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SVP-00-168

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

Quad Cities Nuclear Power Station, Unit 1
Facility Operating License No. DPR-29
NRC Docket No. 50-254

Subject: Request for Approval of Weld Repair and Pipe Flaw Evaluation

Reference:

(1) Letter from J.P. Dimmette, Jr. (ComEd) to U.S. NRC, "Remediation Plan for the Unit 1 Intergranular Stress Corrosion Cracking Susceptible Welds," dated March 31, 1999

(2) Letter from U.S. NRC to O. D. Kingsley (ComEd), "Review of Remediation Plan for Quad Cities, Unit 1 IGSCC Susceptible Welds," dated June 14, 1999

(3) Letter from J.P. Dimmette, Jr. (ComEd) to U.S. NRC, "Request for Approval of Pipe Weld Overlays and Pipe Flaw Evaluations," dated November 24, 1998

(4) Letter from U.S. NRC to O. D. Kingsley (ComEd), "Flaw Evaluation of Recirculation Line Weld 02BS-F4 at Quad Cities Nuclear Power Station, Unit 1," dated April 16, 1999

(5) Letter from J.P. Dimmette, Jr. (ComEd) to U.S. NRC, "Change of Commitment, Remediation Plan for the Unit 1 Intergranular Stress Corrosion Cracking Susceptible Welds," dated October 24, 2000

The purpose of this letter is to describe the weld overlay repair and request NRC approval, in accordance with Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," of a pipe flaw evaluation and partial repair for a weld in the Reactor Recirculation piping at the Quad Cities Nuclear Power Station (QCNPS), Unit 1. In Reference 1 we described our remediation plan for the welds susceptible to Intergranular Stress Corrosion Cracking (IGSCC). This plan was reviewed by the NRC in Reference 2. The plan stated that weld overlay repairs would be performed for three welds that were categorized as Category F, "Cracked – Inadequate or No Repair," in accordance with Generic Letter (GL) 88-01.

A001

As described in Reference 3, three welds with flaws were previously evaluated to be acceptable for continued operation without weld overlay repair until the next refueling outage. This evaluation was accepted by the NRC as documented in Reference 4. The letter concluded that continued plant operation beyond the next fuel cycle was dependent on the satisfactory evaluation of the re-inspection results or implementing acceptable repairs during the next refueling outage. Reference 5 described the deferral of weld overlay repairs for two of these welds, identified as 02AS-S4 and 02AD-F12.

Prior to the current Unit 1 refueling outage, we planned to perform a weld overlay repair on the 02BS-F4 recirculation system-piping weld. During our planning for this repair we expected to find radiation fields in the reactor drywell of less than 250 mrem/hour as a result of recent experience. Based on this, the total radiation dose for the overlay was estimated at 16.1 Person-Rem. After the unit was shutdown on October 14, 2000, initial drywell surveys revealed actual radiation fields of 900 mrem/hour to 1100 mrem/hour. Based on this information, our estimate of the total dose was raised to 70 Person-Rem for this one weld overlay.

Despite extensive efforts to limit exposure through the use of engineering and human factor controls, the total radiation dose for this weld overlay was increased to 88 Person-Rem on Friday, October 20, 2000, based on actual field productivity and experience. As a result, the decision was made to evaluate the feasibility of deferring completion of this overlay to a future outage for which additional steps such as chemical decontamination would be taken to reduce exposure.

The evaluation indicated that it would be acceptable to complete the second weld layer that was in progress, prepare the surface of the weld and conduct an ultrasonic examination of the two weld layers and weld 02BS-F4 during this outage. The evaluation and plan were described to the NRC staff during a conference call on October 20, 2000.

Our plan is to complete the remaining layers of the overlay during the next refueling outage planned in 2002 when chemical decontamination of the reactor recirculation piping will be performed. Dose rates during Q1R17 are expected to be significantly lower due to the performance of a chemical decontamination. Our estimate for the total radiation dose received to complete the weld overlay during the course of both outages is approximately 73 Person-Rem. We estimated the radiation dose avoidance by completing the weld overlay in the next refueling outage would be 15 Person-Rem. A detailed evaluation of the dose associated with this repair is included as Attachment 1 to this letter.

An evaluation of the flaw assuming conservative crack growth rates has been performed by General Electric (GE) and is included as Attachment 2. This evaluation considers the initial flaw size, expected growth rates and plant chemistry parameters. The evaluation concludes that the flaw is acceptable for continued operation for the next cycle without the completion of a full weld overlay repair. It also notes that substantial structural margins are assured for the next operating cycle. The weld overlay installation is described in Attachment 3. A confirmatory examination of the weld is currently in progress to support the final flaw evaluation and the specific results will be provided to the NRC upon completion of the evaluation of those results. The weld is currently being examined using automatic and manual ultrasonic examination techniques. These are also described in Attachment 3.

In accordance with GL 88-01 and based on the information provided in Attachment 2, we request NRC review and approval of the evaluation of the flaw in weld 02BS-F4 and the partial repair of this weld. Based on this approval, we will defer completion of the weld overlay repair to the next refueling outage when dose rates are expected to be lower as a result of chemical decontamination activities. NRC approval is requested by October 30, 2000 in support of unit startup, which is expected on that day.

Should you have any questions concerning this letter, please contact Mr. C.C. Peterson at (309) 654-2241, extension 3609.

Respectfully,

Joel P. Dimmette, Jr.

Joel P. Dimmette, Jr.
Site Vice President
Quad Cities Generating Station

Attachments:

Attachment 1: Personnel Radiation Exposure Projections for Weld Overlay Repair to 02BS-F4
Attachment 2: GE Nuclear Energy Report No. GE-NE-B13-02064-00-18, "Quad Cities Unit 1 Evaluation of the Indication in Weld 02BS-F4 in the 28 inch Recirculation Piping," dated October 2000
Attachment 3: Weld Overlay Description and Confirmatory/Pre-service Examinations For Limited Service Weld Overlay 02BS-F4

cc: Regional Administrator – NRC Region III
NRC Senior Resident Inspector – Quad Cities Generating Station

bcc: Project Manager – NRR
Office of Nuclear Facility Safety, - IDNS
Senior Reactor Analyst – IDNS
Manager of Energy Practice – Winston and Strawn
Director, Licensing and Compliance – ComEd
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NRC Coordinator – Quad Cities Generating Station
NSRB Site Coordinator – Quad Cities Generating Station
Site Vice President - Quad Cities Generating Station
Station Manager - Quad Cities Generating Station
SVP Letter File

Attachment 1
Request for Approval of Weld Repair and Pipe Flaw Evaluation

Personnel Radiation Exposure Projections for Weld Overlay Repair to 02BS-F4

Deferral of completion of the weld overlay repair to weld 02BS-F4 is being pursued due to the unexpectedly high primary system dose rates experienced after the shutdown for the current Quad Cities Nuclear Power Station (QCNPS) Unit 1 refueling outage, identified as Q1R16. These high dose rates have lead to personnel exposures significantly higher than planned for this repair.

Following reactor shutdown for Q1R16, primary system dose rates were discovered to be significantly higher than expected. The actual dose rates are exceptionally high compared to previous Quad Cities experience and compared to other U.S. Boiling Water Reactors (BWRs). Table A-1 shows data for the average Boiling Reactor Average Contact (BRAC) dose rates for BWRs as collected by the BWR Owner's Group for recent outages. The table shows that the average BRAC dose rates range from 53 mrem/hour to 500 mrem/hour, with the previous QCNPS Unit 1 average BRAC dose rate at 142 mrem/hour. The current Q1R16 average BRAC dose rate is 735, which is well above both the previous Unit 1 average and the industry values shown in the table. QCNPS has begun an aggressive effort, with industry participation, to determine the cause of these high dose rates. Chemical decontamination of the piping was not performed during this outage because the dose rates were expected to be lower based on the effects of depleted Zinc injection, dose rates experienced during the Unit 1, April, 1999, mid-cycle outage and the Unit 2 refueling outage.

During pre-outage planning, QCNPS estimated the total personnel exposure for the weld overlay repair to 02BS-F4 to be approximately 16 rem. This estimate was based on expected general area dose rates of 250 mrem/hour, which was consistent with previous post-shutdown radiation levels at the location of the weld. The high primary system dose rates have caused the general area dose rate at the planned weld overlay location to be 900 -1100 mrem/hour.

Figure A-1 shows that an actual radiation exposure of approximately 30 person-rem was incurred to complete two of the planned seven weld layers. The figure also shows that approximately 20 additional person-rem have been incurred to prepare the partial weld overlay for ultrasonic testing (UT). These preparations included placing a cosmetic third weld layer and then grinding the overlay to produce an acceptable surface for UT.

Table A-2 compares the projected personnel exposure that would be required to complete the full seven layer weld overlay in Q1R16 to the radiation exposure required to complete the weld overlay in the next scheduled refueling outage, Q1R17. The table shows that approximately 15 person-rem would be saved by deferring completion to Q1R17. This projection is based on a planned chemical decontamination to reduce general area dose rates at the weld location to 175 mrem/hr and reduce exposure hours through improved work techniques.

Attachment 1
Request for Approval of Weld Repair and Flaw Evaluation

Table A-1
BWR Owner's Group Radiological Data
(plant names removed)

USBWR Radiological Data Reported 6/99, 12/99 or 3/00 Meetings							
Plant	**Unit**	**Avg. BRAC DR mR/hr**	**DW eff mR/hr**	**HWC ppm**	**HWC SCFM**	**Zinc**	**NMCA**
	1	53	8	1.45	8	Depleted	Yes
	3	60	3.6	2.5	62	Depleted	Plan 10/00
	1	97	5.6	No	No	No	Plan 5/01
	2	100	5.4	NR	10	Depleted	May-01
	2	105	8.9	1.6	82	No	No Plans
	3	107	6.3	1.6	88	Depleted	Plan 11/99
	1	110	5.8	0.359	16	Depleted	2001
	2	110	5.8	0.359	16	Depleted	2001
	2	125	4.1	NR	30-60	Depleted	No
	1	128	8	no	No	No	No
	1	129	22	0.3	10.5	Depleted	Yes 11/99
	2	131	5	NR	6	Depleted	Yes
Quad Cities (Note 1)	1	142	8.6	0.4	11	Depleted	Yes
	2	150	NR	1.45	50	Depleted	Yes
	2	153	2.5	NR	8	Depleted	Plan 11/00
	1	165	7.6	1.6	82	No	No Plans
	1	167	12.3	0.72	35	Depleted	Plan 00/01
	2	180	15	0.8	25	No	No
	1	181	4.1	NR	70	Depleted	Yes
	1	200	10	No	No	Depleted	Fall 99
	1	207	9.8	No	No	No	Plan 2002
	1	219	4.4	No	No	Depleted	Evaluating
	1	235	10	0.37	8.6	No	Plan 4/00
	2	238	6.3	No	No	Depleted	Plan 3/00
	1	250	4.3	Installation	Inprogress	Depleted	Plan 2001
Quad Cities	2	262	6.1	0.4	11	Depleted	Yes
	3	263	8.4	0.15	6	Depleted	Yes
	1	267	7.2	NR	6	Depleted	Yes
	1	300	5	NR	22	Depleted	No
	1	338	22.4	1.8	40	Depleted	No
	2	375	14.1	NR	39	Depleted	No
	1	380	11.2	NR	39	Depleted	No
	1	391	7.5	0.5	12-Aug	No	No
	1A	424	14.8	1	NR	Depleted	No
	2	425	15.3	1.6	88	Depleted	Plan 11/99
	1B	432	14.8	1	NR	Depleted	No
	2	446	6.4	No	No	Depleted	01-May
	1	450	25	1.3	32	Depleted	No
	1	470	NR	Plan 2000	Plan 2000	Depleted 19	Plan 8/00
	3	500	8	No	No	No	No
	1	NR	NR	NR	NR	NR	NR
	1	60-250	NR	Yes	NR	No	No
Quad Cities (Note 2)	**1**	**735**	**NR**	**NR**	**NR**	**Depleted**	**Yes**

Note 1: This is the Post-Decontamination BRAC data obtained during Q1P02 in April 1999	
Note 2: This is the Post-Shutdown BRAC data during Q1R16 in October 2000	
Data as 04/01/2000	NR = Not reported
	Sites who did not show different data per unit show identical numbers for both units
	Data from most recently submitted Plant Status Report

Attachment 1
Request for Approval of Weld Repair and Pipe Flaw Evaluation

Table A-2

Comparison of Projected Personnel Exposures to Complete 02BS-F4 Weld Overlay Repair
(data and projections as of October 20, 2000 at 1200 hours)

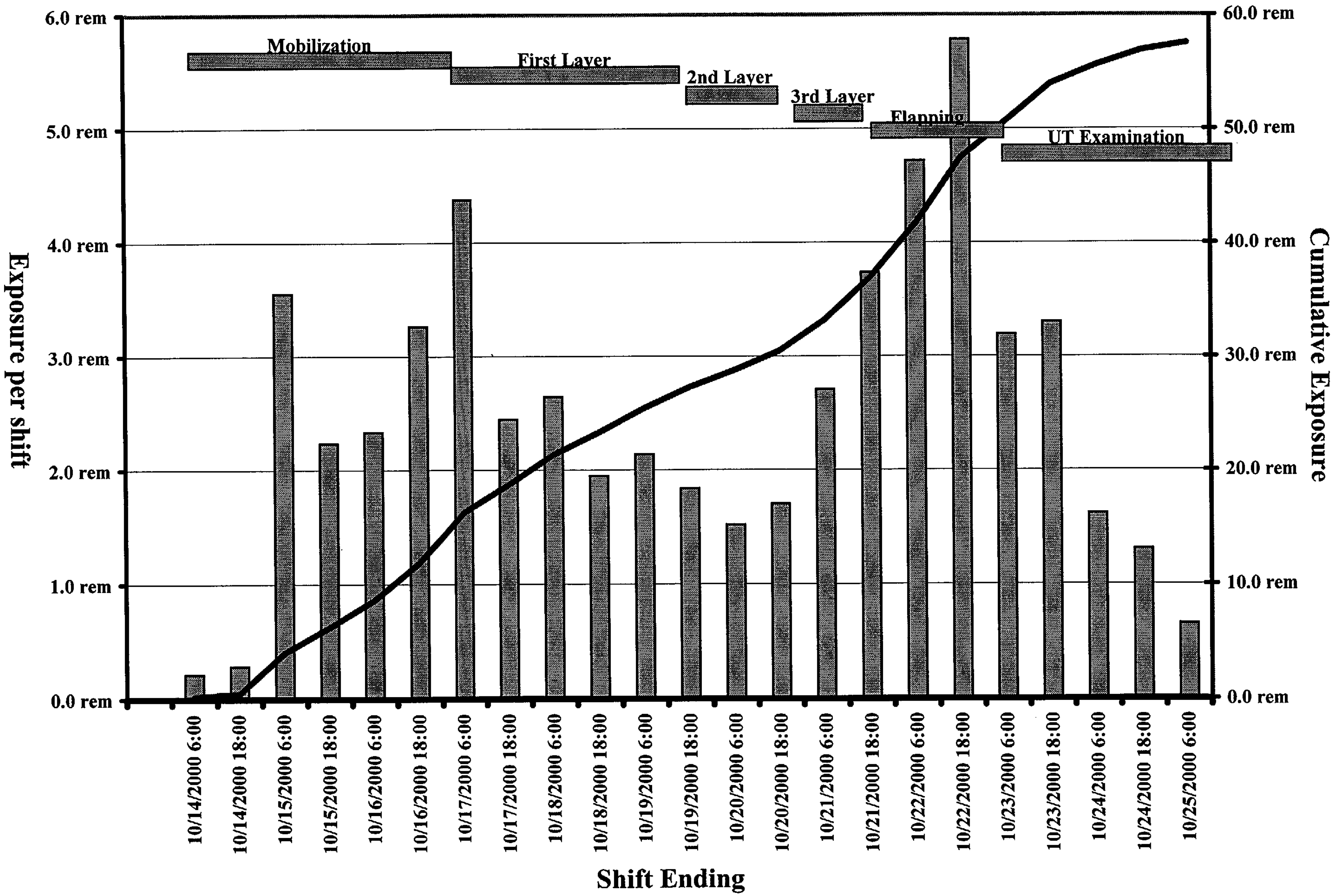
Evolution	Sub-evolution	Sub-evolution Estimate (rem)	Total Evolution Estimate/Actual (rem)
Estimated dose to complete weld overlay in Q1R16	Actual dose to date	25.56	88.06
	Estimate to complete first 3 layers in Q1R16	30.4	
	Estimate to complete remaining 5 layers in Q1R16	32.1	
Estimated dose to complete weld overlay in Q1R17	Actual dose to date	25.56	73.40
	Estimate to complete first 3 layers in Q1R16	30.4	
	Estimate to complete remaining 5 layers in Q1R17	17.4	

Estimated Exposure Savings from weld overlay repairs deferral: 14.67 rem

Attachment 1

Request for Approval of Limited Service Weld Repair and Flaw Evaluation

Figure A-1: Weld Overlay 02BS-F4 Exposure



<u>Attachment 2</u>
Request for Approval of Weld Repair and Pipe Flaw Evaluation

Quad Cites Unit 1

Evaluation of the Indication in Weld 02BS-F4 in the 28 inch Recirculation Piping

REACTOR SERVICES
GE Nuclear Energy
175 Curtner Avenue, San Jose, CA 95125

GE-NE-B13-02064-00-18
DRF #B13-02064-00
Class I
October 2000

Quad Cities Unit 1

Evaluation of the Indication in Weld 02BS-F4 in the 28 inch Recirculation Piping

October 2000

Prepared for

Commonwealth Edison Co.

Prepared by

GE Nuclear Energy

Quad Cities Unit 1

Evaluation of the Indication in Weld 02BS-F4 in the 28 inch Recirculation Piping

October 2000

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IMPORTANT NOTICE REGARDING

CONTENTS OF THIS REPORT

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Table of Contents

1. Background

During the fall 1998 outage at Quad Cities Unit 1 a flaw indication was identified in weld 02BS-F4 in the 28-inch recirculation piping. Table 1 shows the inspection history of this weld. UT reflectors were found in the 1989 and 1996 inspections but were evaluated as root geometry. The 1998 inspection was performed by PDI qualified examiners using EPRI qualified procedures. The 1998 inspection determined that there was an Intergrannular Stress Corrosion Cracking (IGSCC) indication 0.25 in. deep and 27 inches long in the weld region.

As part of the mitigation program to reduce IGSCC susceptibility, the weld was subjected to induction heat stress improvement (IHSI) in 1984. Hydrogen water chemistry was implemented in 1990 and the plant has been operating with NobleChem™ since April 1999. In view of the fact that the weld has two mitigation measures in place, the likelihood of a service induced crack during the last 10 years is low. This suggests that the weld had an existing indication since 1989 and the root geometry attributed by the 1989 and 1996 inspections was most likely the IGSCC indication identified in 1998. The indication was evaluated using Section XI, ASME Code, IWB-3640 and Appendix C [1] procedures using crack growth rates for normal water chemistry based on NUREG-0313, Rev. 2 [2] and shown to be acceptable for continued operation for two, 24 month, operating cycles (until October 2002).

Nevertheless, it was decided that a weld overlay should be performed on weld 02BS-F4 during the October 2000 outage so that a permanent repair would be in place. The weld overlay was designed to be a full structural overlay that meets Code Case 504-1 requirements. However, during the welding process the personnel exposure dose rate was found to be extremely high. If the overlay were to be completed as planned, the resulting personnel exposure would have been unacceptable. The weld overlay application was stopped after three layers (approximately 0.2 in.). The intent is to decontaminate the pipe at the next refuel outage (fall 2002) and address the repair issue at that time.

Under the weld classification of Generic letter 88-01 the 02BS-F4 weld would be designated as Category F (cracked weld with inadequate or no repair) and would require inspection this outage. The weld has been prepared for inspection (removal of surface irregularities to assure a smooth surface for UT) and is being inspected. NRC Generic letter 88-01 also requires staff approval of flaw evaluations in accordance with Section XI, IWB-3640. The flaw evaluation reported here demonstrates that the Section XI requirements are fully satisfied ***without taking structural credit for the weld layers that have already been deposited.*** The evaluation uses a bounding approach in considering NWC crack growth from the 0.25 inch depth determined from the UT in 1998. As long as the depth value from the current inspection is less than the projected crack depth in this analysis, the analysis presented here is conservative and therefore valid.

This report provides the justification for continued operation for the next cycle without taking credit for the weld layers that have been deposited. In particular, it is shown that the Section XI requirements are fully satisfied for both NWC and HWC using NRC approved procedures. Section 2 of the report describes the water chemistry during the last cycle as well as the HWC availability. Section 3 describes the structural analysis and crack evaluations. Section 4 provides the conclusions from the structural analysis and the justification for continued operation.

2.Water Chemistry

Quad Cities Unit 1 has been operating with excellent water conductivity. Figure 1 shows the conductivity data for the last cycle. It is seen that the overall conductivity ranges from 0.1 to 0.15 μS/cm over the last cycle. The plant has been operating with HWC since 1990 and NobleChem was implemented in April 1999. The ECP history during the last cycle is shown in Figure 2. Again, the measured ECP is well below the –230 mV SHE threshold for crack arrest. IGSCC crack growth is expected to be negligible under these conditions.

The HWC availability was 94.5% during 1999 and 96.7% during 2000 to date. It is expected that HWC availability during the coming cycle will be comparable. Based on the low conductivity and ECP, it is reasonable to expect extremely low (near zero) growth rates during the coming cycle.

3. Structural Analysis and Crack Growth prediction

3.1. Structural Analysis

The pipe weld with the indication can be evaluated using the procedures of Section XI, Appendix C and the acceptance criteria of IWB-3640. The evaluation is based on limit load mechanism.

3.1.1. Limit Load Analysis Methodology

The limit load analysis method used in the analysis is consistent with the procedures outlined in Section XI of the ASME Code [1]. A brief description of the method is provided next.

Consider a circumferential crack of length, $l = 2R\alpha$ and constant depth, d. In order to determine the point at which limit load is achieved, it is necessary to apply the equations of equilibrium assuming that the cracked section behaves like a plastic hinge. For this condition, the assumed stress state at the cracked section is as shown in Figure 3 where the maximum stress is the flow stress of the material, σ_f. Equilibrium of longitudinal forces and moments about the axis gives the following equations:

$$\beta = [(\pi - \alpha d/t) - (P_m/\sigma_f)\pi]/2 \tag{1}$$

$$P_b' = (2\sigma_f/\pi)\,(2 \sin\beta - d/t \sin\alpha) \tag{2}$$

Where,
- t = pipe thickness
- α = crack half-angle as shown in Figure 3
- β = angle that defines the location of the neutral axis
- P_m = Primary membrane stress
- P_b' = Failure bending stress

The safety factor, SF, is then incorporated as follows:

$$P_b' = Z{*}SF\,(P_m + P_b + P_e/SF) - P_m \tag{3}$$

P_m and P_b are primary membrane and bending stresses, respectively. P_e is secondary stress and includes stresses from all displacement-controlled loadings such as thermal expansion and dynamic anchor motion. All three quantities are calculated from the analysis of applied loading. The safety factor value is 2.77 for normal/upset conditions and 1.39 for emergency/faulted conditions. The Z factor is discussed next.

Z Factor

The test data considered by the ASME Code in developing the flaw evaluation procedure (Appendix C, Section XI) indicated that the welds produced by a process without using a flux had fracture toughness as good or better than the base metal. However, the flux welds had lower toughness. To account for the reduced toughness of the flux welds (as compared to non-flux welds) the Section XI procedures prescribe a penalty factor, called a 'Z' factor. The examples of flux welds are submerged arc welds (SAW) and shielded metal arc welds (SMAW). Gas metal-arc welds (GMAW) and gas tungsten-arc welds (GTAW) are examples of non-flux welds. Figure IWB-3641-1 of Reference 1 may be used to define the weld-base metal interface. The expressions for the value of the Z factor in Appendix C of Section XI are given as follows:

$$\begin{aligned} Z &= 1.15\,[1 + 0.013(OD\text{-}4)] \text{ for SMAW} \\ &= 1.30\,[1 + 0.010(OD\text{-}4)] \text{ for SAW} \end{aligned} \qquad (4)$$

where OD is the nominal pipe size (NPS) in inches. Except for the root pass, the remaining weld was completed at both the welds using a SMAW process. Therefore, the Z factor in the evaluation was calculated using the expression for SMAW.

3.1.2. Applied Piping Stresses

The applied piping stresses were calculated from the reported axial and bending loads at the subject welds. Table 2 shows the calculated values of the stresses for various load cases. The three thermal load cases are the following: (1) System at 546°F, except between MO-1-0202-6A and -6B and from MO-1001-50 to penetration X-12 which is at 135°F; (2) System at 546°F, except between MO-1-0202-6A and -6B and all of the line 1-1025-20"-A to penetration X-12 which is at 135°F; and (3) System at 340°F, except between MO-1-0202-6A and -6B which is at 135°F.

For the purposes of cross-section area and section modulus calculations, the pipe OD was 28 inches, and the pipe wall thickness was 1.24 inches. The stresses due to weight and seismic were treated as primary stresses (P_m and P_b) and those due to thermal load cases as secondary (P_e).

3.2. Crack Growth Evaluation - NWC

The crack growth due to stress corrosion cracking (SCC) is expected to be significant compared to any fatigue crack growth. The SCC growth rate is a function of sustained stress field including that due to weld residual stresses and the relationship between stress intensity factor, K, and da/dt.

The weld was subjected to an induction heating stress improvement (IHSI) process. The IHSI treatment was intended to eliminate the tensile weld residual stress pattern and produce a compressive residual stress pattern at the inside diameter surfaces of the girth welds. Fully effective IHSI produces compressive stresses up to the inner 50% of the pipe wall. Therefore, even if the IHSI were partially effective, the as-welded residual stresses would have been considerably reduced. For conservatism, this beneficial effect of IHSI was not considered in this evaluation and as-welded residual stress distribution was used. The as-welded residual stress distribution was obtained from Reference 2 and is represented in the polynomial form as the following:

$$\sigma_R = 30.0\,[\,\sigma_0 - \sigma_1(x/t) + \sigma_2(x/t)^2 - \sigma_3(x/t)^3 - \sigma_4(x/t)^4] \qquad (5)$$

where,

σ_0 = 1.0

σ_1 = -6.910

σ_2 = 8.687

σ_3 = -0.480

σ_4 = -2.027

x = Radial distance from inside diameter surface of pipe

t = Pipe thickness

The unit of stress is ksi. The multiplier 30 ksi is based on the fitted stress distribution curve shown in Figure 3 of Reference 2.

The stress intensity factor, K_{IR} due to weld residual stress distribution was calculated using the following expression from Reference 2:

$$K_{IR} = 30.0\ \{\sqrt{(\pi a)}\}[\ \sigma_0 i_0 + \sigma_1(a)i_1 + \sigma_2(a)^2 i_2 + \sigma_3(a)^3 i_3 + \sigma_4(a)^4 i_4] \tag{6}$$

where,

a = Crack depth, in.
σ_0, σ_1, σ_2, σ_3 and σ_4 are as defined in Equation (5)
i_0, i_1, i_2, i_3 and i_4 are as defined in Reference 2.

The stress intensity factor due to sustained applied loading was calculated using the following expression from Reference 2:

$$K_{I,App} = \sigma_{app}[\sqrt{(\pi a)}][1.122 + 0.3989(a/t) + 1.5778(a/t)^2 + 0.6049(a/t)^3] \tag{7}$$

where,

σ_{app} = Applied sustained stress
t = Pipe thickness

The applied stress was calculated by summing the pressure stress and the membrane and bending stresses from weight and thermal loadings. The internal pressure was assumed as 1000 psi. Stresses from only the most limiting thermal case were considered.

The total stress intensity factor, K_{IT} was obtained by:

$$K_{IT} = K_{IR} + K_{I,App} \tag{8}$$

The SCC growth rate relationship used was the following [2]:

$$da/dt = 3.590\text{x}10^{-8}\ (K_{IT})^{2.161}\ \text{inches/hour} \tag{9}$$

where, K_{IT} is in ksi$\sqrt{}$in units.

The crack growth rate predicted by Equation (9) is quite conservative considering the fact that the plant is expected to operate under hydrogen water chemistry (HWC) conditions. Equation (9) reflects SCC growth rate for plants with normal water chemistry (NWC). An alternative report [3] provides a detailed comparison of the crack growth rates predicted by Equation (9) with those predicted using relationships that take into account plant-specific reactor water chemistry. For example, at a K_{IT} value of 20 ksi$\sqrt{in}$, Equation (9) predicts a crack growth rate of $2.3x10^{-5}$ in/hour whereas the calculations using PLEDGE Code [4], which considered the HWC conditions such as hydrogen injection rate and reactor water conductivity expected during the next fuel cycle, predicted a crack growth rate of only $7x10^{-7}$ in/hr. This clearly illustrates the conservative nature of the crack growth rate relationship used in this evaluation.

Furthermore, the K_{IT} calculation in this evaluation is also conservative since it does not consider the beneficial effect of IHSI treatment on the as-welded residual stress distribution.

Figure 4 shows the results of the depth crack growth calculation for two cycles (35000 hours) for the 02BS-F4 weld. It is seen that the crack growth is modest even with a conservative crack growth rate relationship.

3.3. Limit Load Evaluation

Figure 5 graphically shows the results of the limit load evaluation. The projected end-of-cycle depth and length of the indication is also shown on the same plot. The projected indication length was calculated using the NRC accepted $5x10^{-5}$ in/hr crack growth rate. As shown in Figure 5, the calculated allowable flaw depth is greater than the maximum limit of 0.6t for flux welds, for any crack length. It should be noted that this limit has been raised to 0.75t, same as that for the base metal and non-flux welds, in the later editions of the ASME Section XI. Thus, the use of 0.6t limit is conservative.

A review of Figure 5 shows that the projected indication size at the end of the next fuel cycle (i.e. October 2002) is well within the allowable value. Thus, operation for at least one additional fuel cycle of operation following the current outage until Fall 2002 is justified.

3.4. Evaluation based on HWC Crack Growth Rates

As described earlier, crack growth rates under HWC (especially when the ECP is below –230 mV SHE) are expected to be extremely low nearing arrest condition. This is particularly true for the recirculation piping where there no radiolysis or fluence effects (that apply to shroud cracking). Nevertheless, a conservative factor of improvement of 5 (over the NWC crack growth rate from NUREG-0313) was selected to evaluate the crack growth. Figure 6 shows the projected crack growth relative to the allowable flaw size. As expected, there is even more margin when credit is taken for the HWC benefit.

4. Conclusions

The 1998 UT examination of the 02BS-F4 recirculation piping weld identified an indication. This report presents the fracture mechanics evaluation results for this indication. The fracture mechanics evaluation was conducted using the procedures of Appendix C and Paragraph IWB-3640, in ASME Section XI. The crack growth evaluation to determine the projected crack depth at the end of next fuel cycle was conducted using the procedures outlined in NUREG-0313, Revision 2. Two fuel cycles (35000 hours) were assumed following the 1998 UT examination. The analysis was performed without taking credit for the additional weld layers (approximately 0.2 in.) that were applied prior to the stopping of the weld overlay application.

The evaluation results show that the subject indications meet the criteria of IWB-3640, Section XI of the ASME Code. Therefore, continued operation for at least one additional fuel cycle after the current outage (i.e. until October 2002) is justified for the 02BS-F4 weld.

It is important to note that the application of the partial overlay (approximately 0.2 in. thick) provides significant benefits both from the viewpoint of structural reinforcement and stress improvement. The partial overlay thickness is almost half the thickness needed for a full structural overlay and provides important added load carrying capacity. Since the overlay is water backed, there is stress improvement benefit on the ID surface. Finally, the application of NobleChem™ and the high HWC availability assures that regardless of the overlay the likelihood of crack extension in the future is extremely low. Clearly, the 02BS-F4 weld is better today with a partial overlay than it was without it.

5. References

[1] ASME Boiler & Pressure Vessel Code, Section XI, 1989 Edition, No Addenda.

[2] "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," NUREG-0313, Revision 2, January 1998.

[3] "Assessment of Crack Growth Rates Applicable to IHSI Treated Recirculation Piping at Quad Cities Station, Unit 1," GE Report No. GE-NE-B13-01980-030-2, November 1998.

[4] F.P. Ford and P.L. Andresen, "Prediction of Environmentally Assisted Cracking in Boiling Water Reactors, Part I: Unirradiated Stainless Steel Components," GE NEDC-32613, June 1996.

Table 1. Inspection History for Weld 02BS-F4

Year	Manual Exam Procedure	Personnel Training	Exam Result
1989	3 Party Agreement	3 Party Agreement	360° Intermittent Counterbore & Root Geometry
1996	3 Party Agreement	3 Party Agreement	Root Geometry Observed at <50% DAC
1998	3 Party Agreement	PDI	IGSCC – 0.25" deep x 27" Long

Table 2(a). Indication size and Pipe Thickness

Weld ID	Indication Depth (in.)	Indication Length (in.)	Pipe Thickness (in.)
02BS-F4	0.25	27.0	1.24

Table 2(b). Calculated Stresses for various Applied Loading

Load	Weld 02BS-F4 Stresses (ksi)	
	Membrane	Bending
Weight	0.154	0.175
Thermal 1	0.036	3.065
Thermal 2	0.061	3.014
Thermal 3	0.001	1.905
Seismic OBE	0.032	0.193
Seismic DBE	0.063	0.387

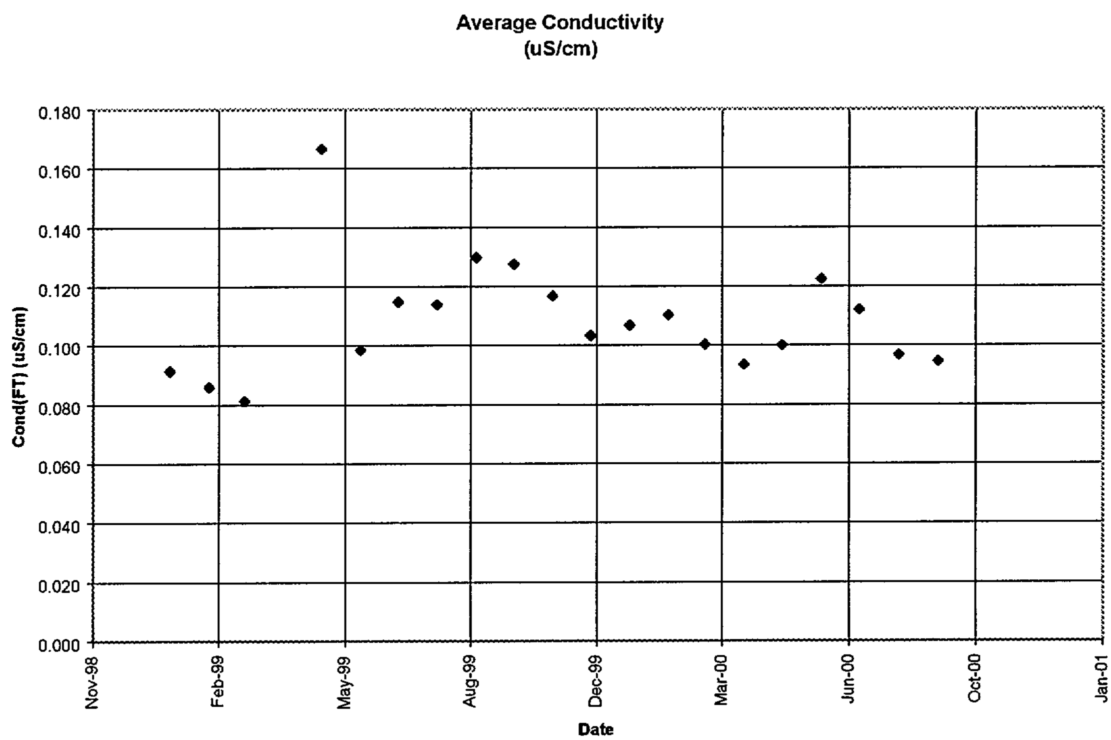

Figure 1. Average Monthly Conductivity during the last cycle

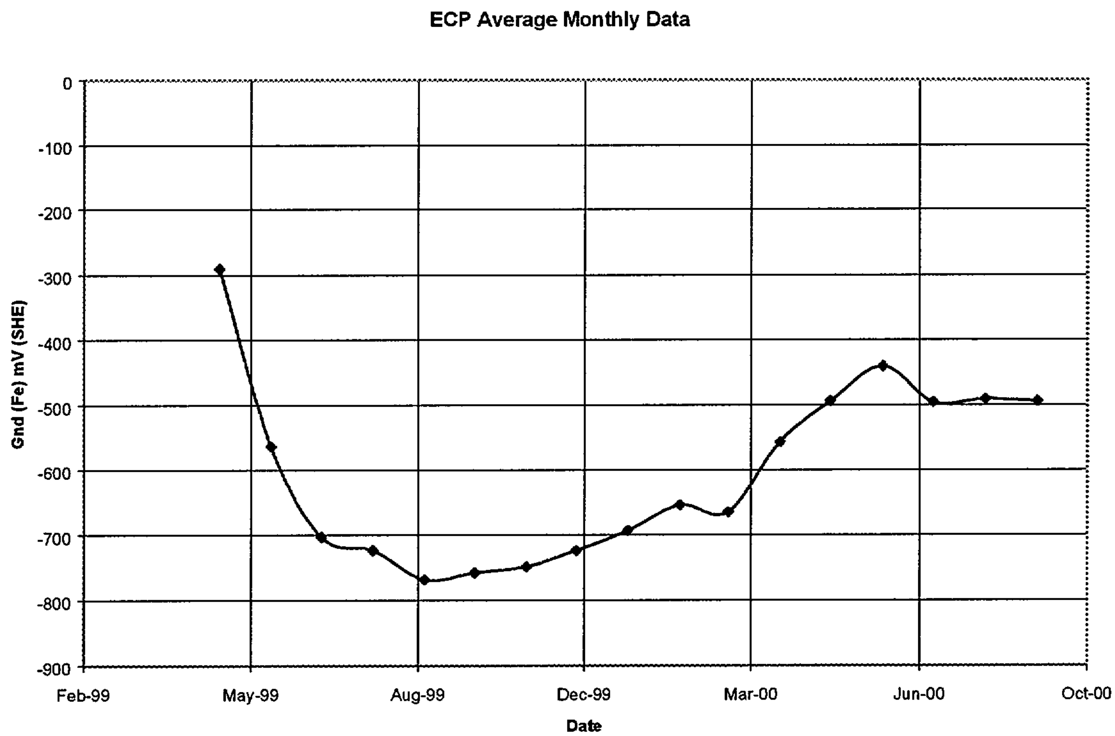

Figure 2. ECP History during the last cycle

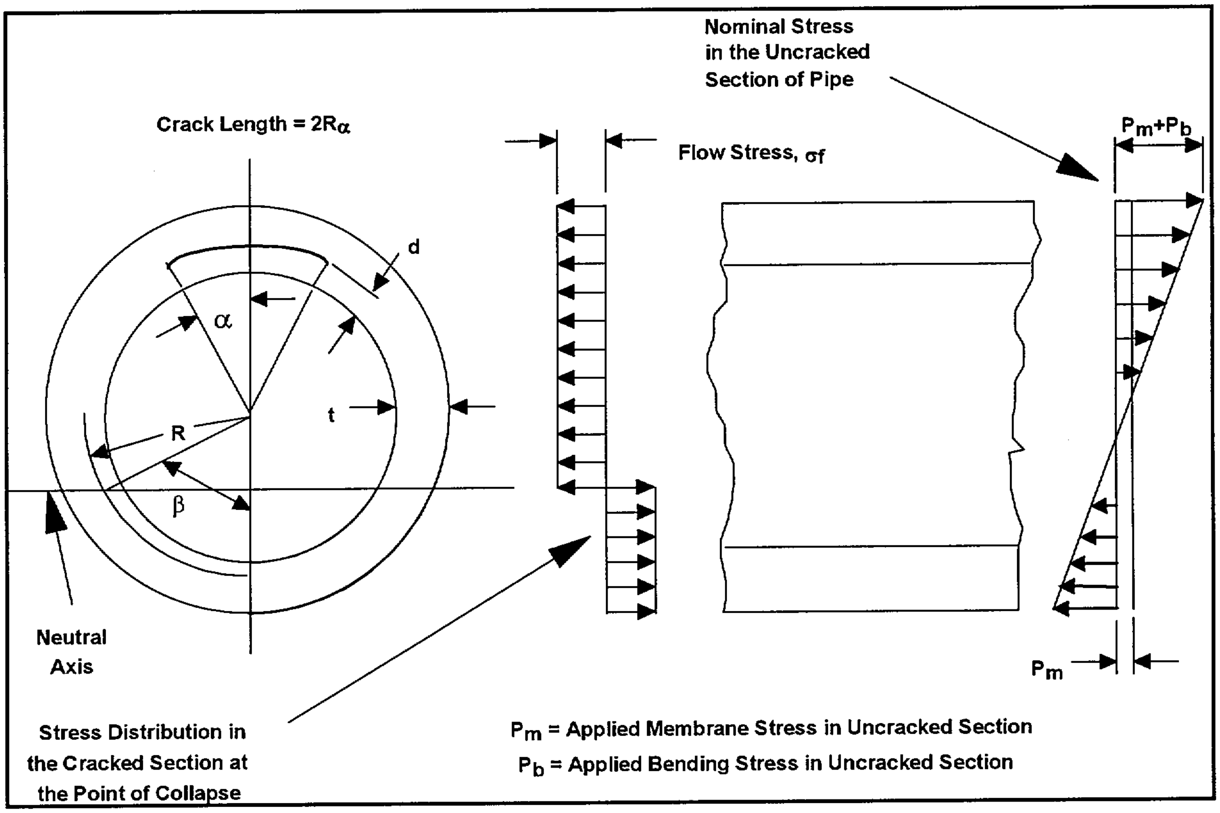

Figure 3. Stress Distribution in a Cracked Pipe at the Point of Collapse

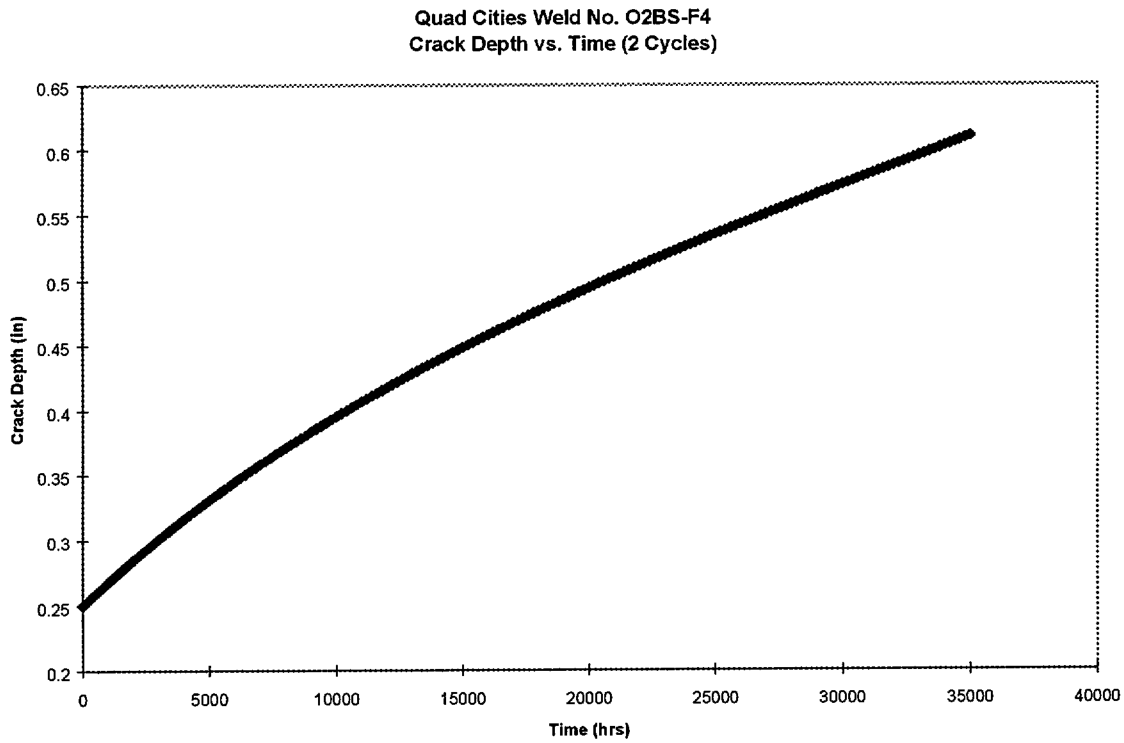

Figure 4. Predicted Crack Growth for 2 Cycles for Weld 02BS-F4 (NWC)

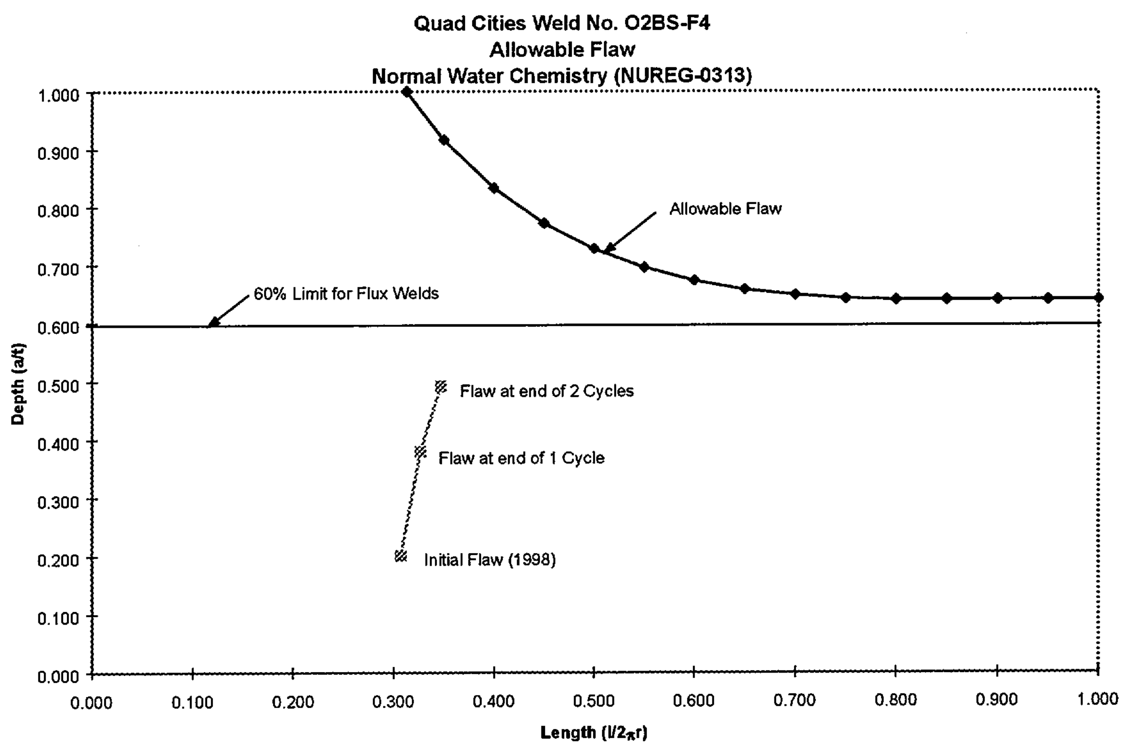

Figure 5. Comparison with Allowable Flaw Size for Weld 02BS-F4 (NWC Crack Growth Rates)

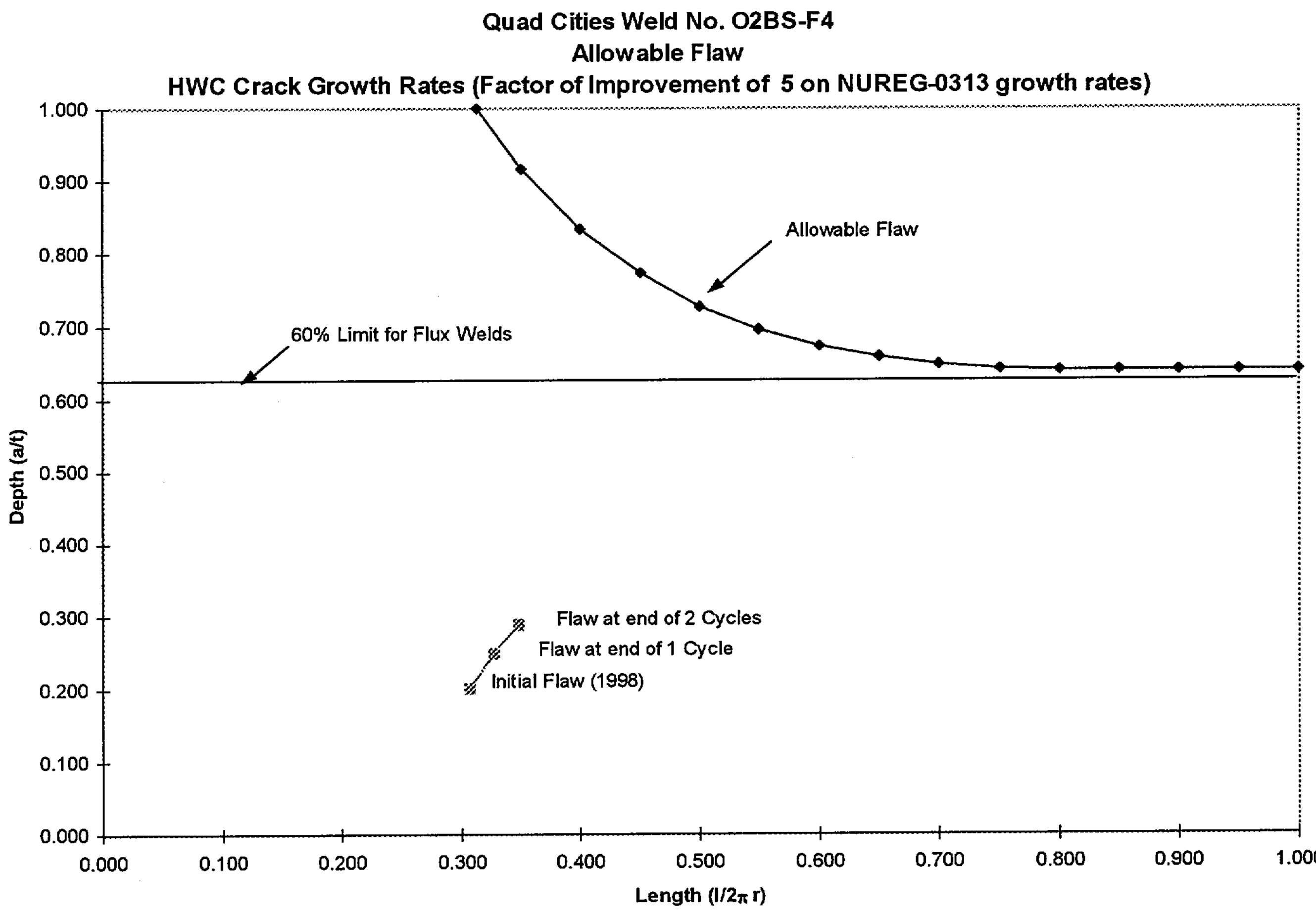

Figure 6. Comparison with Allowable Flaw Size for Weld 02BS-F4 (HWC Crack Growth Rates – FOI of 5 on NWC growth rates)

Attachment 3
Request for Approval of Weld Repair and Pipe Flaw Evaluation

Weld Overlay Description and Confirmatory/Pre-service Examinations
For Limited Services Weld Overlay 02BS-F4

The limited service weld overlay for weld 02BS-F4 was installed using our American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI, Repair/Replacement Program. The overlay weld process was performed per weld procedure WPS 8-8-B, which satisfies ASME B&PV Code Section IX and ASME B&PV Code Section XI requirements for welding. The weld filler metal meets the requirements of ASME B&PV Code Section IX and ASME B&PV Code Section III for Class I materials. Dimensional details of the as-left limited service overlay are presented in Table 1, below. From these dimensions, the average overlay thickness is 0.24". The delta ferrite measurements for the first two layers were measured and found to exceed 7.5 Ferrite Number (FN).

Following the surface preparation, the Limited Services Weld Overlay 02BS-F4 is being subjected to the following confirmatory/pre-service examinations.

1. Surface examination (i.e., dye penetrant) performed to detect unacceptable surface flaws on the weld overlay outside diameter (OD) surface.

2. Volumetric examinations performed by examiners qualified to the Performance Demonstration Initiative (PDI) to detect and size sub-surface flaws. Automated and manual ultrasonic testing (UT) techniques will be performed using PDI qualified procedures GE-UT-232, Version 1 (Procedure for Automated Ultrasonic Examination and Tomoview Analysis of Weld Overlaid Austenitic Piping Welds In Accordance with PDI) and/or Exelon's NDT-C-77, Revision 1 (Manual Ultrasonic Examination of Weld Overlaid Austenitic Piping Welds). These procedures have been qualified by PDI to detect and size flaws in the weld overlay material and the outer 25% of the base metal thickness. The manual UT procedure will be used to supplement the automated UT procedure in areas where physical interferences prevent examination by automated technique or any re-evaluation for further disposition/analysis.

   Due to the unique configuration of the weld repair the automated and manual UT procedures specified above will incorporate both the procedural and supplemental techniques listed below.

- Calibration reference sensitivity is established utilizing the internal diameter (ID) axial and circumferential notches on a 12" diameter austenitic stainless steel calibration block with a 0.25" thick weld overlay (Calibration block # 99905QC).
- 0° Longitudinal Wave, using a 4.0 Mhz search unit calibrated to encompass the total thickness of the base metal and overlay material, to detect lack of bond at the base metal-to-weld overlay metal interface and/or interbead lack of fusion within the overlay material, which may interfere with the 45° and 60° examinations.
- Outside Surface Creeping Wave (ODCR), using a 2.0 Mhz search unit, to detect contamination cracks, interbead lack of fusion within the weld overlay material, propagation

of crack tip into the upper base metal-to-weld overlay metal interface, and to locate any flaws that may interfere with the 45° and 60° examinations.

- 45° and 60° Refracted Longitudinal (RL) Waves, using 2.0 Mhz search units, to detect and size flaws in the original base material. Axial scans will be performed on both the upstream and downstream sides of the overlay for dual axial direction coverage and overlapping with the ODCR examination. 45° and 60° RL circumferential scans will be performed in the clockwise and counterclockwise directions.
- A demonstration was conducted to show that the 45° RL and 60° RL sensitivities established using the calibration block 99905QC can resolve the ID notches found on a thicker austenitic stainless steel calibration block (99906QC, 22" diameter with 0.42" thick weld overlay and 10% base metal thickness ID machined notches). The result of this demonstration proved that the 45° RL and 60° RL examinations can interrogate the original base material below the outer 25% thickness of the base material.

The calibrations and demonstration with the above listed transducers on calibration blocks 99905QC and 99906QC were found to be acceptable by the site Authorized Nuclear Inservice Inspector. Additionally, EPRI/NDE Center has verbally concurred with the above examination techniques, calibration, and demonstration described above for the examination of this Limited Services Weld Overlay.

Table 1: Thickness Measurements for Limited Service Overlay on 02BS-F4:

PRE-REPAIR ORIGINAL PIPE

	ORIGINAL PIPE (US)	EDGE AREA OF OVERLAY	CENTER OF OVERLAY	EDGE AREA OF OVERLAY	ORIGINAL PIPE (DS)
0°	-	1.20"	-	1.23"	-
90°	-	1.21"	-	1.22"	-
180°	-	1.20"	-	1.22"	-
270°	-	1.21"	-	1.23"	-

FINAL ACCEPTED REPAIR

	ORIGINAL PIPE (US)	EDGE AREA OF OVERLAY	CENTER OF OVERLAY	EDGE AREA OF OVERLAY	ORIGINAL PIPE (DS)
0°	1.25"	1.47"	1.44"	1.46"	1.26"
90°	1.26"	1.45"	1.49"	1.45"	1.25"
180°	1.25"	1.46"	1.49"	1.47"	1.26"
270°	1.25"	1.43"	1.47"	1.47"	1.26"