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EVALUATION AND DISPOSITION OF INDICATIONS AT LASALLE COUNTY NUCLEAR STATION UNIT 1

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PAGE 1 OF 1

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REV 1

CERTIFICATION BY REGISTERED PROFESSIONAL ENGINEER

I hereby certify that this document and the calculations contained herein were reviewed by me and to the best of my knowledge are correct and complete. I further certify that, to the best of my knowledge, design margins required by the original Code of Construction have not been reduced as a result of the activities addressed herein. I am a duly Registered Professional Engineer under the laws of the State of Illinois and am competent to review this document.

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TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

1.0 INTRODUCTION

LaSalle County Nuclear Station Unit 1 recently completed its first fuel cycle of operation. During the subsequent refueling outage, Induction Heating Stress Improvement (IHSI) was applied to welds in the Reactor Recirculation and Residual Heat Removal systems. Post-IHSI ultrasonic examinations (UT) revealed linear indications at Welds 1-RR-1001-10 and 1-RR-1005-27A. Figures 1.0-1 through 1.0-4 present the location and geometric details of these indications. For the purpose of this evaluation, these linear indications will be conservatively treated as intergranular stress corrosion cracking (IGSCC) flaws.

IGSCC, which occurs in the weld heat-affected zones (HAZ) of stainless steel piping in boiling water reactors (BWRs), results from the interaction of three critical factors:

1. Corrosive environment,
2. Sensitized material, and
3. Tensile stresses.

The phenomenon, observed in austenitic materials in laboratory work in the 1950s and 1960s, has been observed in BWRs since the early 1960s. In the mid-1970s, cracking was detected in 4" recirculation bypass lines and 10" core spray lines in several BWRs. Between 1975 and 1980, over 200 incidents of IGSCC in austenitic stainless steel BWR piping were reported, primarily in 10" diameter and smaller piping systems. Since 1980, indications of IGSCC have been reported with increasing frequency in both small diameter (12" and less) and larger diameter (greater than 12") piping in the United States and overseas.

The purpose of this report is to demonstrate that the original design margins of safety inherent in the ASME Code for the flawed welds at LaSalle Unit 1 have not been degraded. This is accomplished by calculating the amount of predicted IGSCC flaw growth expected during the next fuel cycle of operation, and then assuring that the remaining uncracked pipe cross-section is within Code safety margins when subjected to applied loads. Sections 2.0 and 3.0 present the evaluation criteria and loads used in the analysis of the flawed welds. Section 4.0 presents the evaluation methods and results. Sections 5.0 and 6.0 present a summary of conclusions and the references used in the evaluation.

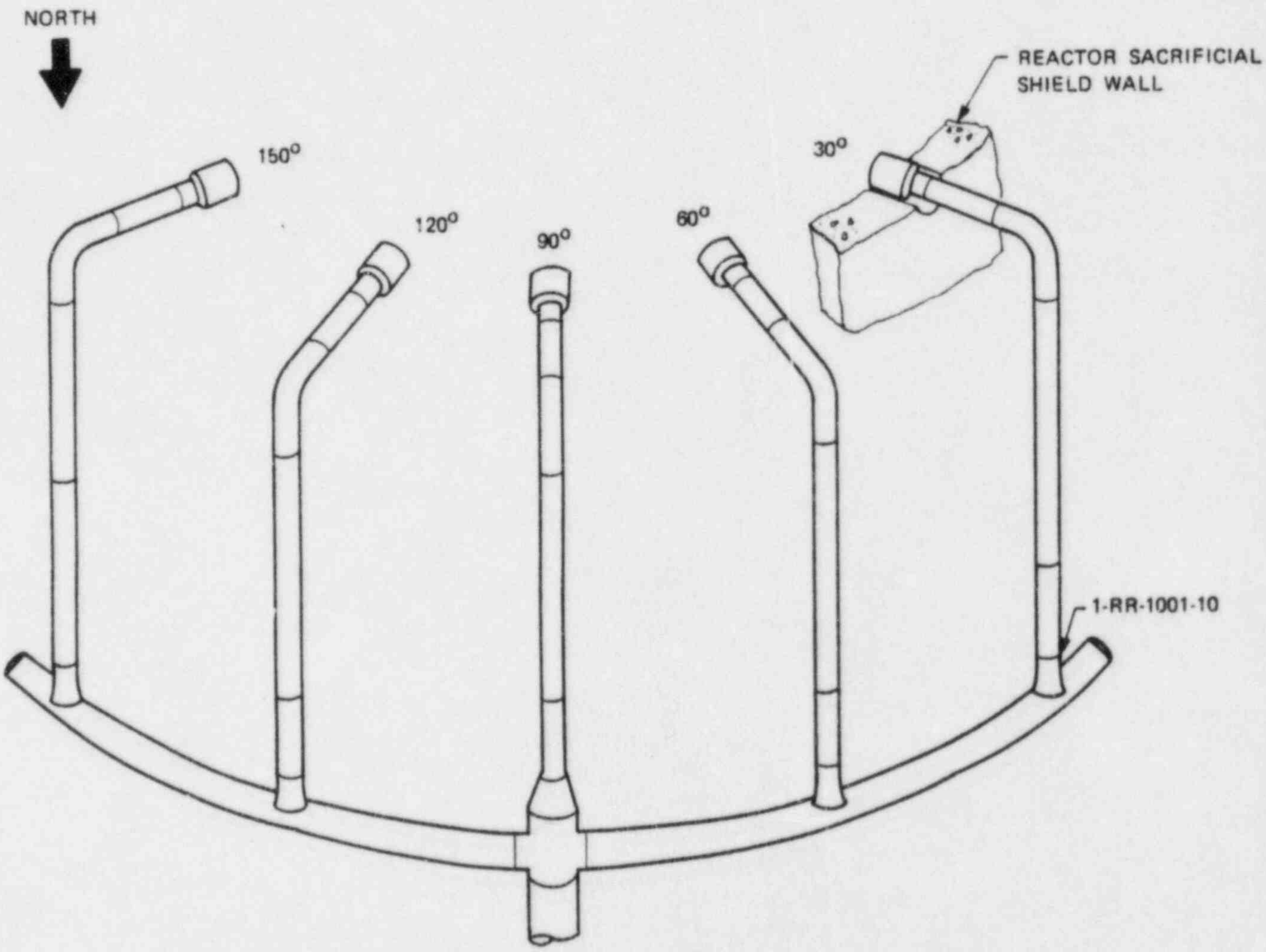
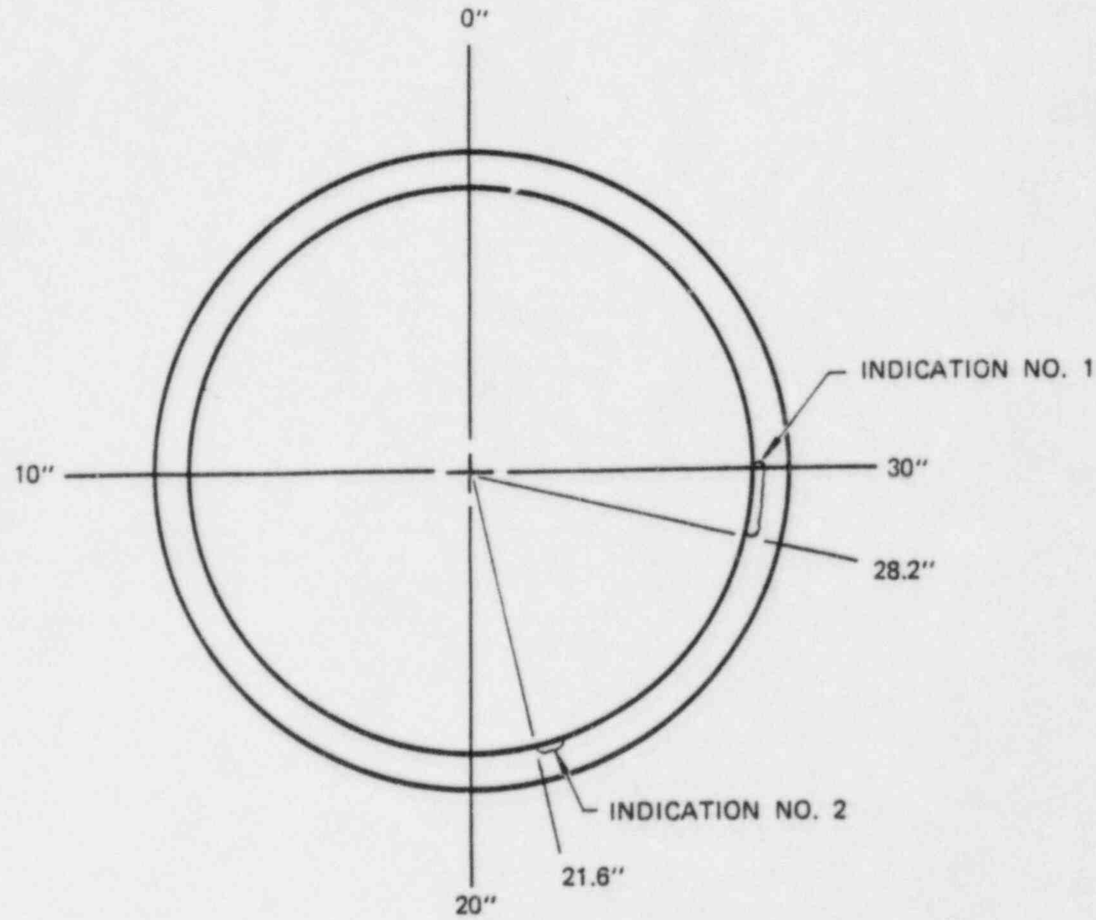


Figure 1.0-1

LASALLE UNIT 1
REACTOR RECIRCULATION SYSTEM
LOOP A
(Reference 1)



Indication No.	Length (in.)	Max. Depth (% Wall Thk.)	Characterization
1	2	22	Inside diameter surface planer flaw.
2	0.25	13	Inside diameter surface planer flaw.

Figure 1.0-2

LASALLE UNIT 1
WELD 1-RR-1001-10
FLAW DETAILS
(Reference 2)

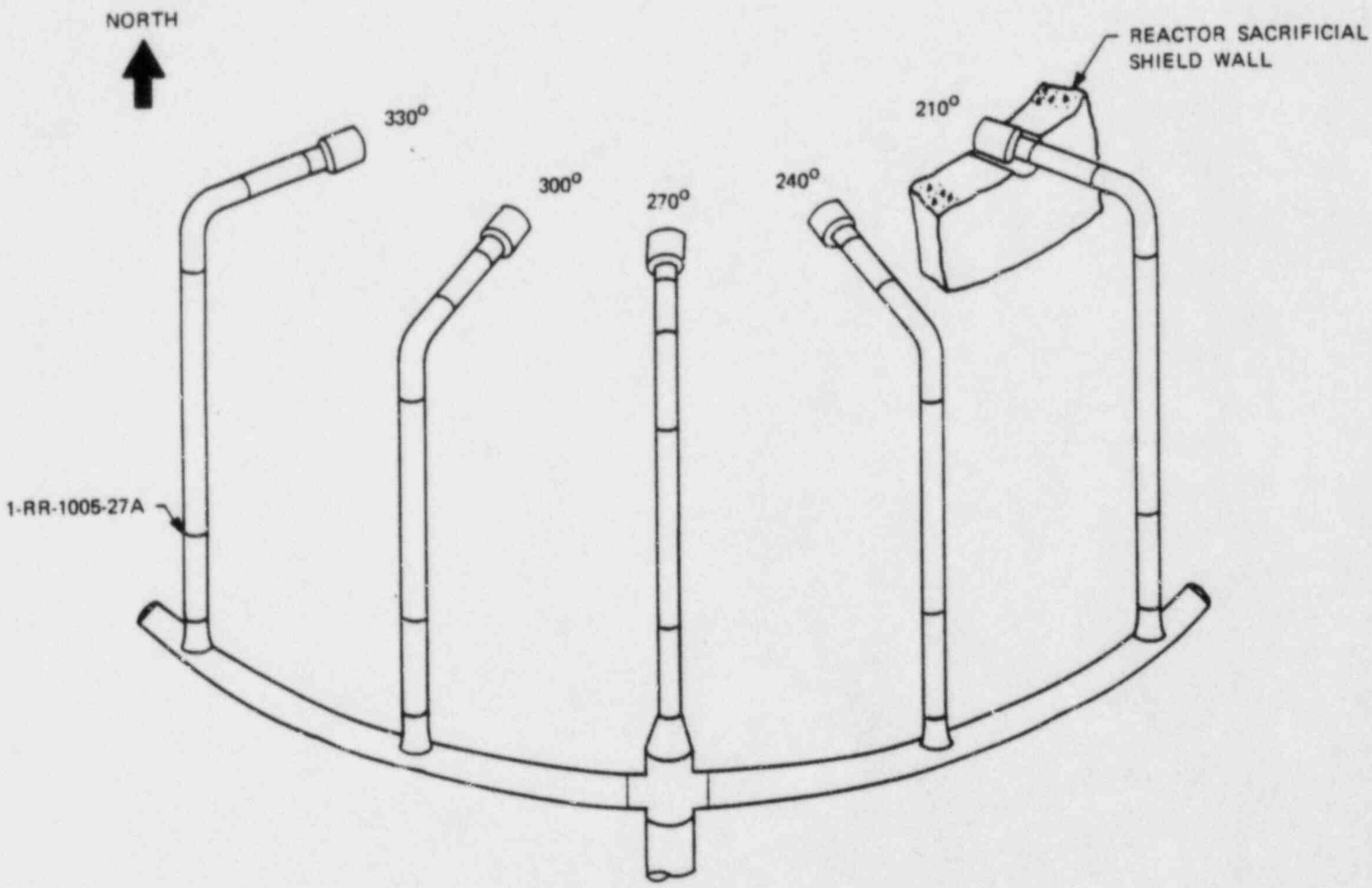
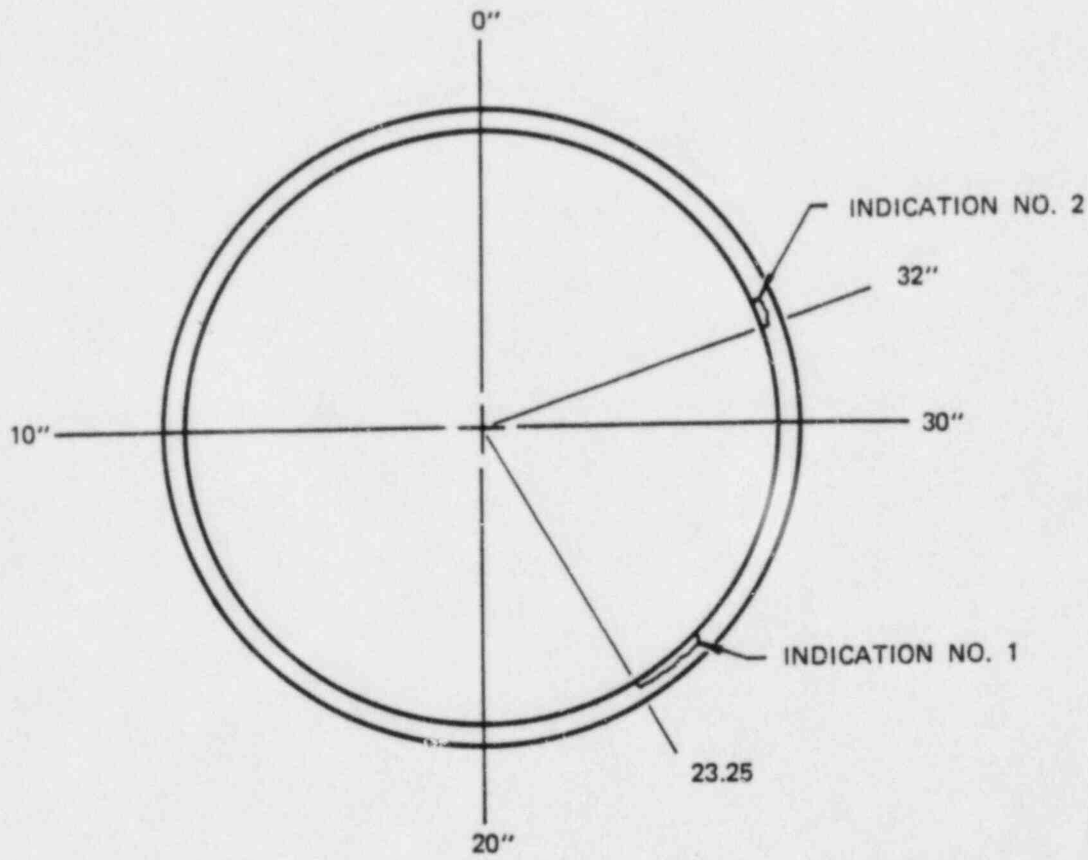


Figure 1.0-3

LASALLE UNIT 1
REACTOR RECIRCULATION SYSTEM
LOOP B
(Reference 3)



Indication No.	Length (in.)	Max. Depth (% Wall Thk.)	Characterization
1	1.75	28	Inside diameter surface planer flaw.
2	0.6	14	Inside diameter surface planer flaw.

Figure 1.0-4

LASALLE UNIT 1
WELD 1-RR-1005-27A
FLAW DETAILS
(Reference 4)

2.0 EVALUATION CRITERIA

The following criteria were used by NUTECH to justify further operation of LaSalle Unit 1 with the assumed defects in Welds 1-RR-1001-10 and 1-RR-1005-27A:

1. The beginning-of-fuel cycle (evaluation period) flaw sizes used in the analyses were the as-measured flaw depths presented in Figures 1.0-2 and 1.0-4 by a conservative 360° circumferential length.

2. The prediction of end-of-fuel cycle (evaluation period) flaw sizes was based upon a conservative crack growth law which closely agrees with the NRR curve presented in Figure 2.0-1 from NUREG-1061, Volume 1 (Reference 5) using a combination of dead weight, internal pressure, and thermal expansion loads.

3. The calculation of IGSCC flaw growth was based upon conservative IHSI-mitigated through-wall residual stress distributions.

4. As currently required by USNRC Generic Letter 84-11 (Reference 6), the predicted end-of-fuel cycle (evaluation period) flaw size was compared to 2/3 of the ASME Section XI (Reference 7) Table IWB-3641-1 allowable flaw depth values for a combination of dead weight, internal pressure, and seismic loads.

5. Because the allowable flaw sizes in ASME Section XI Paragraph IWB-3640 are currently being revised to take account of the low fracture toughness associated with flux welds, the predicted end-of-fuel cycle (evaluation period) flaw size was also compared to the following criteria:

a. Based upon the USNRC Safety Evaluation for the Quad Cities Unit 2 Reload No. 7 refueling outage (Reference 8), the end-of-cycle flaw sizes were compared to 2/3 of the ASME Section XI Table IWB-3641-1 allowable flaw depth values for a combination of dead weight, internal pressure, seismic, <u>and</u> thermal expansion loads.

b. Based upon proposed ASME Section XI Code committee changes to IWB-3640, the end-of-cycle flaw sizes were compared to proposed Table IWB-3641-5 (Reference 9) allowable flaw depth values for a combination of dead weight, internal pressure, seismic, and thermal expansion loads.

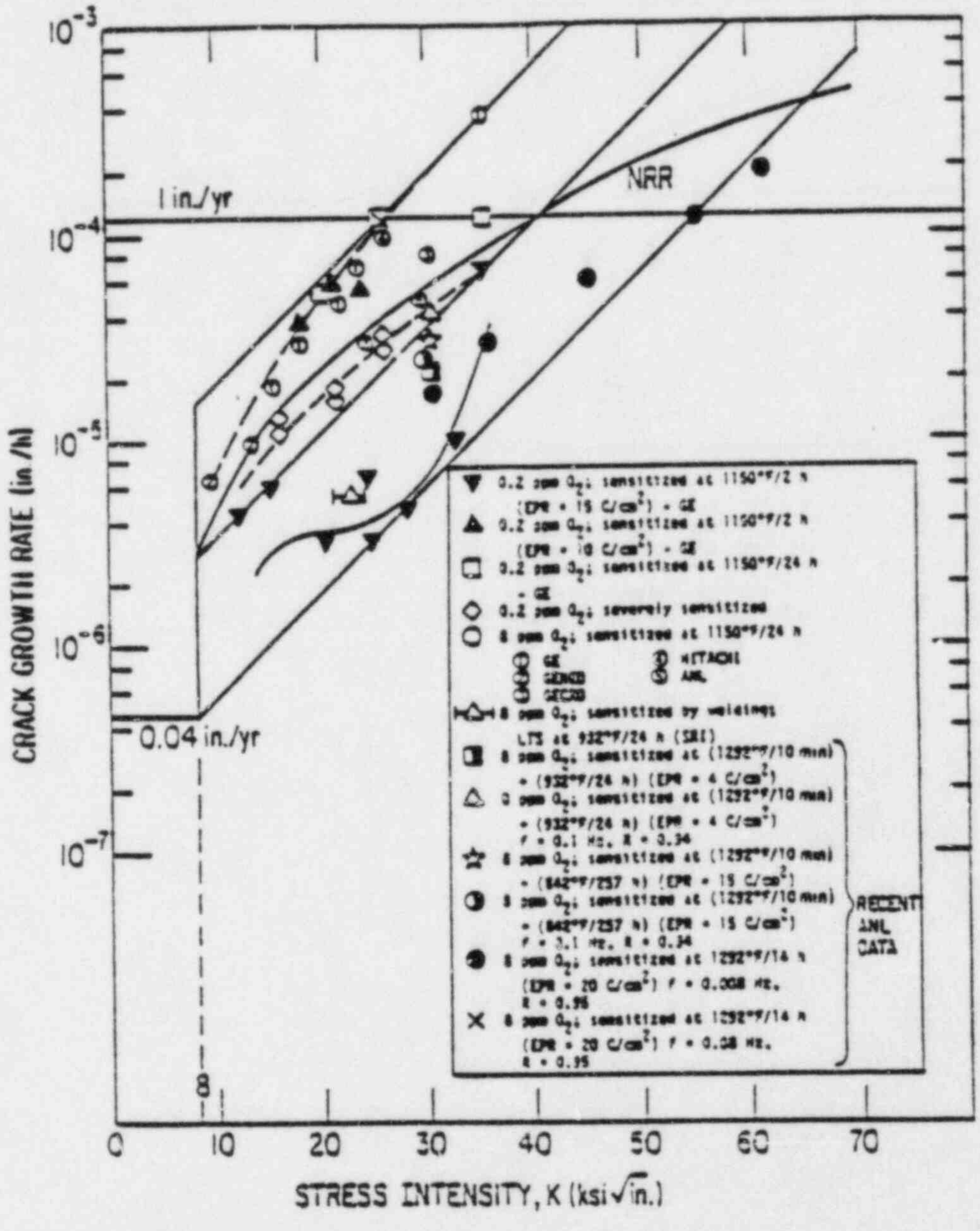


Figure 2.0-1

NUREG-1061, VOLUME 1
STRESS-CORROSION CRACK GROWTH RATES
(Reference 5)

3.0 APPLIED AND RESIDUAL STRESSES

In the calculation of predicted IGSCC flaw growth expected during a given period of time, sustained stresses acting on a flawed weldment must first be determined. These stresses include dead weight, internal pressure, piping system thermal expansion, and welding residual stresses. To determine if the end-of-evaluation period cracked pipe cross-section is within Code safety margins under applied loads, the magnitude of primary piping system stresses including dead weight, internal pressure, and seismic must be determined and, to satisfy recent evaluation criteria, the magnitude of secondary stresses including piping system thermal expansion must also be determined. This section presents the stresses used to evaluate the acceptability of flawed Welds 1-RR-1001-10 and 1-RR-1005-27A at LaSalle Unit 1.

3.1 Primary Stresses

Dead weight, internal pressure, and seismic primary piping system stresses were obtained from GE Design Reports 22A7426 and 22A7427 for the LaSalle Unit 1 recirculation system piping (References 10 and 11). Table 3.1-1 summarizes the stress values used to evaluate Welds 1-RR-1001-10 and 1-RR-1005-27A.

3.2 Secondary Stresses

Secondary stresses due to piping system thermal expansion were obtained from GE Design Reports 22A7426 and 22A7427. Table 3.1-1 contains these stresses for Welds 1-RR-1001-10 and 1-RR-1005-27A. Residual stresses due to IHSI-mitigation were obtained from EPRI Document NP-2662-LD (Reference 12). Figure 3.2-1 presents the axial and hoop residual stress distributions used in the LaSalle Unit 1 flawed weld evaluations.

Table 3.1-1

LASALLE UNIT 1
FLAWED WELD EVALUATION APPLIED STRESSES

Weld ID	Internal Pressure (psi)	Dead Weight + Seismic* (psi)	Thermal Expansion (psi)
1-RR-1001-10	7,782	838	4,980
1-RR-1005-27A	7,782	644	6,393

* Operating basis earthquake (OBE)

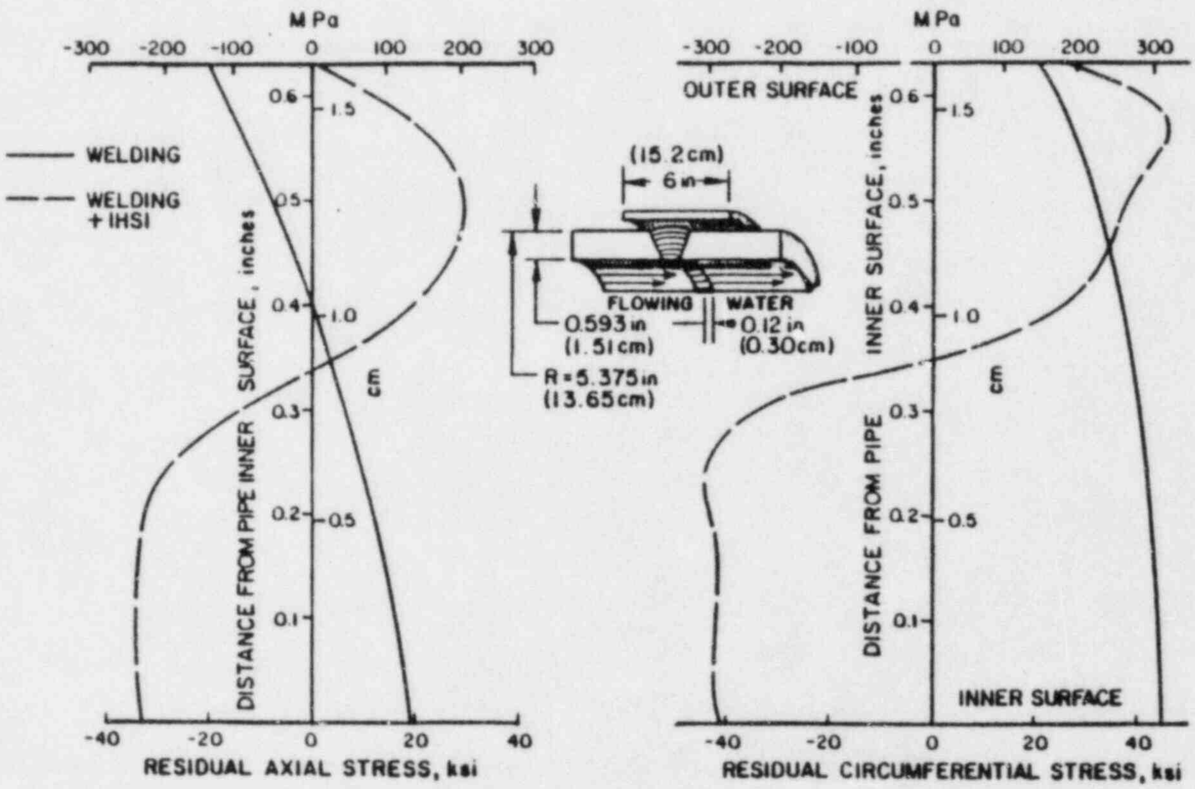


Figure 3.2-1

PRE- AND POST-IHSI THROUGH-WALL
RESIDUAL STRESS DISTRIBUTIONS
(Reference 12)

4.0 EVALUATION METHODS AND RESULTS

This section presents the evaluation methods and results used to assess the acceptability of the IHSI-mitigated assumed flaws at LaSalle Unit 1 for Welds 1-RR-1001-10 and 1-RR-1005-27A.

4.1 Crack Growth Analysis

Table 4.1-1 presents the pipe and flaw geometric details and sustained stress combinations needed to predict crack growth in the LaSalle Unit 1 flawed welds. NUTECH's NUTCRAK computer program (Reference 13) was used to predict crack growth using the following conservation crack growth law:

$$\frac{da}{dt} = 3.58 \times 10^{-8} K^{2.161}$$

Where:

da = differential crack size (inches)
dt = differential time (hours)
K = applied stress intensity factor ($ksi\sqrt{in}$)

As discussed in Section 2.0, this crack growth law closely agrees with the NRR curve presented in Figure 2.0-1 from NUREG-1061, Volume 1 (Reference 5).

Table 4.1-2 presents the predicted end-of-fuel cycle flaw depths for Welds 1-RR-1001-10 and 1-RR-1005-27A. As seen in the table, no growth is predicted during the next 18-month fuel cycle. In addition, the evaluation indicates that no IGSCC crack growth is expected for the balance of plant life.

4.2 Flawed Pipe Evaluation

As discussed in Section 2.0, the predicted end-of-fuel cycle flaw depths for the LaSalle Unit 1 flawed welds were compared to three different evaluation criteria. Table 4.2-1 presents flaw geometric details and primary stress combinations needed to evaluate the requirements of USNRC Generic Letter 84-11 (Reference 6) and ASME Section XI (Reference 7) Table IWB-3641-1. Table 4.2-2 presents flaw geometric details and primary plus secondary stress combinations needed to evaluate the requirements of the USNRC Safety Evaluation for Quad Cities Unit 2 (Reference 8) and ASME Section XI Table IWB-3641-1. Table 4.2-3 presents flaw geometric details and primary plus secondary stress combinations needed to evaluate the requirements of proposed ASME Section XI Table IWB-3641-5 (Reference 9).

Table 4.1-1

LASALLE UNIT 1
PIPE AND FLAW GEOMETRIC DETAILS
AND SUSTAINED STRESS COMBINATIONS

Weld ID	Nominal O.D.(1) (in.)	t(2) (in.)	a(3) (in.)	ℓ(4)	Sustained Stress(5) (psi)
1-RR-1001-10	12.75	0.76	0.167	360°	13,600
1-RR-1005-27A	12.75	0.65	0.182	360°	14,819

Notes:

1. O.D. = outside diameter
2. t = pipe wall thickness
3. a = beginning-of-fuel cycle flaw depth
4. ℓ = evaluation flaw length
5. In addition to dead weight, internal pressure, and thermal expansion, sustained stress combinations conservatively include small contribution from OBE seismic.

Table 4.1-2

LASALLE UNIT 1
PREDICTED END-OF-FUEL CYCLE FLAW DEPTHS

Weld ID	Beginning-of-Fuel Cycle Flaw Depth Ratio[1]	End-of-Fuel Cycle Flaw Depth Ratio[2]
1-RR-1001-10	0.22	0.22
1-RR-1005-27A	0.28	0.28

Notes:

1. Beginning-of-fuel cycle flaw size used flaw depth ratio ($\frac{a}{t}$) from Table 4.1-1 and 360° circumferential length.

2. Predicted end-of-fuel cycle flaw depth based upon combination of dead weight, internal pressure, thermal expansion, and post-IHSI residual stresses.

Table 4.2-1

LASALLE UNIT 1
GENERIC LETTER 84-11/TABLE IWB-3641-1
PREDICTED VS. ALLOWABLE FLAW DEPTH RATIOS

Weld ID	ℓ[1] (in.)	FLR[2]	SR[3]	IWB-3641-1 FDR[4]	GL 84-11 FDR[5]	Predicted FDR[6]
1-RR-1001-10	2.25	0.06	0.51	0.75	0.50	0.22
1-RR-1005-27A	2.35	0.06	0.5	0.75	0.50	0.28

Notes:

1. Combined flaw lengths, ℓ, from Figures 1.0-2 and 1.0-4.
2. FLR = flaw length ratio = combined flaw length divided by nominal pipe circumference.
3. SR = dead weight plus internal pressure plus seismic stresses (Table 3.1-1) divided by allowable stress intensity, S_m. From ASME Section III (Reference 14) Appendix I, Table I-1.2, S_m = 16,950 psi for 304 stainless steel pipe and fittings at 550°F operating temperature (References 10 and 11).
4. FDR = flaw depth ratio ($\frac{a}{t}$) from ASME Section XI (Reference 7) Table IWB-3641-1.
5. Allowable flaw depth ratio ($\frac{a}{t}$ x 2/3) per USNRC Generic Letter 84-11 (Reference 6).
6. Predicted end-of-fuel cycle flaw depth ratio from Table 4.1-2.

Table 4.2-2

LASALLE UNIT 1
USNRC SAFETY EVALUATION/TABLE IWB-3641-1
PREDICTED VS. ALLOWABLE FLAW DEPTH RATIOS

Weld ID	ℓ[1] (in.)	FLR[2]	SR[3]	IWB-3641-1 FDR[4]	GL 84-11 FDR[5]	Predicted FDR[6]
1-RR-1001-10	2.25	0.06	0.80	0.75	0.50	0.22
1-RR-1005-27A	2.35	0.06	0.87	0.75	0.50	0.28

Notes:

1. Combined flaw lengths, ℓ, from Figures 1.0-2 and 1.0-4.
2. FLR = flaw length ratio = combined flaw length divided by nominal pipe circumference.
3. SR = dead weight plus internal pressure plus seismic plus thermal expansion stresses (Table 3.1-1) divided by allowable stress intensity, S_m, defined in Note 3, Table 4.2-1.
4. FDR = flaw depth ratio ($\frac{a}{t}$) from ASME Section XI (Reference 7) Table IWB-3641-1.
5. Allowable flaw depth ratio ($\frac{a}{t}$ x 2/3) per USNRC Safety Evaluation for Quad Cities Unit 2 (Reference 8).
6. Predicted end-of-fuel cycle flaw depth ratio from Table 4.1-2.

Table 4.2-3

LASALLE UNIT 1
PROPOSED TABLE IWB-3641-5
PREDICTED VS. ALLOWABLE FLAW DEPTH RATIOS

Weld ID	ℓ[1] (in.)	FLR[2]	SR[3]	IWB-3641-5 FDR[4]	Predicted FDR[5]
1-RR-1001-10	2.25	0.06	0.66	0.6	0.22
1-RR-1005-27A	2.35	0.06	0.5	0.6	0.28

Notes:

1. Combined flaw lengths, ℓ, from Figures 1.0-2 and 1.0-4.
2. FLR = flaw length ratic = combined flaw length divided by nominal pipe circumference.
3. SR = M [(dead weight plus internal pressure plus seismic stresses) + (thermal expansion stresses divided by 2.77)] divided by allowable stress intensity, S_m, defined in Note 3, Table 4.2-1. Used worst M = 1.08 for SAW weldment less than 24 inches in diameter.
4. FDR = flaw depth ratio $(\frac{a}{t})$ from proposed ASME Section XI Table IWB-3641-5 (Reference 9).
5. Predicted end-of-fuel cycle flaw depth ratio from Table 4.1-2.

5.0 SUMMARY AND CONCLUSIONS

LaSalle County Nuclear Station Unit 1 recently completed IHSI of Reactor Recirculation and Residual Heat Removal system welds. Post-IHSI ultrasonic examinations revealed linear indications at Welds 1-RR-1001-10 and 1-RR-1005-27A. For evaluation purposes, these indications were assumed to be IGSCC.

The crack growth analyses and flawed pipe evaluations presented in this report demonstrate that the original design margins of safety inherent in the Code for the flawed welds have not been degraded. In addition, the analysis indicates that the IHSI-mitigated IGSCC flaws are not expected to grow during the next fuel cycle or, indeed, for the balance of plant life.

6.0 REFERENCES

1. Morrison Construction Company Drawing I-RR-1001, "Commonwealth Edison Company LaSalle County Station Unit 1 - Inservice Inspection - Reactor Recirculation Loop A," Revision E.

2. General Electric Document, "Weld Evaluation Summary," Weld ID No. 1-RR-1001-10, dated January 16, 1986.

3. Morrison Construction Company Drawing I-RR-1005, "Commonwealth Edison Company LaSalle County Station Unit 1 - Inservice Inspection - Reactor Recirculation Loop B," Revision D.

4. General Electric Document, "Weld Evaluation Summary," Weld ID No. 1-RR-1005-27A, dated January 16, 1986.

5. U.S. Nuclear Regulatory Commission Document No. NUREG-1061, Volume 1, "Investigation and Evaluation of Stress-Corrosion Cracking in Piping of Boiling Water Reactor Plants," April 1984, Second Draft attached to SECY-84-301, dated July 30, 1984.

6. USNRC Generic Letter 84-11, "Inspections of BWR Stainless Steel Piping," April 19, 1984.

7. ASME Boiler and Pressure Vessel Code Section XI, 1983 Edition with Addenda through Winter 1983.

8. USNRC Document, "Safety Evaluation by the Office of Nuclear Reactor Regulation - Inspection and Repair of Reactor Coolant System Piping at Quad Cities Unit 2," attached to J. A. Zwolinski (USNRC) letter to D. L. Farrar (CECo), dated January 7, 1986.