April 13, 2010

NRC 2010-0050
10 CFR 50.55a

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

Point Beach Nuclear Plant, Unit 1
Docket 50-266
Renewed License No. DPR-24

Unit 1 Refueling 32 Analytical Evaluation Report for the Reactor Vessel
Point Beach Nuclear Plant

During the recent Point Beach Nuclear Plant (PBNP), Unit 1 outage, reactor vessel examinations were performed in accordance with the Fourth Ten-Year Interval Inservice Inspection Plan. Phased array ultrasonic examinations of the reactor vessel inlet nozzle-to-pipe weld (RC-32-MRCL-AIII-03), resulted in an American Society of Mechanical Engineers (ASME) Section XI Code rejectable indication in the "A" loop. The weld is a dissimilar metal weld (between the cast stainless elbow and carbon steel nozzle using stainless steel filler material). The indication was recorded 18 inches from top dead center (TDC) and 2.1 inches from the weld centerline on the nozzle side of the weld in the nozzle forging, and approximately 0.9 inches from the buttering.

The indication can be seen in the "toward," "away," "clockwise," and "counterclockwise" directions, indicating that it is volumetric in nature (e.g., slag inclusion). The indication orientation is predominantly circumferential in nature.

The indication was found to be acceptable for further service without repair for the remainder of the life of Unit 1, including the period of renewed operation, using the acceptance criteria of ASME Section XI, Paragraph IWB-3600.

The weld was required to be examined in accordance with ASME Section XI, and the examination techniques were performed in accordance with Section XI requirements, 10 CFR 50.55a and approved Relief Request 21. Relief Request 21 was approved by the Commission via letter dated August 25, 2008 (ML081690887).

In accordance with the requirements of the 1998 Edition with 2000 Addenda of Section XI of the ASME Code, the enclosed analytical evaluation report is being submitted in accordance with Subarticle IWB-3514 for Pressure Retaining Dissimilar Metal Welds in Vessel Nozzles.

This letter contains no new Regulatory Commitments or revisions to existing Regulatory Commitments.

Please feel free to contact Mr. James Costedio, Licensing Manager, at 920/755-7427 if there are questions associated with this report.

Very truly yours,

NextEra Energy Point Beach, LLC

FOR

Larry Meyer
Site Vice President

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Point Beach Nuclear Plant, USNRC
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PSCW
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ENCLOSURE

NEXTERA ENERGY POINT BEACH, LLC
POINT BEACH NUCLEAR PLANT, UNIT 1

SECTION XI FLAW EVALUATION OF
INDICATION RECORDED ON RC-32-MRCL-AIII-03
OF THE REACTOR VESSEL
INLET NOZZLE-TO-PIPE WELD

WESTINGHOUSE NON PROPRIETARY CLASS 3

LTR-PAFM-10-50-NP
Revision 0

Section XI Flaw Evaluation of Indication Recorded on RC-32-MRCL-AIII-03 of the Point Beach Unit 1 Inlet Nozzle to Pipe Weld

March 2010

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Introduction

During the spring 2010 inspection of the reactor vessel inlet nozzle to pipe weld, an indication was identified in the "A" loop. This technical note summarizes the detection and characterization of the indication, as well as the flaw evaluation which was required. The indication was found to be acceptable for further service without repair, using the acceptance criteria of Section XI, paragraph IWB-3600.

Indication Detection and Sizing

During Phased Array (PA) ultrasonic detection examinations on RC-32-MRCL-AIII-03 (Elbow to Inlet Nozzle at 328.5°) dissimilar metal weld (Cast Stainless Elbow with Stainless weld & Stainless buttering), an indication was recorded at 18 inches from top dead center (TDC) and 2.1 inches from the weld centerline on the nozzle side of the weld in the nozzle forging and approximately 0.9 inches from the buttering. This indication can be seen in the "toward", "away", "clockwise", and "counterclockwise" directions, indicating that it is volumetric in nature (e.g., slag inclusion). The indication orientation is predominantly circumferential in nature. During the previous 10-Year Reactor Vessel examination in 1998, two (2) indications were recorded in this region and sized to be allowable due to being "buried" (subsurface).

Sizing scans were performed with the specified PA search unit, which assisted with the characterization of the indication. The sizing scans characterized the indication as being 0.71 inches long and 0.505 inches in through-wall dimension (18.10% a/t). The cladding thickness (not counted) is 0.25 inches, and the nominal pipe thickness (minus clad) is 3.27 inches. The surface "S" dimension was conservatively considered to be zero (0) to account for near-surface uncertainties.

Due to the fact that no vendor to-date has been capable of meeting the ASME Section XI, Appendix VIII, Supplement 10 (dissimilar metal weld)-required 0.125 inch root mean square (RMS) acceptance criteria, the NRC has issued RIS-2003-01 which allows the use of procedures that do not meet all of the Supplement 10 criteria provided the best available technology is used for indication sizing. In cases where the 0.125 inch RMS is not achieved, the Performance Demonstration Initiative (PDI) developed a policy (PDI 03-01) which describes how the error can be documented. This is called the RMS Error (RMSE) number. The vendor at PBNP, IHI Southwest Technologies (ISwT) has an RMSE of 0.212 inches (0.087 inches greater than 0.125 RMS), which has to be applied to any indication(s) sized per the PBNP relief request (RR-21).

In addition, because of the type of flaws that all UT vendors are tested on (i.e. – all flaws are open to the inside surface), there are no procedurally demonstrated techniques for determining that indications close to the inside surface are, in fact, sub-surface. Hence, any indication is treated as surface-connected during the ASME Section XI evaluation(s).

With RMSE applied, the indication size is considered conservatively to be 0.592 inches through wall. The maximum allowable flaw depth, per the flaw acceptance standards of IWB-3500, would be 12.5% of the wall thickness (approximately 0.41 inches through-wall).

A supplemental eddy current scan of the region was completed, and verified that the indication is not surface connected. Eddy current was applied using two excitation frequencies (30 kHz and 80 kHz). The high frequency was selected for sensitivity to surface flaws and the lower frequency for surface and sub-surface flaws. Point Beach Procedure ISwT-AET1 was used for the scans using two ECT probes oriented at 45 deg. with respect to each other to obtain sensitivity to flaws in any direction. The ECT probes were scanned using the same tooling and coordinate system used for the AUT by replacing two of the AUT search units with ECT probe modules. ECT indications were smaller than those obtained from the calibration standard flaws, with the exception of one region where the amplitude was slightly exceeded, but this region was within the nozzle weld area, which is well beyond the area of the UT indication.

Based on current findings, it is considered that this indication or group of indications is most likely to be embedded fabrication flaws; however, it is being evaluated as a surface-connected flaw due to the proximity to the inside surface. The location is actually near to the buttering of the nozzle, and also near to the clad-to-base metal interface, as shown in the sketch of Figure 1.

Flaw Evaluation Results

In preparation for the reactor vessel inspections, a set of flaw evaluation charts had been prepared for the reactor vessel weld regions [1,2], and the indication was compared with these flaw evaluation results, and found to be acceptable for further service without repair. The flaw evaluation chart of interest is Figure A-6.3 of reference 1, which is provided in an updated version as the last figure of this technical note, with the indication plotted.

For inside surface flaws in the inlet nozzle region the most severe transients are:

Loss of Flow	(normal/upset)
Large Loss of Coolant	(emergency/faulted)

It should be mentioned that residual stresses are known to exist in the adjacent weld, but since the reactor vessel, including the weld butter, is stress relieved, the stress values in this region are small, measurements of stress relieved heavy section welds have shown residual stresses of about 5 ksi at each surface, with the stresses decreasing and becoming compressive in the center of the weld. These stresses are present at all times, and will have an effect on fracture at low temperatures, when the toughness is in the transition region. At RCS operating temperatures, the residual stresses have no effect on the failure conditions. This has been demonstrated experimentally in the Heavy Section Steel Technology Intermediate Vessel test program. Therefore, residual stresses have not been used in the calculations discussed for this region.

It will be seen from Figure A-6.3 of WCAP 11477 [1] that the allowable depth for any indication, regardless of shape, is at least 20 percent of the wall thickness. The allowable depth line is across the very upper edge of the figure. This line is the result of a direct application of the Section XI acceptance criteria.

The allowable flaw depth is determined by calculation of the stress intensity factor (K) as a function of postulated flaw depth for each of the governing transients, and then determining where the K value exceeds the allowable toughness.

For the Loss of Flow transient for the Nozzle to Pipe Weld, the fracture toughness will be on the upper shelf, since there is no irradiation effect, and the initial RT_{NDT} for this weld is 60F, from the available certified material test reports [2]. The following critical flaw depths for normal/upset conditions result from these calculations:

Flaw Shape (a/l)	Critical flaw Depth (a/t)
0.01	1.0
0.1667	1.0
0.5	1.0

The flaw evaluation chart is then determined from the worst case of the results above and the results for the governing faulted condition, the Loss of Coolant Accident. The results for the governing emergency/faulted condition:

Flaw Shape (a/l)	Critical flaw Depth (a/t)
0.01	1.0
0.1667	1.0
0.5	1.0

The allowable flaw depth for this location must also be compared with the allowable depth based on the primary stress limit criteria of Section III, NB-3200. The allowable depth from this calculation is 58% of the wall thickness, for a continuous circumferential flaw, so the fracture mechanics limits of IWB-3600 will be governing.

Therefore, we see that the allowable flaw depth is very large, regardless of the flaw shape for this location. For conservatism the allowable flaw depth in the chart of Figure 2 has been cut off at a/t = 0.2.

Use of the Flaw Evaluation Chart of Figure 2

Once the indication is discovered, it must be characterized as to its location. Length (l) and depth dimension (a for surface flaws, 2a for embedded flaws), including its distance from the inside surface (S) for embedded indications. This characterization is discussed in further detail in Article IWA 3000 of Section XI. Since the "S" dimension could not be determined using PDI procedures, it is assumed to be zero.

The following parameters must be calculated from the above dimensions to use the charts (see Figure 2):

Flaw Shape Parameter, a/l

Flaw Depth Parameter, a/t

where

t = wall thickness of region where indication is located
l = length of indication
a = depth of surface flaw; or half depth of embedded flaw in the width direction

Once the above parameters have been calculated, these two parameters for each indication may be plotted directly on the appropriate evaluation chart. Their location on the chart determines the acceptability immediately.

The evaluation chart for surface flaws is shown in Figure 2, for circumferential flaws. Note that there are three lines in the chart, representing the acceptable length of service, 10, 20, and 30 years.

Fatigue crack growth is the only mechanism of growth for this indication, as the hydrogen overpressure applied to the reactor water chemistry precludes stress corrosion cracking in the materials present in this region.

The crack growth during service is negligible. Fatigue crack growth was calculated in WCAP-11477 [1] for several flaws of similar depth to the location of interest. The flaws in the table below had a length of 6 times the depth and were open to the surface and exposed to the water environment. The flaws in the table show no appreciable flaw growth from service. The indication of interest is not nearly as elongated as the six to one flaw whose results are shown here, and is not exposed to the water environment, so the growth will be even less than predicted here. Therefore, there is no difference in the allowable depth as a function of service time for this location, and the allowable depths for 10, 20 and 30 years of service are the same.

Initial	Crack Length After Year			
Crack Depth (in.)	10	20	30	40
0.202	0.20245	0.20245	0.20246	0.20247
0.303	0.30374	0.30376	0.30378	0.30380
0.405	0.40514	0.40517	0.40519	0.40522
0.506	0.50640	0.50644	0.50648	0.50652

Leak Before Break

The identification of this indication in the reactor vessel nozzle region has no effect whatsoever on the margins for leak before break at Point Beach Unit 1. The concept of leak before break is a simple one, in that a margin is established between the size flaw which could lead to detectable leakage, and the size flaw which could cause failure.

The indication which has been discovered here will have no effect of the margins between leak and break, because it is extremely unlikely to extend during service. Let us examine the two aspects of leak before break, the propensity for a through wall flaw to leak, and the critical flaw size, or the size flaw which could lead to failure. The leak rate for a through-wall flaw in this region is a function of the wall thickness of the nozzle or pipe, and the internal pressure; these factors are unaffected by the presence of the indication. As for the size flaw which could cause failure of the pipe, it is a function of the material properties of the nozzle and pipe, and the loadings which exist, and neither of these is affected by the indication.

The leak before break margins are effectively higher now than at any other time in the operating history of Point Beach Nuclear Plant, because there is a new awareness, and a much higher sensitivity to small amounts of leakage now. After a number of recent operating events, the industry made a conscious effort to improve their leak detection capability. As a result, virtually all pressurized water reactors (PWRs) in the US have a leak detection capability of less than or equal to 0.1 gpm, as "needed." All plants also monitor seven day moving averages of reactor coolant system leak rates.

Action levels have been standardized for all PWRs, and are based on deviations from:

- The seven day rolling average,
- Specific values, and
- The baseline mean.

Action response times following a leak detection vary, based on the action level exceeded and range up to containment entry to identify the source of the leak. Utilities take the commitment of shutdowns due to unidentified leakage seriously. This is exemplified with utility shutdowns in July 2009, due to a 0.2 gpm leakage, and another in August 2009, with 0.09 gpm leakage. This improvement in leak detection sensitivity is due to multiple measures being monitored. The leakage rate used as a basis for leak before break at Point Beach is 1 gpm. [3]

The newly required generic leak rate action levels are identified in PWROG report, WCAP-16465 [4], and are below:

> Each PWR utility is required to implement the following standard action levels for RCS inventory balance in their RCS leakage monitoring program.
>
> A. Action levels on the absolute value of unidentified RCS inventory balance (from surveillance data):
>
> Level 1 - One seven day rolling average of unidentified RCS inventory balance values greater than 0.1 gpm.
>
> Level 2 - Two consecutive unidentified RCS inventory balance values greater than 0.15 gpm.
>
> Level 3 - One unidentified RCS inventory balance value greater than 0.3 gpm.
>
> Note: Calculation of the absolute RCS inventory balance values must include the rules for the treatment of negative values and missing observations.
>
> B. Action levels on the deviation from the baseline mean:
>
> Level 1 - Nine consecutive unidentified RCS inventory balance values greater than the baseline mean [μ] value.

Level 2 - Two of three consecutive unidentified RCS inventory balance values greater than [μ + 2σ], where σ is the baseline standard deviation.
Level 3 - ,One unidentified RCS inventory balance value greater than [μ +3σ].

It should be noted that the NRC staff, in their approval of the aging management evaluation for Westinghouse Class 1 piping and associated components [5] identified a concern about maintaining the leak before break status for a plant if an indication was to be found and evaluated in cast stainless steel. Likewise, concern was expressed about such an evaluation of a fatigue crack. Since the indication of interest here is not located in cast stainless steel, and is not a fatigue crack, or even exposed to the water environment, no concern about the leak before break status of the plant exists here.

Similarly, the leak before break evaluation for Point Beach Nuclear Plant Unit 1 primary loop piping approved by the NRC in reference [6] is not affected by this indication.

References

1. A.A. Chan and W.H. Bamford, "Handbook on Flaw Evaluation for Point Beach Units 1 and 2 Reactor Vessels", Westinghouse Electric Report WCAP-11477 Rev. 1, July 1990.

2. W.H. Bamford, et al, "Background and Technical Basis for the Handbook on Flaw Evaluation for Point Beach Units 1 and 2 Reactor Vessels", Westinghouse Electric Report WCAP-11478 Rev. 1, July 1990.

3. D.C. Bhowmick, "Technical Justification for Eliminating Large Primary Loop Pipe Rupture as the Structural Design Basis for the Point Beach Nuclear Plant Units 1 and 2 for the Power Uprate and License Renewal Program" Westinghouse Electric Report WCAP-14439-P, Rev. 2, September 2003

4. Westinghouse Electric Report, WCAP-16465-NP, Rev. 0, "Pressurized Water Reactor Owners Group Standard RCS Leakage Action Levels and Response Guidelines for Pressurized Water Reactors," September, 2006

5. Westinghouse Electric Report, WCAP-14575-A, "Aging Management Evaluation for Class 1 Piping and Associated Pressure Boundary Components", December 2000.

6. NRC Safety Evaluation dated June 6, 2005, "Point Beach Nuclear Plant, Units 1 and 2, Issuance of Amendments re: Leak Before Break Evaluation for Primary Loop Piping." Amendments 219/224, TAC docs. MC1279 and MC 1280. (ML 043360295).

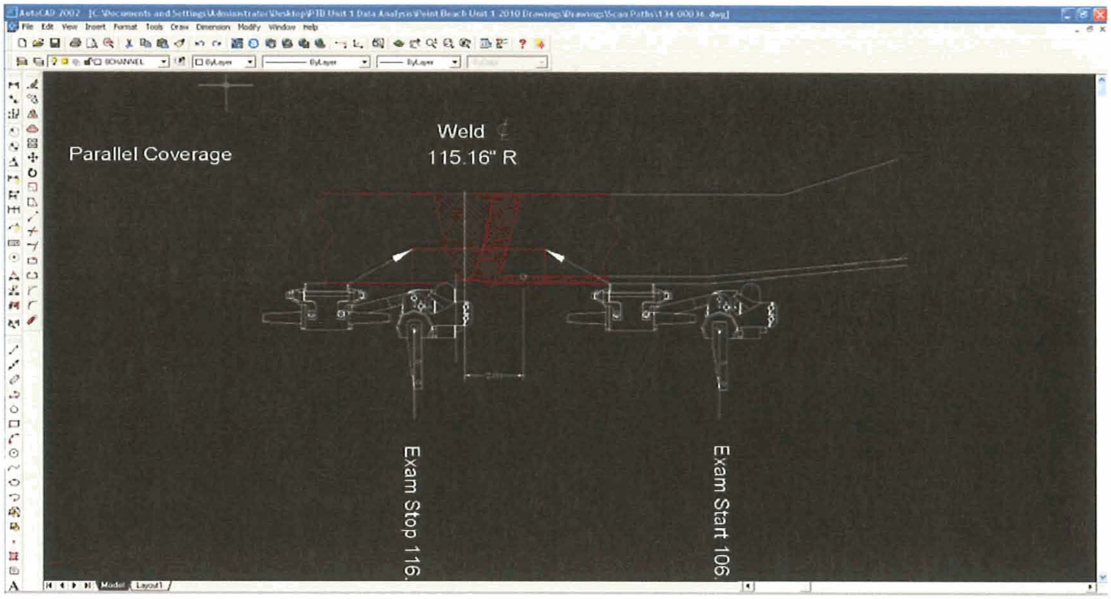
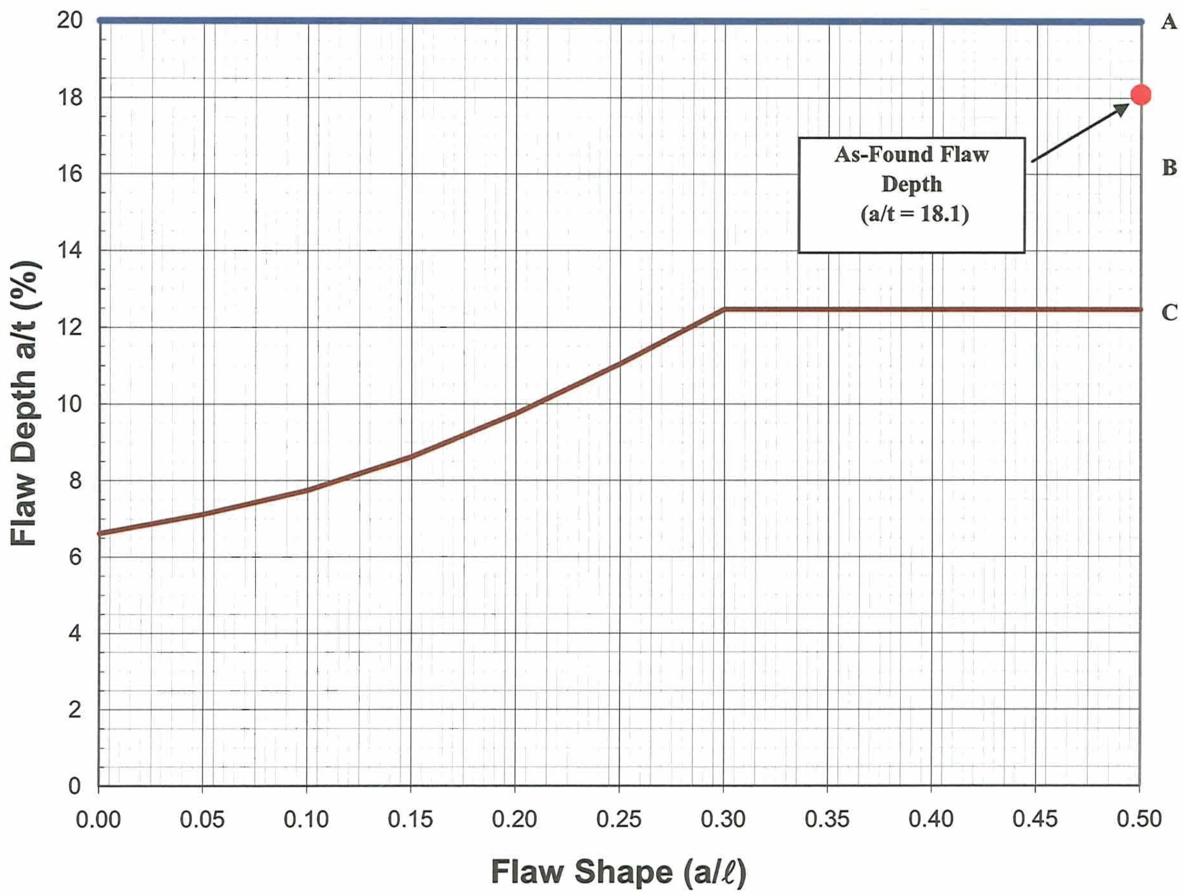


Figure 1: Inspection Volume of the Region of Interest, and Location of the Indication



LEGEND

A- The 10, 20, 30 year acceptable flaw limits

B- Within this zone, the surface flaw is acceptable by ASME Code analytical criteria in IWB-3600

C- ASME Code allowable (Table IWB-3514-1)

Figure 2 (Figure A-6.3) Evaluation Chart for Inlet Nozzle Safe-end to Nozzle Weld [1]

X	Inside Surface	X	Surface Flaw	____	Longitudinal Flaw
____	Outside Surface	____	Embedded Flaw	X	Circumferential Flaw