. A formal Prompt Operability Determination is required to validate this Immediate Operability Determination.

APPENDIX B

WATER CHEMISTRY DATA (REFERENCE 8)

Tour Name: Primary: SAMPLES: U3 RWST (DATA SHEET)   Printed at 9/18/2013 9:01:21 AM

Tag Name	Tag Description	Value	Comment	TimeStamp
PTN:3CHEM:PRI:RWST:Boron	U3 Refueling Water Storage Tank (RWST): Boron = "3R:WST-B O	2469 ppm		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:pH	U3 Refueling Water Storage Tank (RWST): pH = "3R:WST-P H"	4.22		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:SC	U3 Refueling Water Storage Tank (RWST): Specific Conductivity [	8.68		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Silica	U3 Refueling Water Storage Tank (RWST): Silica (SiO2) = "3R:W			9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Fluoride	U3 Refueling Water Storage Tank (RWST): Fluoride = "3R:WST-F	2.77 ppb		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Chloride	U3 Refueling Water Storage Tank (RWST): Chloride = "3R:WST-C	12.97 ppb		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Sulfate	U3 Refueling Water Storage Tank (RWST): Sulfate = "3RWST SO	1 ppb		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Sodium	U3 Refueling Water Storage Tank (RWST): Sodium			9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Potassium	U3 Refueling Water Storage Tank (RWST): Potassium			9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Magnesium	U3 Refueling Water Storage Tank (RWST): Magnesium = "3RWS	1 ppb		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Calcium	U3 Refueling Water Storage Tank (RWST): Calcium = "3RWST C	1 ppb		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Aluminium	U3 Refueling Water Storage Tank (RWST): Aluminium = "3RWST			9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:Turbidity	U3 Refueling Water Storage Tank (RWST): Turbidity = "3R:WSTT	0.193		9/16/2013 11:15:00 AM
PTN:3CHEM:PRI:RWST:SS	U3 Refueling Water Storage Tank (RWST): Suspended Solids (SS			9/16/2013 11:15:00 AM

Locked OPEN & Backseated 7/25/12
Closed 7/20/12
Locked OPEN & Back Seated 7/10/12
Closed 4/12/12
Locked OPEN & Backseated 3/1/12
Closed 2/29/12
L O B ~~2/3/12~~ 2/3/11