Progress Energy

SEP 2 9 2011

SERIAL: BSEP 11-0093

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: Brunswick Steam Electric Plant, Unit No. 2
Renewed Facility Operating License No. DPR-62
Docket No. 50-324
Response to Request for Additional Information Regarding Reactor Pressure Vessel Shell-to-Flange Indication Analytical Evaluation (NRC TAC No. ME6033)

Reference: Letter from Phyllis N. Mentel (CP&L) to U.S. NRC Document Control Desk, "Reactor Pressure Vessel Shell-to-Flange Analytical Evaluation," April 13, 2011, ADAMS Accession Numbers ML111101020 and ML11110A022)

Ladies and Gentlemen:

By letter dated April 13, 2011, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., submitted a request for NRC review and approval of an analytical evaluation for an indication identified in the Brunswick Steam Electric Plant (BSEP), Unit No. 2 reactor pressure vessel shell-to-flange weld. On August 2, 2011, via electronic mail, the NRC provided a request for additional information (RAI) regarding the analytical evaluation. The response to the RAI is provided in Enclosure 1 to this letter.

No regulatory commitments are contained in this letter. Please refer any questions regarding this submittal to Ms. Annette Pope, Supervisor - Licensing/Regulatory Programs, at (910) 457-2184.

Sincerely,

Phyllis N. Mentel

Phyllis N. Mentel
Manager - Support Services
Brunswick Steam Electric Plant

A001
NRR

Progress Energy Carolinas, Inc.
Brunswick Nuclear Plant
P.O. Box 10429
Southport, NC 28461

WRM/wrm

Enclosures:

1. Response to Request for Additional Information
2. Report No. 110470.401, Rev. 2, "Brunswick Steam Electric Plant Unit 2 Reactor Pressure Vessel Flaw Evaluation," Structural Integrity Associates, Inc.

cc (with enclosures):

U. S. Nuclear Regulatory Commission, Region II
ATTN: Mr. Victor M. McCree, Regional Administrator
245 Peachtree Center Ave, NE, Suite 1200
Atlanta, GA 30303-1257

U. S. Nuclear Regulatory Commission
ATTN: Mr. Philip B. O'Bryan, NRC Senior Resident Inspector
8470 River Road
Southport, NC 28461-8869

U. S. Nuclear Regulatory Commission **(Electronic Copy Only)**
ATTN: Mrs. Farideh E. Saba (Mail Stop OWFN 8G9A)
11555 Rockville Pike
Rockville, MD 20852-2738

Chair - North Carolina Utilities Commission
P.O. Box 29510
Raleigh, NC 27626-0510

Mr. Jack M. Given, Jr., Bureau Chief
North Carolina Department of Labor
Boiler Safety Bureau
1101 Mail Service Center
Raleigh, NC 27699-1101

Response to Request for Additional Information

By letter dated April 13, 2011, Carolina Power & Light Company (CP&L), now doing business as Progress Energy Carolinas, Inc., submitted a request for NRC review and approval of an analytical evaluation for an indication identified in the Brunswick Steam Electric Plant (BSEP), Unit No. 2 reactor pressure vessel shell-to-flange weld. On August 2, 2011, via electronic mail, the NRC provided a request for additional information (RAI) regarding the analytical evaluation. The response to the RAI follows.

NRC Question 1

What was the date during which the indication was identified and sized under PE Contract 66325, Work Authorization 170?

CP&L Response

Indication was identified and sized on March 8, 2011.

NRC Question 2

The evaluation notes that pre-operational inspection data showed that small reportable indications were identified near the location of the current indication of interest. Please:

a. Identify when the pre-operational examinations were conducted,

b. Provide the dimensions of the all reportable indications identified during the preoperational examinations which were located near the current indication of interest.

CP&L Response

The pre-operational examinations were conducted on June 13, 1972.

The dimensions of the indication of interest are provided in the section titled "Flaw Evaluation Results" of Report No. 110470.401, Rev. 2, "Brunswick Steam Electric Plant Unit 2 Reactor Pressure Vessel Flaw Evaluation." A copy of the report is provided in Enclosure 2.

No other indications were identified as being located near the current indication of interest.

NRC Question 3

Clarify whether the bending stress is compressive or tensile at the location of the indication.

CP&L Response

The bending stress applied to the flaw is tensile. Since most of the applicable load cases resulted in compressive bending stresses near the inner diameter surface, the largest tensile bending stress is selected for this evaluation (i.e., 2 ksi from the shutdown load case).

NRC Question 4

Confirm whether the indication being evaluated is closer to the inside diameter or the outside diameter of the reactor vessel.

CP&L Response

The indication is closer to the inside surface of the reactor pressure vessel.

NRC Question 5

Clarify why the smallest bending stress was chosen from Attachment 1 to the submittal.

CP&L Response

As stated in the response to Question 3, the bending stress applied to the flaw is tensile. Since most of the applicable load cases resulted in compressive bending stresses near the inner diameter surface, the largest tensile bending stress is selected for this evaluation (i.e., 2 ksi from the shutdown load case).

NRC Question 6

The staff was unable to verify the calculation of KIC_70. The staff independently calculated a KIC_70 of 102 ksi-$in^{0.5}$ while the attached analysis reports a KIC_70 of 94.7 ksi-$in^{0.5}$. Please confirm which value is correct, and if the attachment calculation is in error, how this occurred.

CP&L Response

The calculated KIC_70 of 102 ksi-$in^{0.5}$ is correct. The exponential function $e^{(0.02(70-10))}$ was not evaluated correctly in MathCAD. Consequently, a value of K_{Ic} was determined, which is approximately 7% lower than the correct value. This error resulted in a conservatively lower estimate of the material fracture toughness. This has no effect on the conclusions of the calculation since the bounding allowable fracture toughness was determined from the upper bound K_{IC} taken at 550 °F.

Report No. 110470.401, Rev. 2, "Brunswick Steam Electric Plant Unit 2 Reactor Pressure Vessel Flaw Evaluation," correcting the error, is provided in Enclosure 2.

NRC Question 7

ASME Code, Section XI, IWB-3612 states that for normal conditions $K_I < K_{Ia}/10^{0.5}$ must be true. The analysis verifies that $K_I < K_{Ic}/10^{0.5}$ is true. Justify the use of K_{Ic} instead of K_{Ia} in determining whether this criterion is met.

CP&L Response

Use of K_{Ic}, rather than K_{Ia}, is an appropriate fracture toughness to use in an analysis performed to assess flaw stability under slow strain rate loading. The ASME Code and the NRC have acknowledged this through acceptance of Code Case N-640, *Alternative Reference Fracture Toughness for Development of P-T Limit Curves, Section XI, Division 1*, and subsequently, through incorporation of acceptance criteria for flaws in ferritic components in the ASME Code, Section XI, IWB-3600 and Non-mandatory Appendix G, which are based upon the slow strain rate fracture toughness, K_{Ic}, rather than the crack arrest toughness, K_{Ia}.

As stated in the letter dated April 13, 2011, the applicable code of record for the fourth 10-year inservice inspection interval for Unit 2 is the 2001 Edition of the ASME Code, Section XI, with 2003 Addenda. The 2001 Edition through 2003 Addenda was cited in the original calculation; however, the acceptance criteria of the NRC-approved 2007 Edition were applied in the original calculation.

Rather than request NRC approval to use the 2007 edition of the ASME Code, the flaw evaluation has been revised to use K_{Ia} rather than K_{Ic}. A revised flaw evaluation which uses K_{Ia} rather than K_{Ic} is contained in Report No. 110470.401, Rev. 2, "Brunswick Steam Electric Plant Unit 2 Reactor Pressure Vessel Flaw Evaluation." A copy of the revised report is provided in Enclosure 2.

Report No. 110470.401, Rev. 2
"Brunswick Steam Electric Plant Unit 2
Reactor Pressure Vessel Flaw Evaluation,"
Structural Integrity Associates, Inc.

Structural Integrity Associates, Inc.

10731 E. Easter Avenue, Suite 100
Centennial, CO 80112-3765
Phone: 303-792-0077
Fax: 303-792-2158
www.structint.com
dsommerville@structint.com

September 20, 2011
Report No. 1100470.401 Rev. 2
Quality Program: ☒ Nuclear ☐ Commercial

Mr. John Becker
Progress Energy
Brunswick Nuclear Plant
8470 River Road SE
Southport, NC 28461-8869

Subject: Brunswick Steam Electric Plant Unit 2 Reactor Pressure Vessel Flaw Evaluation

Dear Mr. Becker:

This letter report documents the results of a flaw evaluation of an indication detected in the Brunswick Steam Electric Plant Unit 2 (BSEP U2) reactor pressure vessel (RPV) circumferential weld joining the vessel closure flange to the adjacent shell.

Revision 1 of this letter report incorporates correction of an error identified in the MathCAD calculations in which the exponential function used to determine K_{IC} did not evaluate correctly. The error resulted in a conservatively lower estimate of allowable fracture toughness and occurred in the non-limiting load combination; therefore, this error had no effect on the overall conclusions presented in the Rev. 0 letter report. For completeness, the error was corrected in the revision 1 letter report, and was identified by a revision bar in the left hand margin of page 11 of 12.

Revision 2 of this letter report uses flaw stability acceptance criteria based upon the crack arrest fracture toughness, K_{Ia}, curve rather than the crack initiation fracture toughness, K_{Ic}, curve. Although use of K_{Ic} fracture toughness data is technically appropriate and consistent with currently accepted industry methods for RPV flaw evaluations, use of the K_{Ia} curve is consistent with the rules contained in the edition of the ASME Boiler and Pressure Code, Section XI, licensed for the current in-service inspection interval at BSEP U2. This revision affects pages 3 and 9 through 13 of 13. The conclusions made in the original, Revision 0, calculation remain valid and unchanged.

INTRODUCTION

An indication was identified in the circumferential weld which joins the RPV closure flange forging to the adjacent shell, during in-service inspections (ISI). Progress Energy (PE) contracted Structural Integrity Associates, Inc. (SI), in PE Contract 66325, Work Authorization 170, to perform a flaw evaluation of the reportable indication using methods consistent with ASME XI, IWB-3600 [1].

METHODOLOGY

The evaluation documented in this report is performed using the methods of ASME XI, IWB-3600 and ASME XI, Non-mandatory Appendix A [1]. Since the methods are described in detail in Reference [1] they are not repeated here.

DESIGN INPUT

The following design input and documentation was provided by PE in support of this evaluation:

1. Pre-operation inspection data [2],
2. B220R1 Inspection Data [3],
3. BSEP Main Closure Flange (MCF) stress analysis [4],
4. Evaluation Interval [5]
   a. End of current licensed life = December 27, 2034
   b. End of original licensed life = December 27, 2014
5. Main Closure Flange forging material initial RT_{NDT} [10]

EVALUATION

This section documents the key assumptions and results of the flaw evaluation. Attachment 1 contains a summary of the stresses extracted from the original stress report [4]. Attachment 2 contains the detailed calculations performed for this flaw evaluation.

Assumptions:

The following assumptions are made for this evaluation:

1. The reportable indication is assumed to be located at the location of highest stresses reported in the BSEP MCF stress report (at 41.5 inches below the upper surface of the vessel closure flange forging).
2. The stress cycles considered for the remainder of the plant licensed life are determined by scaling the number of cycles defined on the Thermal Cycle Diagram [6] by the following factor:

$$\#Cycles_{EOL} = \frac{Year_{EOL} - Year_{Current}}{40} \cdot \#Cycles_{40Years} = \frac{34-10}{40} \cdot 458 = 275$$

3. A conservative R ratio of 1 is used to calculate anticipated fatigue crack growth (FCG) through the end of the evaluation interval. This value is conservative since it maximizes the FCG calculated. Since the flaw is subsurface no other crack growth mechanisms need to be considered in this evaluation.
4. An 8 ksi cosine distribution consistent with that considered in References [7, 8] is assumed for the weld residual stress distribution.

5. All stresses, except for the weld residual stress, are conservatively scaled by a scaling factor defined as the largest ratio of power uprate to pre-power uprate pressures identified in the power uprate design specification [9]. This is shown on page 1 of the calculation contained as Attachment 2. The scale factor is considered conservative since it is applied to both thermal and pressure stresses and it is calculated by taking the largest increase in pressure reported in the design specification [9] for a single point in time for a single transient but applied uniformly for all load cases.

Initial Flaw Size:

The initial flaw size is taken from the inspection report [3] and is summarized below in the Results section. The pre-operational inspection data [2] shows that small reportable indications were identified during the pre-operational examinations. Current inspection methods are expected to result in more accurate sizing; therefore the original and current inspection results are not expected to match.

Loads:

Attachment 1 summarizes the results of the stress analysis contained in Reference [4] taken at the bounding location in the vessel closure flange. The reference [4] analysis considered all transients defined in the RPV thermal cycle diagram [6]. Residual stresses are taken from References [7, 8].

Flaw Evaluation Results:

The results of the evaluation documented in Attachment 2 are summarized below for convenience:

a_0 =	0.225 in.	Initial flaw depth, total through-wall dimension is $2a_0$
l_o =	6.4 in.	Initial flaw length
n =	275 cycles	Total load cycles through 60 years
ΔK =	2.1 ksi-in$^{0.5}$	Largest range of stress intensity factor for all load cycles
Δa =	2.0×10^{-6} in	End of evaluation interval growth in flaw depth, total through-wall flaw growth is $2\Delta a$
Δl =	4.0×10^{-6} in	End of evaluation interval growth in flaw length
a_f =	0.225 in	End of evaluation interval flaw depth, total through-wall dimension is $2a_f$
l_f =	6.40 in	End of evaluation interval flaw length
K_I =	9.5 ksi-in$^{0.5}$	Applied stress intensity factor at end of evaluation interval
$K_{I\ Allowable}$ =	23 ksi-in$^{0.5}$	Allowable stress intensity factor

CONCLUSIONS

The results of this evaluation support the following conclusions:

1. The indication reported in References [2, 3] is likely a fabrication induced flaw; therefore, it is considered to have been present in this component for the life of the plant and is not the result of new crack initiation.
2. The reported indication is acceptable per the methods of ASME XI, IWB-3600 [1]; therefore, it may be left as-is for operation through the end of the plant licensed life.

REFERENCES

1. ASME Boiler and Pressure Vessel Code, Section XI, 2001 Edition with Addenda through 2003.
2. Preoperational Inspection Data attached to email from Damon Priestly (PE) to Daniel Sommerville (SI) dated 3/16/2011. SI File No. 1100470.203.
3. Nuclear Generation Group UT Report UT-11-001, SI File No. 1100470.203.
4. Carolina Power and Light Company Calculation No. 0B11-0023, Rev. 0, RPV Stress Report, SI File No. 1100470.201.
5. Email containing BSEP Unit 2 license expiration date, sent from Larry Yemma (PE) to Daniel Sommerville (SI), SI File No. 1100470.204.
6. GE Dwg. 729E762, Rev. 0, "Reactor Thermal Cycles," SI File No. CPL-35Q-245.
7. White Paper on Reactor Vessel Integrity Requirements for Level A and B Conditions, EPRI TR-100251, January 1993.
8. BWRVIP-60-A: BWR Vessel and Internals Project, Evaluation of Stress Corrosion Crack Growth in Low Alloy Steel Vessel Materials in the BWR Environment, EPRI, Palo A lot, CA: 2003. 1008871.
9. Reactor Vessel – Power Uprate, Design Specification 25A5062, Rev. 1, SI File No. CPL-61Q-205P.
10. Stevens, Gary L., "Revised Brunswick Pressure-Temperature Curves," SIR-99-015, Rev. 0.

Prepared by:

9/20/2011

Daniel Sommerville, P.E.
Associate

Date

Reviewed by:

9/20/2011

David Dijamco
Consultant

Date

Approved by:

9/20/2011

Daniel Sommerville, P.E.
Associate

Date

Attachment 1:
Stress Summary

	Original Stress Report [4]							
Event	Pressure, psi	Temperature, F	ID, psi	OD, psi	σ_m (4), ksi	σ_b (5), ksi	σ_{Res} (1), ksi	Reference
Hydrotest	1250	150	-8367	33481	12.557	20.924	8	[4, 50-S1]
Preload	0	70 (2)	-13274	14258	0.492	13.766	8	[4, 53-S1]
Startup	1005	546 (2)	-26003	46347	10.172	36.175	8	[4, 57-S1]
Shutdown	40	100 (2)	2393	-1921	0.236	2.157	8	[4, 61-S1]
Rapid Cooldown	40	100 (2)	-10628	11798	0.585	11.213	8	[4, 65-S1]
Steady State	1005	546	-9476	29672	10.098	19.574	8	[4, 69-S1]
Overload	1375	546	-7808	35404	13.798	21.606	8	[4, 73-S1]

Notes:
1. Assumed 8 ksi with cosine distribution as reported in [6, 7]
2. Assumed value
3. Location is 41.5 inches below surface of RPV closure flange surface (near centerline of flange to shell circumferential weld)
4. Membrane stress is calculated as (ID Stress + OD Stress) / 2, Using Hydrotest as an exampe, σ_m=(-8367+33481)/2 = 12557 psi
5. Bending stress is calculated as OD Stress -σ_m, Using Hydrotest as an exampe, s_b=33481-12557 = 20924 psi

Event	Cycles
Hydrotest @ 1250 psi	130
Startup-Shutdown	120
SCRAM	208
Hydrotest @ 1563 psi (1)	2
	458

Notes:
1. Excluded from fatigue crack growth calculation since no shop hydrotest expected in remaining life.
2. Taken from [4, 22-F1]

Attachment 2:
Flaw Evaluation

Input Data:

$a := \frac{0.45}{2}$	in	Flaw half depth, See ASME XI, Appendix A, Fig. A-3300-1(a). Rounded up from 0.43 inches reported from NDE.
$l := 6.4$	in	Flaw length, See ASME XI, Appendix A, Fig. A-3300-1(a).
$S := 0.35$	in	Distance between Low Alloy Steel (LAS) surface and flaw tip, See ASME XI, Appendix A, Fig. A-3300-1(a). Rounded down from 0.367 reported from NDE.
$t := 6.1$	in	Vessel shell thickness, excluding clad.
$e := \frac{t}{2} - S - a$	in	Flaw eccentricicty, See ASME XI, Appendix A, Fig. A-3300-1(a)

$e = 2.475$ in

$sys\_70 := 50$	ksi	Yield strength of material at 70 F. for both SA-533, Gr. B, Cl. 1 and SA-508 to Code Case 1332 Par. 5 (Essentially yield strength consistent to SA-302 Gr. B)
$sys\_550 := 42.6$	ksi	Yield strength of material at 550 F, see comment above for materials

A bounding scaling factor is calculated using the largest increase in pressure defined in paragraph 4.4.1 of the Power Uprate Design Specification

$$SF := \frac{700}{665} \qquad SF = 1.053$$

$$sm := SF \begin{pmatrix} 12.557 \\ 0.492 \\ 10.172 \\ 0.236 \\ 0.585 \\ 10.098 \\ 13.798 \end{pmatrix} + 8$$

ksi, membrane stress taken from stress report [4] scaled by bounding scaling factor described above and summed with assumed residual stress [7,8].

$$sm^T = (21.22 \;\; 8.52 \;\; 18.71 \;\; 8.25 \;\; 8.62 \;\; 18.63 \;\; 22.52)$$

$$sb := \begin{pmatrix} -20.924 \\ -13.766 \\ -36.175 \\ SF \cdot 2.157 \\ -11.213 \\ -19.574 \\ -21.606 \end{pmatrix}$$

ksi, bending stress taken from stress report [4] scaled by bounding scaling factor described above. All bending stresses which are compressive on the flaw are not treated with SF (this is conservative).

$$sb^T = (-20.92\ \ -13.77\ \ -36.17\ \ 2.27\ \ -11.21\ \ -19.57\ \ -21.61)$$

$RTNDT := 10$ F Highest initial RTNDT for closure flange materials. See Reference [10].

Determine M_m and M_b from Fig. A-3310-1 and Fig. A-3310-2

$2\frac{e}{t} = 0.811$ $\frac{2a}{t} = 0.074$

$Mm := 1.1$ Conservatively taken as 1.1 since 2a/t and 2e/t not on Fig. A-3310-1

$Mb := 1$ Conservatively taken as 1 since 2a/t and 2e/t not on Fig. A-3310-2

Proximity Check (See criterion on ASME XI, Fig. IWA-3320-1):

$0.4a = 0.09$ $S > 0.4a$ Therefore, flaw is subsurface

Calculate plastic zone size correction (See ASME XI, Appendix A, Eq. (4)):

$i := 0, 1 .. 6$

$$qy\_550_i := \frac{\left(\frac{sm_i \cdot Mm + sb_i \cdot Mb}{sys\_550}\right)^2}{6} \quad \text{in.}$$

$$qy\_550^T = \left(5.36 \times 10^{-4}\ \ 1.78 \times 10^{-3}\ \ 0.02\ \ 0.01\ \ 2.77 \times 10^{-4}\ \ 7.75 \times 10^{-5}\ \ 9.23 \times 10^{-4}\right)$$

$$qy\_70_i := \frac{\left(\frac{sm_i \cdot Mm + sb_i \cdot Mb}{sys\_70}\right)^2}{6} \quad \text{in.}$$

$$qy\_70^T = \left(3.89 \times 10^{-4}\ \ 1.29 \times 10^{-3}\ \ 0.02\ \ 8.58 \times 10^{-3}\ \ 2.01 \times 10^{-4}\ \ 5.62 \times 10^{-5}\ \ 6.7 \times 10^{-4}\right)$$

Calculate Shape Factor (See ASME XI, Appendix A, Eq. (3)):

$$Q\_550 := 1 + 4.593 \cdot \left(\frac{a}{l}\right)^{1.65} - qy\_550$$

$$Q\_550^T = (1.018 \;\; 1.017 \;\; 0.996 \;\; 1.007 \;\; 1.018 \;\; 1.018 \;\; 1.017)$$

$$Q\_70 := 1 + 4.593 \cdot \left(\frac{a}{l}\right)^{1.65} - qy\_70$$

$$Q\_70^T = (1.018 \;\; 1.017 \;\; 1.002 \;\; 1.01 \;\; 1.018 \;\; 1.018 \;\; 1.018)$$

Calculate K_I for FCG (See ASME XI, Appendix A, Eq. (2)):

$$KI_i := \left[\left(sm_i - 8\right) \cdot Mm + sb_i \cdot Mb\right] \cdot \sqrt{\frac{\pi \cdot a}{Q\_550_i}} \quad \text{ksi}$$

$$KI^T = (-5.3 \;\; -11 \;\; -20.6 \;\; 2.1 \;\; -8.8 \;\; -6.6 \;\; -4.7)$$

Smallest Q between 550 F and 70 F is ued, for each load case, in order to be bounding, residual stress removed from membrane term since it is a mean stress and does not affect the range of stress intensity factor considered for FCG.

Calculate Fatigue Crack Growth (See ASME XI, Appendix A, Paragraph A-4300, Eq. (1, 2):

Let R conservatively equal 1. $R := 1$

Then $Co := 1.99 \cdot 10^{-10} \cdot \left[25.72 \cdot (2.88 - R)^{-3.07}\right]$ $Co = 7.37 \times 10^{-10}$

Assuming:
1. 24 years of additional operation and the cycles given in the orignial stress report.
2. All cycles result in a stress range given by the bounding stresses considered in this evaluation.

$n := \frac{24 \cdot 458}{40}$ $n = 275$ Assume 25 years worth of cycles to get to end of 60 years.

$DKI_i := KI_i$ ksi Assume range of stress intensity factor is given by zero load and the bounding conditions given here. If DKI < 0 then it is set equal to 0.

$$DKI^T = (-5.3 \;\; -11 \;\; -20.6 \;\; 2.1 \;\; -8.8 \;\; -6.6 \;\; -4.7) \quad \text{ksi}$$

$$DKI := \begin{pmatrix} 0 \\ 0 \\ 0 \\ DKI_3 \\ 0 \\ 0 \\ 0 \end{pmatrix}$$

$DKI^T = (0\ \ 0\ \ 0\ \ 2.1\ \ 0\ \ 0\ \ 0)$ ksi

$DADN_i := Co \cdot (DKI_i)^{3.07}$ Fatigue crack growth per load cycle, in/cycle

$DADN^T = (0\ \ 0\ \ 0\ \ 7.528 \times 10^{-9}\ \ 0\ \ 0\ \ 0)$

$DA_i := n \cdot DADN_i$ Cumulative FCG for all considered cycles, in.

$DA^T = (0\ \ 0\ \ 0\ \ 0.000002\ \ 0\ \ 0\ \ 0)$

Since crack growth, Δa, is so small, the fatigue crack growth calculation is not performed by iterativley calculating a new K_I then a new DA/DN then a new da, and so on.

$a\_final := a + DA$ in Final flaw depth, in

$2 a\_final^T = (0.45\ \ 0.45\ \ 0.45\ \ 0.45\ \ 0.45\ \ 0.45\ \ 0.45)$

$l\_final := l + 2 \cdot 2 \cdot DA$ in Final flaw lengths, in

$l\_final^T = (6.4\ \ 6.4\ \ 6.4\ \ 6.4\ \ 6.4\ \ 6.4\ \ 6.4)$

FCG is negligible.

Calculate K_I for Flaw Stability Check (See ASME XI, IWB-3613):

$$KI_i := (sm_i \cdot Mm + sb_i \cdot Mb) \cdot \sqrt{\frac{\pi \cdot a}{Q\_550_i}}$$

Smallest Q between 550 F and 70 F is ued, for each load case, in order to be bounding, residual stress included.

$$KI^T = (2 \quad -3.7 \quad -13.1 \quad 9.5 \quad -1.4 \quad 0.8 \quad 2.6) \text{ ksi}$$

$$KI\_bounding := \max(KI_0, KI_1, KI_2, KI_3, KI_4, KI_5) \qquad KI\_bounding = 9.5 \text{ ksi}$$

Calculate K_{IC} (See ASME XI, Appendix A, Paragraph A-4200):

$$KIa\_70 := 26.8 + 12.445 \cdot \exp[0.0145 \cdot (70 - RTNDT)] \qquad KIa\_70 = 56.5 \text{ ksi}$$

$$KIa\_100 := 26.8 + 12.445 \cdot \exp[0.0145 \cdot (100 - RTNDT)] \qquad KIa\_100 = 72.7 \text{ ksi}$$

$$KIa\_150 := 26.8 + 12.445 \cdot \exp[0.0145 \cdot (150 - RTNDT)] \qquad KIa\_150 = 121.6 \text{ ksi}$$

$$KIa\_550 := 26.8 + 12.445 \cdot \exp[0.0145 \cdot (550 - RTNDT)] \qquad KIa\_550 := 200 \text{ ksi}$$

$$KI\_Allowable\_LAB := \frac{KIa\_100}{\sqrt{10}} \qquad KI\_Allowable\_LAB = 23 \text{ ksi}$$

For Level A/B events and for pressure > 20% design pressure

$$KI\_Allowable\_LowP := \frac{KIa\_70}{\sqrt{2}} \qquad KI\_Allowable\_LowP = 40 \text{ ksi}$$

For events with pressure < 20% design pressure

Thus, the highest KI applied is compared the lowest allowable fracture toughness, from all events. The applied stress intensity factor remains less than allowable fracture toughness, see acceptance criteria in ASME XI, IWB-3612.

ATTACHMENT 2
Sheet 1 of 1
Record of Lead Review

Document SI Report 11100470.401 **Revision 2**
Brunswick Steam Electric Plant U2 RPV Flaw Evaluation

The signature below of the Lead Reviewer records that:

- the review indicated below has been performed by the Lead Reviewer;
- appropriate reviews were performed and errors/deficiencies (for all reviews performed) have been resolved and these records are included in the design package;
- the review was performed in accordance with EGR-NGGC-0003.

☐ **Design Verification Review** ☐ **Engineering Review** ☒ **Owner's Review**
☐ Design Review
☐ Alternate Calculation
☐ Qualification Testing

☐ **Special Engineering Review**__________

☐ YES ☐ N/A **Other Records are attached.**

A. Borodotsky aron borodotsky I have reviewed this document. 2011.09.20 15:31:03 -04'00' BESS 3/25/2011
Lead Reviewer **(print/sign)** **Discipline** **Date**

Item No.	Deficiency	Resolution
1	Editorial Per markup	Incorporated

FORM EGR-NGGC-0003-2-10
This form is a QA Record when completed and included with a completed design package. Owner's Reviews may be processed as stand alone QA records when Owner's Review is completed.