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March 23, 1984

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Attention: Mr. J. R. Miller, Chief
Operating Reactors, Branch 3

Gentlemen:

DOCKET 50-266
REACTOR VESSEL OUTLET NOZZLE-TO-SHELL WELD
FLAW INDICATION EVALUATION
POINT BEACH NUCLEAR PLANT, UNIT 1

In our March 12, 1984 letter we transmitted for your concurrence our procedure for evaluating the acceptability of the flaw indications in the Point Beach Unit 1 reactor vessel outlet nozzle-to-shell welds. In that letter we also advised you that it was our intention to demonstrate that the indications meet the acceptance criteria of ASME Section XI thereby confirming the continued serviceability of the reactor vessel.

The evaluation consisted of two approaches: experiments by Southwest Research Institute on the Point Beach calibration block and fracture analysis of the flaw indications. The calibration block experiments were undertaken because of the difficulties in sizing the reflectors in the outlet nozzles. Basically, these indications were oriented normal to the incident beam and almost parallel to the examination surface. This provided a high amplitude response but an insignificant change in metal path. The Code does not outline or specify the technique for sizing this type of reflector. The calibration block experiments in which the position and orientation of the nozzle indications was duplicated with a 3/8" flat-bottomed hole in the calibration block permitted the performance of beam spread analyses using the same search unit assembly as was used at Point Beach to more accurately size the reflectors. Based on these studies, beam spread correction factors were determined. The beam spread corrected sizing demonstrates that the two previously unacceptable indications are, in fact, within the limits specified in the Code. Attachment 1 documents the corrected characterization of the reflectors of concern.

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Notwithstanding the above conclusions, a fracture integrity evaluation of the flaw indications has now been completed by Southwest Research Institute. The analysis, which utilized a linear elastic fracture mechanics approach in accordance with ASME Section XI requirements, shows that the observed flaw indications, as initially sized, are stable under all postulated loading conditions. Additionally, the margins of safety specified in IWB-3612 are met. To aid in your preliminary review, Attachment 2 is provided which summarizes the key aspects of the flaw indication evaluation.

We expect to forward complete reports on Southwest Research Institute's beam spread analysis and fracture integrity evaluation by March 30, 1984. Should you have further questions please do not hesitate to contact us. If you desire, a meeting can be arranged to discuss the evaluation reports.

It is our present conclusion that the requirements of ASME Section XI have been satisfied and that the serviceability of the Point Beach Unit 1 reactor vessel is confirmed for continued operation.

Very truly yours,

CW Fay

Vice President-Nuclear Power

C. W. Fay

Copy to NRC Resident Inspector

Attachments

Attachment 1

POINT BEACH NUCLEAR PLANT, UNIT 1
REACTOR VESSEL OUTLET NOZZLE
REFINED FLAW CHARACTERIZATION

Southwest Research Institute's experiments on the calibration block corroborated the original calibration very well. These studies also produced correction factors based on the beam spread associated with the particular search unit used at Point Beach. These correction factors, for these indications only, were determined to be 1.035" in length and 0.709" in the through-wall dimension. Applying these factors to the two indications in the outlet nozzles yields the following refined reflector characterization:

Flaw #2	Flaw #4
"A" outlet nozzle-to-shell weld.	"C" outlet nozzle-to-shell weld.
2a = 0.531"	2a = 0.431"
l = 0.9"	l = 0.255"
a/l = 0.295	a/l = 0.5 (Limit)

a/t% Allowable	a/t% Measured	a/t% Allowable	a/t% Measured
4.05	2.9	6.5	2.4

where t = 9.125"

As can be seen, both indications are within Code allowable.

SUMMARY OF SOUTHWEST RESEARCH INSTITUTE'S FLAW INDICATION EVALUATION
POINT BEACH NUCLEAR PLANT, UNIT 1

The following summarizes key data, assumptions, and results from Southwest Research Institute's evaluation of the flaw indications, as originally sized, in the Point Beach Nuclear Plant, Unit 1 reactor vessel outlet nozzle-to-shell welds.

1. Initial Flaw Characterization:

		Flaw #2*	Flaw #4
(a)	Location:	"A" outlet nozzle-to-shell weld.	"C" outlet nozzle-to-shell weld.
		Nozzle azimuth: 104°	Nozzle azimuth: 311°
		Flaw center: 7" deep in weld	Flaw center: 5" deep in weld
(b)	Parameters:	2a = 1.24"	2a = 1.14"
		1 = 1.935"	1 = 1.29"
		a/1 = 0.32	a/1 = 0.44

(c) ASME Section XI:

Flaw #2 a/t% Allowable	Flaw #2 a/t% Measured	Flaw #4 a/t% Allowable	Flaw #4 a/t% Measured
4.3	6.8	3.8	6.2

where t = 9.125"

*In our March 12, 1984 letter in paragraph 1.a.(1), the characterization of the proximity indication in the "A" outlet nozzle contained two clerical errors. It is correctly characterized as follows:

2a = 0.2"
1 = 0.2"
Area < 1-2/3% of the larger indication

It remains an acceptable indication per the limits of Table IWB-3512.

2. Weld Material Data:

The nozzle-to-shell welds are primarily welded with Mn-Mo-Ni wire HT #8T1554B and Linde #80 flux Lot 8479. The weld chemistry (Wt %) is as follows:

C	Mn	P	S	Si	Cr	Ni	Mo	Cu
0.08	1.58	0.014	0.012	0.45	0.07	0.60	0.40	0.19

The initial RT_{NDT} and Charpy upper shelf energy are taken to be 25°F and 65 ft-lbs, respectively. The end-of-life fluence at the nozzles is predicted to be 1.65×10^{16} n/cm^2. This fluence value is less than the minimum values used in Regulatory Guide 1.99 to predict irradiated material property shifts; therefore, the unirradiated RT_{NDT} and Charpy upper shelf energy values will be used in the analysis. For purposes of this nozzle analysis only, the Charpy upper shelf energy, 65 ft-lbs, has been conservatively correlated to an irradiated upper shelf fracture toughness of 149 ksi $\sqrt{in}$. This value, $K_{Ic} = K_{Ia} =$ 149 ksi $\sqrt{in}$ was used to provide a conservative upper bound on the ASME Code K_{Ia} and K_{Ic} curves. Note that this conservative upper shelf fracture toughness was established to expedite completion of the analysis. We believe a significantly higher value can be demonstrated but have not chosen to pursue further refinement since the current analysis is satisfactory with the conservative value. Hence, this conservatism is unique and applicable only to this analysis.

3. Fatigue Crack Growth Analysis:

Fatigue growth of the flaw indications was analyzed per ASME, Section XI, Appendix A. It was calculated that the present crack depth of Flaw #2 in the "A" outlet nozzle $2 \times a_i = 2 \times 0.62 = 1.24$" could grow to a crack depth $2 \times a_f = 2 \times 0.625 = 1.25$" by end of life. Since the predicted growth of Flaw #2 is so small, interaction with the satellite (proximity) flaw is not expected to occur. Cumulative crack growth for Flaw #4 in the "C" outlet nozzle was calculated and is insignificant. The final crack size a_f is used in the flaw integrity evaluations.

4. Flaw Integrity Assessment:

The acceptance criterion for normal, upset, and test conditions (Levels I and II) is given by:

$$K_{Ia}/K_I > \sqrt{10}$$

The maximum stresses are obtained for both Flaw #2 and Flaw #4 during the cold hydro test at 3,125 psia. The acceptance criteria are satisfied for both flaw indications, hence they are acceptable.

Flaw #2

$$\frac{K_{1a}}{K_1} = \frac{149}{43.5} = 3.43 > \sqrt{10}$$

Flaw #4

$$\frac{K_{1a}}{K_1} = \frac{149}{38.02} = 3.92 > \sqrt{10}$$

The acceptance criterion for a flaw to withstand crack initiation under emergency and faulted (Levels III and IV) conditions is given by:

$$K_{1c}/K_1 > \sqrt{2}$$

Three accident conditions were considered: loss-of-coolant accident; large steam line break (with and without off-site power); and locked rotor pressure transient (loss of load).

For all transients considered it was noted that those with high pressure provided the worst case. It was also noted that in the cooldown transient cases the thermal stresses tended to be compressive at the crack location and, thus, lessen the contribution of the pressure stress. The cold hydro test condition stresses, therefore, bounded the stresses imposed by the accident conditions. Hence, the flaws are still acceptable.