SL-5060, Rev. 0
Safety Related

Quad Cities Unit 1
Core Spray Flaw Evaluation Report

SL- 5060, Rev. 0

March 28, 1996

Prepared By: Phuong huy Hoang
Dr. Phuong Hoang, Senior Project Engineer

Thomas J. Behringer
T. J. Behringer, PE, SE - Senior Project Engineer

Reviewed By: C. Michael Launi
C. M. Launi, PE - Senior Project Engineer

Thomas J. Behringer for A. AlDabbagh
Dr. A. AlDabbagh, SE - Principal Engineer

Approved By: Thomas J. Behringer
Thomas J. Behringer, PE, SE - Senior Project Engineer

ComEd

SARGENT & LUNDY CERTIFICATION OF REPORT NO. SL-5060

REV. 0

Quad Cities Unit 1
Core Spray Flaw Evaluation Report

I certify that this Design Document was prepared by me or under my supervision and that I am a Registered Engineer under the laws of the State of Illinois.

Thomas J. Behringer 3/28/96

Thomas J. Behringer
Senior Project Engineer

Quad Cities Unit 1
Core Spray Flaw Evaluation Report

DESIGN REPORT TABLE OF CONTENTS

		Page
1.0	EXECUTIVE SUMMARY	6
2.0	INTRODUCTION	7
3.0	DESCRIPTION OF INDICATIONS	10
4.0	MATERIALS EVALUATION	18
5.0	LOAD DEFINITIONS AND LOAD COMBINATIONS	20
6.0	CORE SPRAY PIPING MODELLING AND ANALYSIS	26
7.0	FLAW EVALUATIONS	29
8.0	LEAKAGE FLOW EVALUATIONS	38
9.0	CORE SPRAY SYSTEM LOCA EVALUATION	42
10.0	BOUNDING FAILURE ASSESSMENT	45
11.0	LOOSE PARTS EVALUATION	50
12.0	SUMMARY AND CONCLUSIONS	54
13.0	REFERENCES	57

LIST OF TABLES

Table		Page
3.1	Summary of Flaw Lengths	17
7.1	Flaw Evaluation Stress Values	31
7.2	Flaw Evaluation Results	33
7.3	Flaw Evaluation Sensitivity Analysis - Stress Values	37
7.4	Flaw Evaluation Sensitivity Analysis - Results	37

LIST OF FIGURES

Figure		Page
2.1	Core Spray Piping Inside the RPV Annulus	9
3.1	Core Spray A-Loop 290° Upper Sparger.Inlet Thermal Sleeve and Elbow	11
3.2	Core Spray B-Loop 260° Lower Sparger Inlet Thermal Sleeve and Elbow	12
3.3	Ultrasonic Test Examination Area and Flaw Location Elbow to Thermal Sleeve Weld 17	15
3.4	Ultrasonic Testing Examination Area and Flaw Location Elbow to Thermal Sleeve Weld 3	16
6.1	Core Spray Piping Analysis Model	27
7.1	Cross Section of Flawed Pipe Section	34
8.1	Leakage Rate at a Differential Pressure of 47 psi	40
8.2	Leakage Rate at a Differential Pressure of 64 psi	41

1.0 Executive Summary

Cracks were observed at two locations on the core spray downcomers during the Q1R14 planned in-vessel visual inspection covering 100% of the internal core spray piping header and downcomer welds. This flaw evaluation report provides a summary of the design criteria, design inputs and results of the conservative evaluations performed to assess the extent, causes and impact of the cracking on safety and plant operation. The cracks are typical for IGSCC in austenitic stainless steel. ComEd has evaluated the maximum impact of leakage on the peak cladding temperature during the DBA-LOCA in combination with the bounding single failure. This evaluation found that the peak cladding temperature during DBA-LOCA would remain below 2200° F. In addition, beyond-design-basis bounding assessments using both the deterministic and probabilistic approach were made. These bounding assessments found that, even with an assumed 360° failure of any one of the four core spray downcomers, adequate core cooling would be maintained when subjected to all design basis events. The worst case scenarios (reactor recirculation suction line failure combined with LPCI failure or reactor recirculation suction failure combined with LPCI failure and a SSE) present an insignificant risk since their probabilities are much less than 1x 10^{-6}/year. Failure of a core spray downcomer could potentially result in a loose part and debris within the vessel. ComEd has evaluated the impact of the loose parts and debris, and since the largest pieces would be confined to the annulus region, no safety concerns were identified. ComEd will continue to monitor the condition of the degraded core spray welds by following the BWRVIP guidance for the next scheduled refueling outage, Q1R15, currently scheduled for the Spring of 1998.

2.0 INTRODUCTION

The portion of the core spray line addressed by this condition assessment is located in the reactor pressure vessel (RPV) annulus of Quad Cities Unit 1. The RPV annulus portion of the core spray lines consists of two symmetrical loops with RPV penetrations at the 5° and 185° azimuths. These two loops feed the two upper and two lower core spray spargers through four core shroud penetrations. A representative example of the RPV annulus portion of the core spray system is illustrated in Figure 2.1.

In February of 1996 Quad Cities Site Engineering initiated the Q1R14 planned In-Vessel Visual Inspections (IVVI) of the core spray piping. The inspections were performed in accordance with the enhanced visual inspection technique (e.g., cleaning, 0.5 mil resolution and 1"-3" camera to subject distance). On February 22, 1996, cracking was detected in the core spray piping downcomers. Subsequently, on March 14, 1996, Ultrasonic Examinations (UT) were performed on the RPV Internal Core Spray Piping at Quad Cities Unit 1. The purpose of these examinations was to confirm the visual indications found in the HAZ of the 260° and the 290° core spray downcomer elbow to thermal sleeve welds.

Crack like indications were confirmed at thermal sleeve weld number 17 on the A-Loop and thermal sleeve weld number [illegible] the B-Loop. The location of the cracks are on the A-Loop's upper sparger inlet thermal elbow sleeve at 290° and at the B-Loop's lower sparger inlet thermal sleeve elbow at 260°. The seam elbow is made of round stainless steel and is 6 inches in diameter. The affected piping is located in the reactor annulus between the reactor vessel wall and the core shroud wall. The elbow is not part of the reactor coolant pressure boundary nor is it part of the core shroud. The adjacent thermal sleeve collars are attached to the shroud through the T-Box on one side and to the 6 inch diameter core spray line on the other side. The thermal sleeve collars have the function of providing a leakage seal between the pipe and the annulus as shown in Figures 3.1 and 3.2. All indications are located in the elbow section and are circumferentially oriented

The main analytical approach used here to justify continued operation, is a limit load analysis which only requires primary loads in the evaluation. However, additional sensitivity studies were also performed in which secondary loads from the thermal, seismic and LOCA events are included. The loads used to evaluate these flaws were developed from a piping analysis model of the Quad Cities core spray system. This report provides the analysis details including the assessment criteria, design inputs and results for the various evaluations performed for the core spray cracking identified during the Q1R14 in-vessel inspections. Section 3 of this report provides a summary of the method and extent of the examinations performed as well as the detailed definition of the indications identified. Section 4 provides the materials evaluation with an assessment of the root cause and definition of material properties and the crack growth rate used in the flaw evaluation. The definition of the loading cases and load combinations used are provided in Section 5. A detailed description of the core spray line modeling and analyses along with a summary of the results is provided in Section 6. The flaw and leakage evaluations are described in Sections 7 and 8, respectively. Section 9 provides a description of the core spray system LOCA evaluation. Failure assessments and loose parts evaluations are included in Sections 10 and 11, respectively. A summary of the results and conclusions is provided in Section 12, while the references are presented in Section 13.

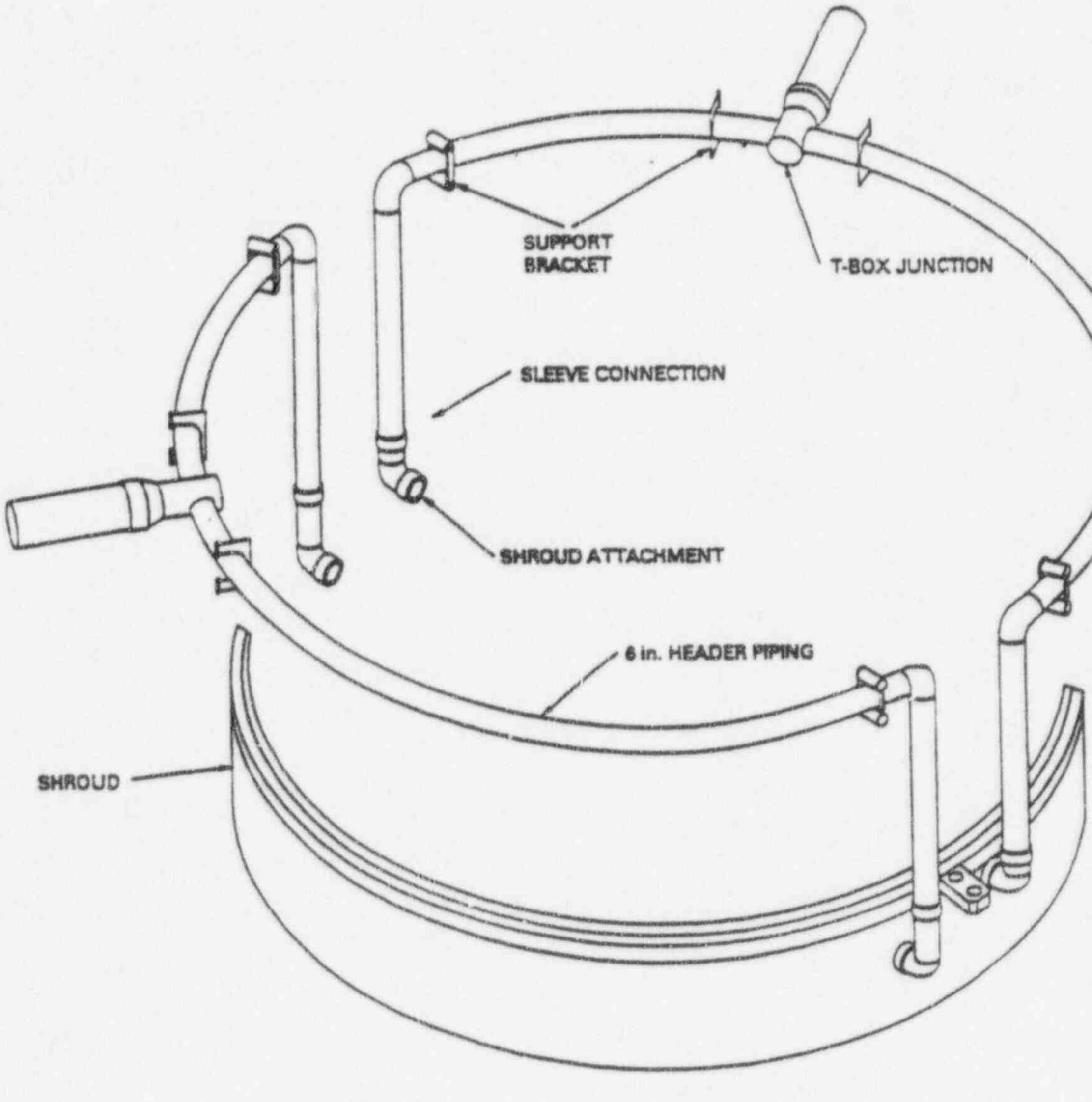


Figure 2.1 Core Spray Piping in the RPV Annulus

3.0 DESCRIPTION OF INDICATIONS

3.1 Visual Inspections

An enhanced visual inspection (0.5 mil wire resolution) was performed during the Q1R14 refuel outage, of each of the internal core spray piping header and downcomer welds. Crack indications were identified in two locations (Reference 1). A description of the flaws identified, along with the length of each flaw as determined by visual sizing techniques is provided below. It should be noted that the flaws were very tight and that a best effort approach was used to visually size the lengths and locations of the flaws. The intent of the subsequent UT examinations was to identify any undetected ID cracking and to provide a more accurate measurement of the length of the OD crack indications. The crack orientations as described below are given from the perspective of standing at the RPV and facing the core shroud.

3.1.1 Core Spray 290° Upper Sparger Inlet Elbow to Thermal Sleeve Weld

The IVVI examinations identified an indication in the HAZ at the outside radius of the elbow at Weld #17. The visual length of this indication was approximately 2.5" to 3.5" long measured from the bottom of the elbow seam in the counter clockwise direction, and approximately 3" to 4" long in the clockwise direction as shown in Figures 3.1 and 3.3.

3.1.2 Core Spray 260° Lower Sparger Inlet Elbow to Thermal Sleeve Weld

The IVVI examinations identified an indication in the HAZ at the outside radius of the elbow at Weld #3. The visual length of this indication was estimated to be approximately 2.5" to 3.5" long measured from the bottom of the elbow seam in the counter-clockwise direction and 1.0" in the clockwise direction as shown in Figures 3.2 and 3.4.

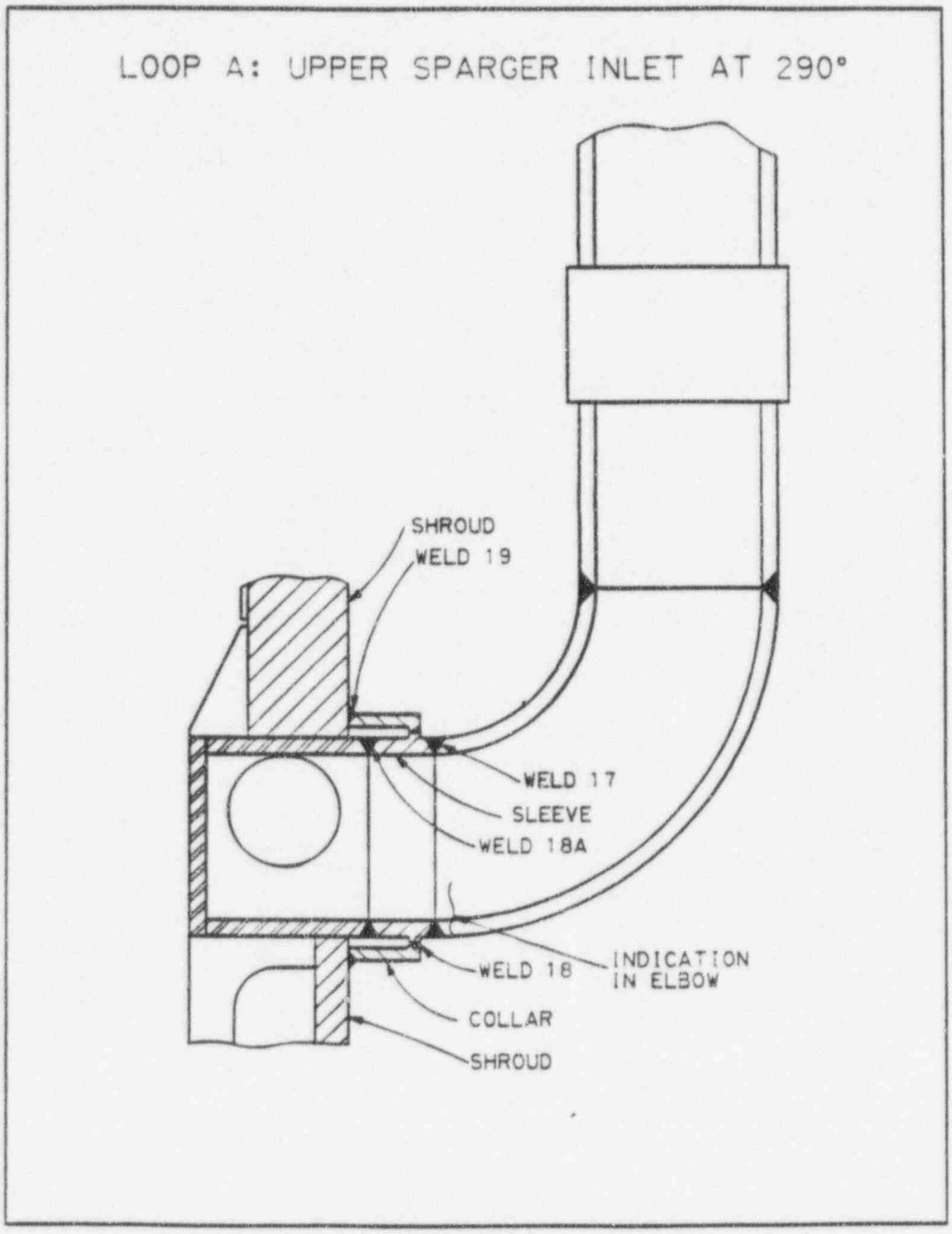


Figure 3.1 Core Spray A-Loop 290° Upper Sparger Inlet Thermal Sleeve and Elbow

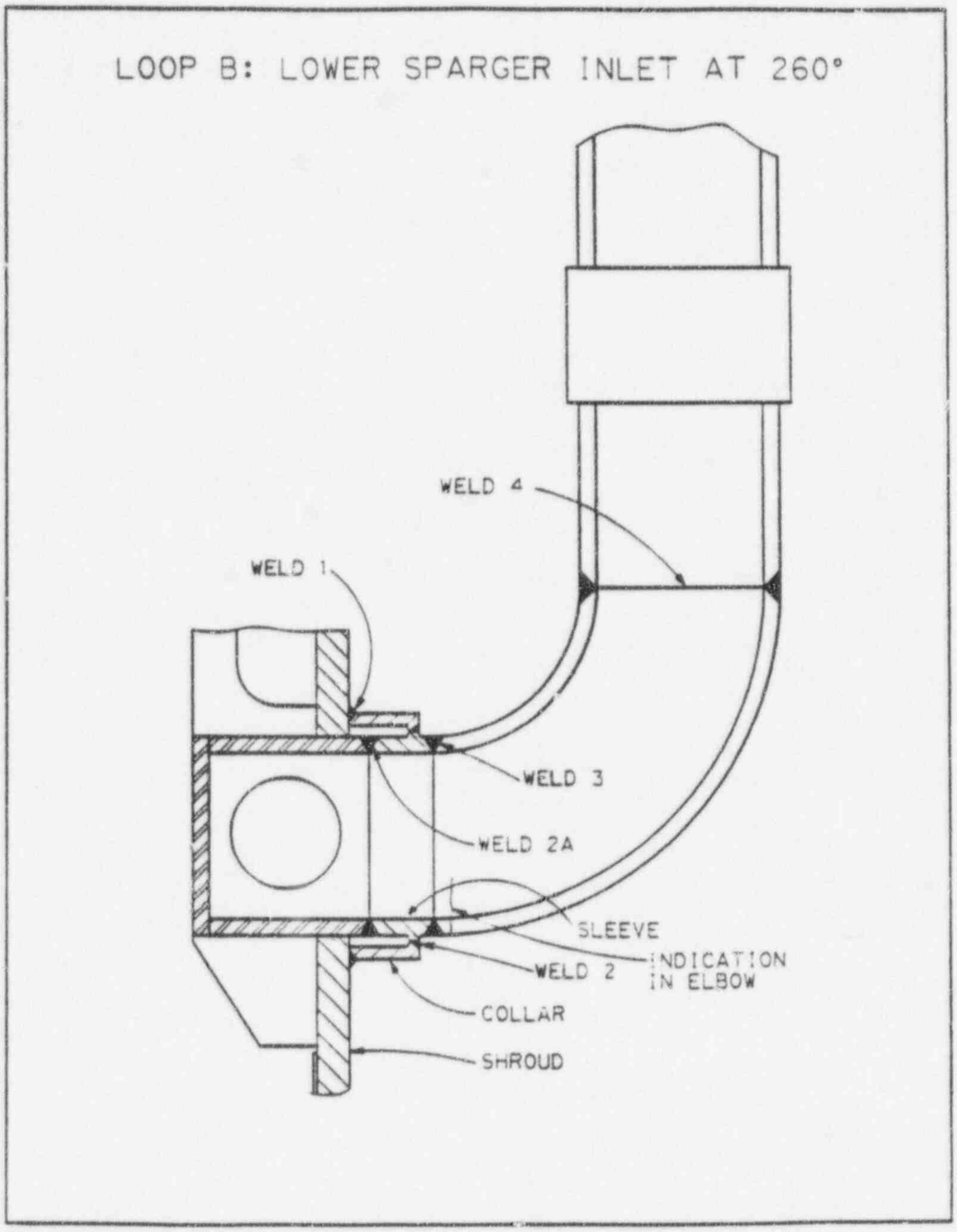


Figure 3.2 Core Spray B-Loop 260° Lower Sparger Inlet Thermal Sleeve & Elbow

3.2 Ultrasonic Examination

The purpose of the ultrasonic (UT) examination was to characterize the length of the OD connected flaws which were first detected with an enhanced visual inspection and to detect any ID connected flaws which may be less than through wall in depth. The ultrasonic technique used for the examination of the visually detected flaws in the core spray piping was originally developed by GE for use at Dresden during the D2R14 inspections. The technique was previously qualified at Dresden using mockups. The ultrasonic examinations were performed utilizing the "Smart 2000" ultrasonic data acquisition system. The core spray piping examination fixture was affixed with 3 separate 80° refracted longitudinal wave (OD Creeping Wave) search units. The UT technique and qualification process was independently reviewed by EPRI (Reference 2) and is described briefly in Section 3.2.1.

The ultrasonic probes were attached to a semicircular fixture on a long pole. The fixture was manually positioned on the thermal sleeve collar and secured with actuators to "clamp" it in place to ensure proper coupling to the surface to be examined. The transducers were positioned on the elbow side of the weld with the sound beams directed back towards the thermal sleeve. The semicircular fixture was graduated in inches to aid in determining the relative location of the transducer when the crack tip was located.

3.2.1 UT Qualification

The qualifications performed at Dresden (Reference 2) using the probes and mockups established that the technique is effective in determining the presence or absence of cracking. This technique also proved to be useful in verifying the endpoints of the visually-detected flaws, whether the flaw extremities were connected to the inside or the outside surface. However, it was not practical to measure crack depth or to positively distinguish inside-surface from outside-surface cracking because of the thin wall. Provided below and in Figures 3.3 and 3.4 is a description of the results of the UT examination (Reference 3). This UT length sizing approach using the UT fixture as a measuring tool, is considered to be more accurate than visual sizing techniques used during the initial enhanced visual inspection.

3.2.2 Core Spray 290° Upper Sparger Inlet Elbow to Thermal Sleeve Collar Weld #17

The UT examination of this location, Figure 3.3, confirmed the presence of the crack. The crack was identified by UT to extend from 110° to 230° . The UT was able to identify both ends of the flaw. The flaw UT length is the same as the visual length on the OD but slightly skewed with respect to the controidal axis. The UT examination did not cover the area from 330° to 30° due to obstruction. However, no visual indication was found in this area.

3.2.3 Core Spray 260° Lower Sparger Inlet Elbow to Thermal Sleeve Collar Weld #3

The flaw UT length starts from 110° and extends to 180° on the elbow side HAZ. The flaw length of 4.6" is longer than the visual length of 2.5" - 3.5". The 1" long visual length in the clockwise direction from the seam weld was not observed by the UT examination due to the search unit lift off at the seam weld crown. No flaws were observed clockwise from the 1" location to the 100° position during the UT examination. The upperbound lengths of the UT and visual results were selected for use in performing the flaw evaluations.

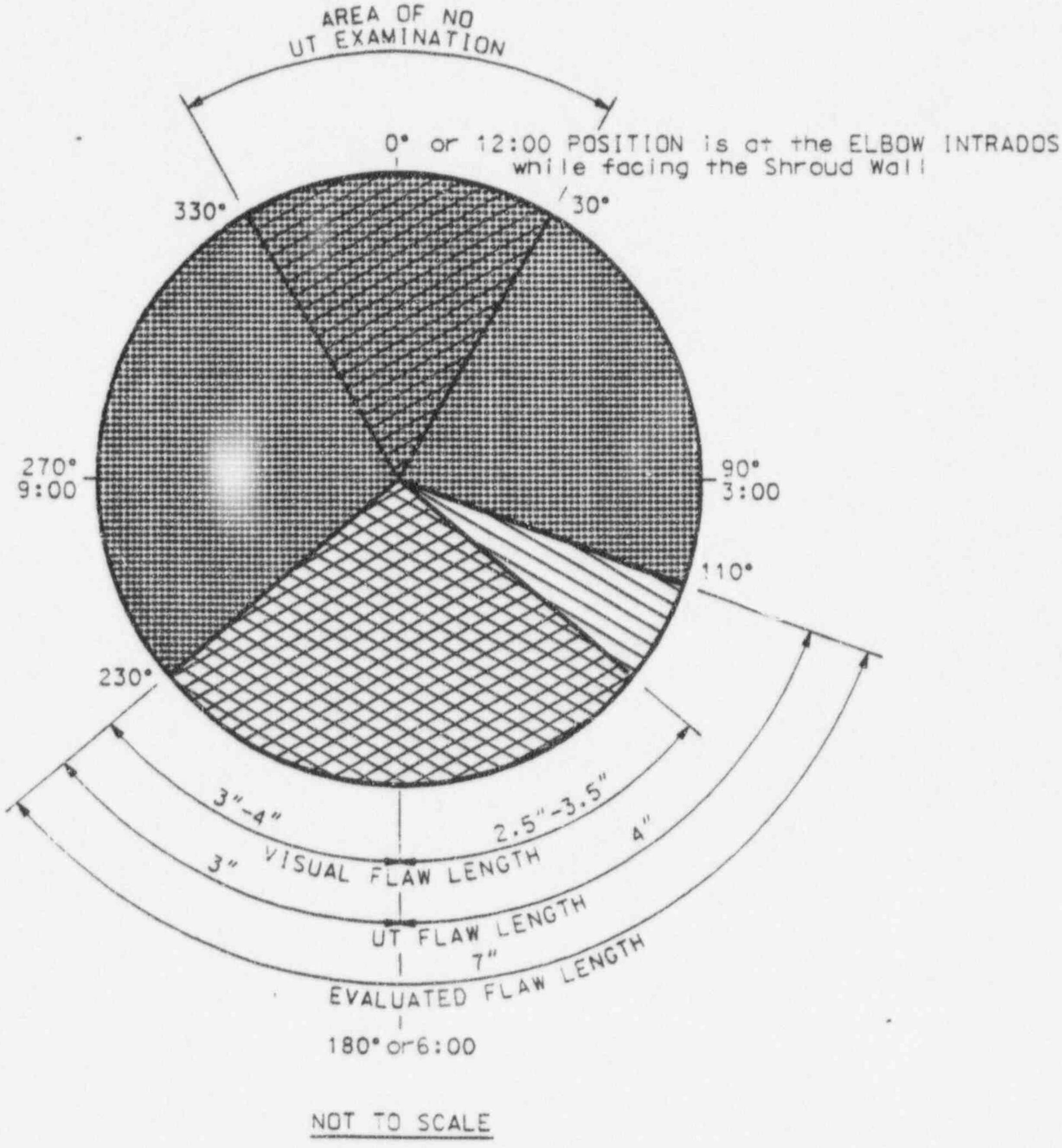
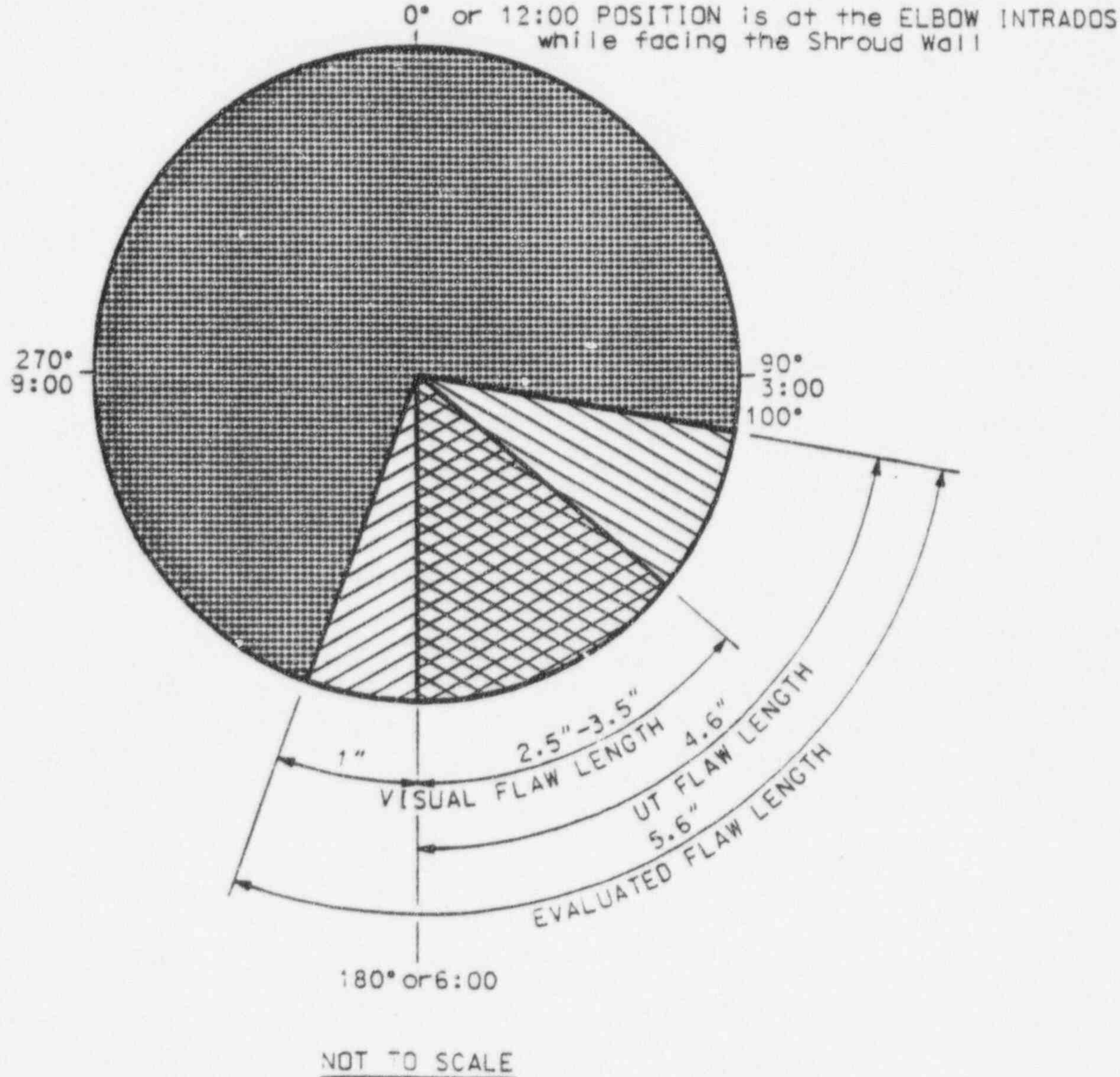


Figure 3.3 Ultrasonic Testing Examination Area And Flaw Location Elbow to Thermal Sleeve Weld 17



NOT TO SCALE

Figure 3.4 Ultrasonic Testing Examination Area And Flaw Location Elbow to Thermal Sleeve Weld 3

3.3 Crack Growth Length

The flaw lengths as determined by the VT and UT examinations were increased by a crack growth length to establish the evaluated flaw length (EFL). A crack growth length for an evaluation period of a 24 month hot operating cycle with a 100% availability factor was added to both ends of the flaw. A summary of the flaw lengths evaluated is provided in Table 3.1.

Table 3.1 Summary of Flaw Lengths

Flaw Location	Measured Flaw Length (Inches)[1]	Crack Growth Rate (Inches/ Hour)[2]	Crack Growth Length (Inches)[3]	Evaluated Flaw Length (Inches)[4]
Upper Sparger Elbow	7"	5.00 E-5	0.86"	8.72"
Lower Sparger Elbow	5.6"	5.00 E-5	0.86"	7.32"

Notes:

1. Measured flaw lengths are the bounding results obtained from the VT and UT examinations.

2. 5.00 E-5 Inches/Hour represents an upper bound conservative industry limit for IGSCC crack growth in ductile material (Reference 4).

3. Crack growth is based on 24 months or 17,280 hours of hot operation.

4. Evaluated Flaw Length (EFL) = Measured Length + 2(CGL)

4.0 MATERIALS EVALUATION

4.1 Overview

Cracks were found in the Heat Affected Zones (HAZ) of the elbow to thermal sleeve welds of the core spray spargers (see Figures 3.1 and 3.2). All of the cracks observed were initiated in the base material HAZ of the elbow. The thermal sleeve to elbow weld is a girth butt weld where large axisymetric residual stresses may be present. Consequently, both the appearance and location of the cracking is consistent with Intergranular Stress Corrosion Cracking (IGSCC). This particular degradation mechanism is well documented for stainless steel components exposed to the high temperature reactor water of BWRs. Several other BWRs including Quad Cities sister plants Dresden Units 2 and 3, have reported core spray piping cracks which were identified as IGSCC.

4.2 Fabrication

The General Electric Company (GE) design specifications as well as the fabricator records (Willamette) have been reviewed. All of the components are fabricated from solution heat-treated type 304 austenitic stainless steel ASTM A-312, Grade TP-304. The elbow is a seam welded elbow. Records indicate the elbow weld was fabricated with the Gas Tungsten Arc Welding (GTAW) process.

4.3 Crack Growth Rate

The principle driving force propagating IGSCC cracks comes from the weld residual stresses, because the applied loads during normal operation are insignificant. The residual stresses are self-relieving and will diminish as the crack extends. As the stress intensity factor at the tip of the growing crack drops below the threshold stress intensity for IGSCC (K_{IGSCC}), crack extension will stop. Therefore, the existing crack will propagate only as long as the residual stress field is sufficiently high to support crack propagation. These arguments suggest that a lower IGSCC crack growth rate may be justified. However, ComEd has used the currently accepted bounding crack growth rate of 5.00×10^{-5} inches/hour (Reference 4).

4.4 Material Behavior

The ductile or brittle response of the material of cracked core spray components is evaluated with respect to initial characteristics and environmental degradation. All of the materials used in fabrication, were austenitic stainless steels as indicated in Section 4.2 of this report. These materials do not undergo phase transformation during thermal processing. The most significant material response to thermal processing is grain boundary precipitation of chromium carbides, and this response produces a zone adjacent to the grain boundaries that is depleted in chromium. This condition is termed sensitization and can be produced during welding. This condition influences the electrochemical response of the material (increasing susceptibility to IGSCC), but does not alter the ductility or toughness of the material.

Exposure of austenitic stainless steels to irradiation can lead to a loss of ductility and an increased sensitivity to Irradiation Assisted Stress Corrosion Cracking (IASCC). The onset of IASCC occurs at approximately 5×10^{20} n/cm^2. The neutron fluence in the area of the core spray is less than 6×10^{18} n/cm^2, therefore, no reduction in toughness or increased sensitivity to IASCC is expected.

4.5 Conclusion

In conclusion, the cracking observed in the core spray system is the result of IGSCC in austenitic stainless steels. It has been shown in various tests that application of hydrogen water chemistry will be beneficial in slowing the crack growth rate. Use of this water chemistry strategy to mitigate IGSCC has been in place at Quad Unit 1 since 1990. The improved water chemistry coupled with the fact the stresses driving the cracking are residual stresses (self relieving) indicates that the rate of crack growth will be exceedingly slow. Therefore, the crack growth rate of 5×10^{-5} inches/hour represents a conservative upper bound limit. In addition, the material properties of the core spray system will remain ductile throughout the life of the system.

5.0 LOAD DEFINITIONS AND LOAD COMBINATIONS

5.1 Load Cases

DWGT	=	Dead Weight
TH01	=	Thermal 1 Normal Operation
TH02	=	Thermal 2 Feedwater Transient
TH03	=	Thermal 3 Core Spray - DBA Short Term (DBA1)
TH04	=	Thermal 4 Core Spray - DBA Intermediate Time Frame (DBA2)
TH05	=	Thermal 5 Core Spray - DBA Long Term (DBA3)
TH06	=	Thermal 6 Core Spray - ADS Blowdown, small and intermediate breaks
TH07	=	Thermal 7 HPCI Event - No Core Spray
P_1	=	Pressure 1 (Internal Piping Pressure - Normal = 0 PSID)
P_2	=	Pressure 2 (Internal Piping Pressure - Injection Mode = 47 PSID)
P_3	=	Pressure 3 (Internal Piping Pressure - Injection Mode = 64 PSID)
DWH1	=	Drag Load 1 (External Drag Loads on the Pipe Surface - Normal Flow)
DWH2	=	Drag Load 2 (External Drag Loads on the Pipe Surface - Recirculating Line Break Flow)
DWH3	=	Drag Load 3 (External Drag Loads on the Pipe Surface - Main Steam Line Break Flow)
OBDX	=	X Direction OBE Differential Seismic Displacement
OBDZ	=	Z Direction OBE Differential Seismic Displacement
OBDY	=	Y Direction OBE Differential Seismic Displacement
OBE1	=	OBE Response Spectra Analysis
SSDX	=	X Direction SSE Differential Seismic Displacement (2 x OBDX)
SSDZ	=	Z Direction SSE Differential Seismic Displacement (2 x OBDZ)
SSDY	=	Y Direction SSE Differential Seismic Displacement (2 x OBDY)
SSE2	=	SSE Response Spectra Analysis
SDIS	=	RRLB Core Shroud Displacement
INJF	=	Core Spray Injection Force

5.1.1 Dead Weight (DWGT)

The core spray piping from the RPV nozzle to the shroud penetrations consists of 6" nominal Outside Diameter (OD) schedule 40 pipe and 8" nominal OD schedule 40 pipe. The weight of the T-Box repair hardware

(160 lbs) is also included in the analysis. The piping is normally below the water level except for a LOCA event. In a LOCA event, the water level may drop below the core shroud penetrations. The weight of water contained inside the piping is included and the buoyancy force is conservatively neglected.

5.1.2 Thermal Expansion & Pressure Loads (TH01-06, PZER & PDES)

The radial and longitudinal differential thermal expansions of the RPV and the shroud are included in the thermal expansion analyses for the core spray piping. The radial dilation of the RPV under internal pressure is also considered for each thermal mode. Calculations for thermal displacements at support locations are documented in Section 3 and thermal mode definitions are in Section 6 of Reference 5.

Definition of Thermal Modes

Mode	Title	Pipe	RPV	Shroud	Annulus Water Temp	RPV (psig)
1	NORM. Oper	522	522	536	522	1050
2	FW TRANS	300	522	433	300	1050
3	CS-DBA1	195	522	536	270	27
4	CS-DBA2	195	522	270	270	27
5	CS-DBA3	179	232	232	232	7
6	CS-ADS	209	522	298	298	50
7	HPCI-NOCS	366	522	366	366	150

Normal Condition (TH01)

Temperature within the annulus region of the RPV is 522°F which is the temperature of Region B as specified in the Reactor Thermal Cycles diagram (Reference 6). The temperature of the shroud (536°F) is taken as the average temperature of the annulus region (522°F) and core region water temperature (550°F). Core spray piping temperature is the same as the temperature of Region B.

Feedwater Transient Condition (TH02)

A Loss Of Feedwater Pumps (LOFP) is considered for upset conditions. In this event, the water temperature in the annulus region is dropping rapidly to 300°F while the temperature of the RPV remains at the normal operating temperature of 522°F. The average temperature of the shroud under this transient condition is 433°F. The temperature of the core spray piping is considered to be the temperature of the water in the annulus region.

Core Spray (THO3) - DBA Short Term (CS-DBA1)

This mode describes the condition shortly after core spray is initiated due to a Design Basis Accident (DBA) recirculation line break. The reactor has depressurized to 27 psig. Cold core spray water (120°F) is injecting, cooling the piping while the RPV and core shroud remain hot (522°F and 536°F, respectively). The pipe temperature is estimated as the average of the core spray water temperature and the annulus water temperature (270°F) which is based on T_{SAT} at 27 psig reactor pressure.

Core Spray (THO4) - DBA Intermediate Term (CS-DBA2)

This mode describes the condition in CS-DBA1, but at a later time when the core shroud has cooled along with the piping. Since the RPV cools much more slowly than the core shroud, it is assumed to remain at its normal operating temperature (522°F) as a bounding condition. Core shroud temperature is based on T_{SAT} at 27 psig which is 270°F. The piping temperature is the same as in CS-DBA1.

Core Spray (THO5) - DBA Long Term (CS-DBA3)

This mode describes the condition at a later time than CS-DBA2 (≈ 6 hrs after accident) when the RPV has cooled along with the core shroud and piping. The reactor pressure has decreased to 7 psig with T_{SAT} = 232°F annulus water temperature and 125°F core spray water.

Core Spray (TH06) - ADS Blowdown for Small or Intermediate Breaks

This mode describes a bounding condition for the case of a small or intermediate break in which the ADS system depressurizes the vessel to allow the core spray and LPCI systems to operate. The ADS relief valves close at a pressure of 50 psig so this pressure is used as a minimum for this event. The bounding thermal condition is judged to be the point at which the core shroud and piping temperature have cooled and the RPV remains hot. Core shroud temperature is based on T_{SAT} at 50 psig which is 298°F. RPV temperature is analyzed as 522°F. Piping temperature is based on the average of 120°F core spray water and 298°F annulus water temperature.

HPCI (TH07) - Unassisted HPCI Event, No Core Spray

This mode is for a small break event in which the HPCI system operates alone to maintain reactor water level. The minimum operating reactor pressure for HPCI is 150 psig. This pressure is used as a basis for the minimum reactor annulus water temperature, T_{SAT} = 366°F. The bounding thermal condition for this event is the point where the core shroud and piping have cooled while the RPV remains hot. The core shroud and piping are analyzed at the annulus temperature of 366°F and the RPV at 522°F.

5.1.3 Drag Load (DWH1, DWH2, DWH3)

The drag load of the reactor water on the core spray piping is evaluated in the normal operating condition (DWH1) and during a Reactor Recirculation Line Break (RRLB) condition (DWH2). The drag loads during an RRLB were found to envelope those of a Main Steam Line Break (DWH3). Drag load calculations are provided in Section 5 of Reference 5.

5.1.4 Core Spray Injection Force (INJF)

Since the thermal sleeve in the core spray RPV nozzle is a slip joint and not welded to the nozzle, the hydraulic force of the water is applied externally to the core spray piping at the 8" x 6" T-Box in the axial direction of the 8" diameter thermal sleeve. The force was based on a maximum allowed core spray flow rate of 5650 gpm as listed in the Quad Cities UFSAR (Reference 7). A force of 2441 lbs., based on 71.4 psid, was used to calculate the injection force which are provided in Section 5 of Reference 5.

5.1.5 Displacement Analyses (OBDX, OBDZ, SSDX, SSDZ SDIS)

The core spray piping is anchored to the core shroud at Node Points (NP) 307 and 327 as shown in Figure 6.1. It is attached to the RPV by supports located at node points 75, 125, 145 and 195. Displacement of the core shroud relative to the RPV results in differential support motion which is analyzed for OBE and SSE seismic events as well as for the RRLB events.

The OBE seismic core shroud displacements are 0.25" in the N-S direction and 0.435" in the E-W direction (Ref. 5 Section 1). SSE displacements are twice the OBE displacements.The seismic displacements are analyzed separately in the X and Z-directions (X = north-south axis, Z = east-west axis). The vertical Y displacements are negligible. Since the SSE seismic displacements are twice the OBE displacements, only the OBE is analyzed and the results are doubled to obtain the SSE results.

The RRLB event was determined to bound the MSLB event with respect to loads on the core spray piping. It was analyzed by calculating the cracked shroud displacement in the direction of each recirculation suction nozzle at 155° and 335°

5.1.6 Seismic Inertial Analyses (OBE1, SSE2)

OBE 1% damping and SSE 2% damping were used in the piping analyses. Two spectra, one at the RPV penetrations and one at the core shroud penetrations are enveloped for this analysis.

A uniform acceleration of .08g's (OBE) and .16g's (SSE) was used in the vertical Y-direction for all frequencies. The maximum of the X+Y or Y+Z combined seismic responses are used. The X-direction and Z direction seismic displacement results are combined separately with the inertial seismic and the two combinations are enveloped. Y-direction seismic displacements are negligible. The contributions of residual modal mass, hydrodynamic mass and the gap type supports are included in the analysis results.

Additional Analysis Notes

For the 6" diameter piping the normal drag force is in the same direction as the weight. Since the buoyancy force acts in the opposite direction of the drag force and weight it is conservatively not included.

For Thermal Mode 1 and RRLB Drag load, the piping moves toward the gap at the bracket supports at NP's 125 and 145. The free displacements at the supports are input at NP's 125 and 145 in the final Thermal Mode 1 and RRLB drag analyses so that the supports do not take a load in the gap direction.

5.2 Load Combinations For Limit Load Flaw Evaluations

Normal & Upset (P = 0)
DWGT + DWH1
DWGT + DWH1 + OBXY
DWGT + DWH1 + OBYZ

Emergency and Faulted (P = 0)
DWGT + DWH1 + SSXY
DWGT + DWH1 + SSYZ
DWGT + DWH2
DWGT + DWH3

Beyond Design Basis Condition
DWGT + DWH2 + SSXY
DWGT + DWH2 + SSYZ
DWGT + DWH3 + SSXY
DWGT + DWH3 + SSYZ

Load Combinations For Leakage Evaluations (P= 47 and 64 PSID)
DWGT + TH04 + P2
DWGT + TH04 + P3

6.0 CORE SPRAY PIPING MODELLING AND ANALYSIS

6.1 Piping Models

The purpose of the piping analysis is to provide forces and moments on the 6" diameter core spray piping in the reactor annulus to be used for flaw evaluations, piping stress analysis qualification and for repair hardware design. The subject piping was analyzed using the PIPSYS program for the load conditions described in Section 5.0. The affected portion of the piping representing the upper and lower core spray spargers was analyzed utilizing two separate models delineated as the "upper sparger" and "lower sparger". A summary of core spray piping stress analysis is provided in Section 17 of Reference 5.

The piping model is based on the design basis drawings (References 8 through 13) and is shown in Figure 6.1. It consists of core spray piping inside the reactor. From the 8" RPV nozzle and 8"X6" Tee-Box, the 6" piping follows the circumference of the reactor above the core shroud to two vertical legs which drop down and penetrate the core shroud horizontally after a 90° elbow. The model ends at these anchored shroud penetrations, NP 307 and NP 327. The piping is supported directly to the RPV at NP's 75, 125, 145, and 195.

This core spray piping exists in mirror image on both sides of the reactor, with the only difference being that one drops to a lower elevation on the core shroud to connect with the lower core spray sparger. The piping model for the lower sparger was modified by shortening the vertical legs to create the upper sparger model. Since the two piping systems are 180° apart, the coordinate system used in the models points in opposite spatial directions for the two models. Isometric drawings in Figure 6-1 show the appropriate coordinate system. The piping is 6" diameter Sch. 40, TP-304 stainless steel with a short leg of 8" diameter Sch. 40, TP-304 piping at the reactor nozzle. Flexible anchors are modelled at the core shroud penetrations with the model terminating at the 6" diameter 90° elbow outlet. Stiffnesses for the penetration assembly were calculated based on a finite element analysis as documented in Reference 14. The 8"x6" tee-box is modelled as 8" diameter Sch. 40 piping.

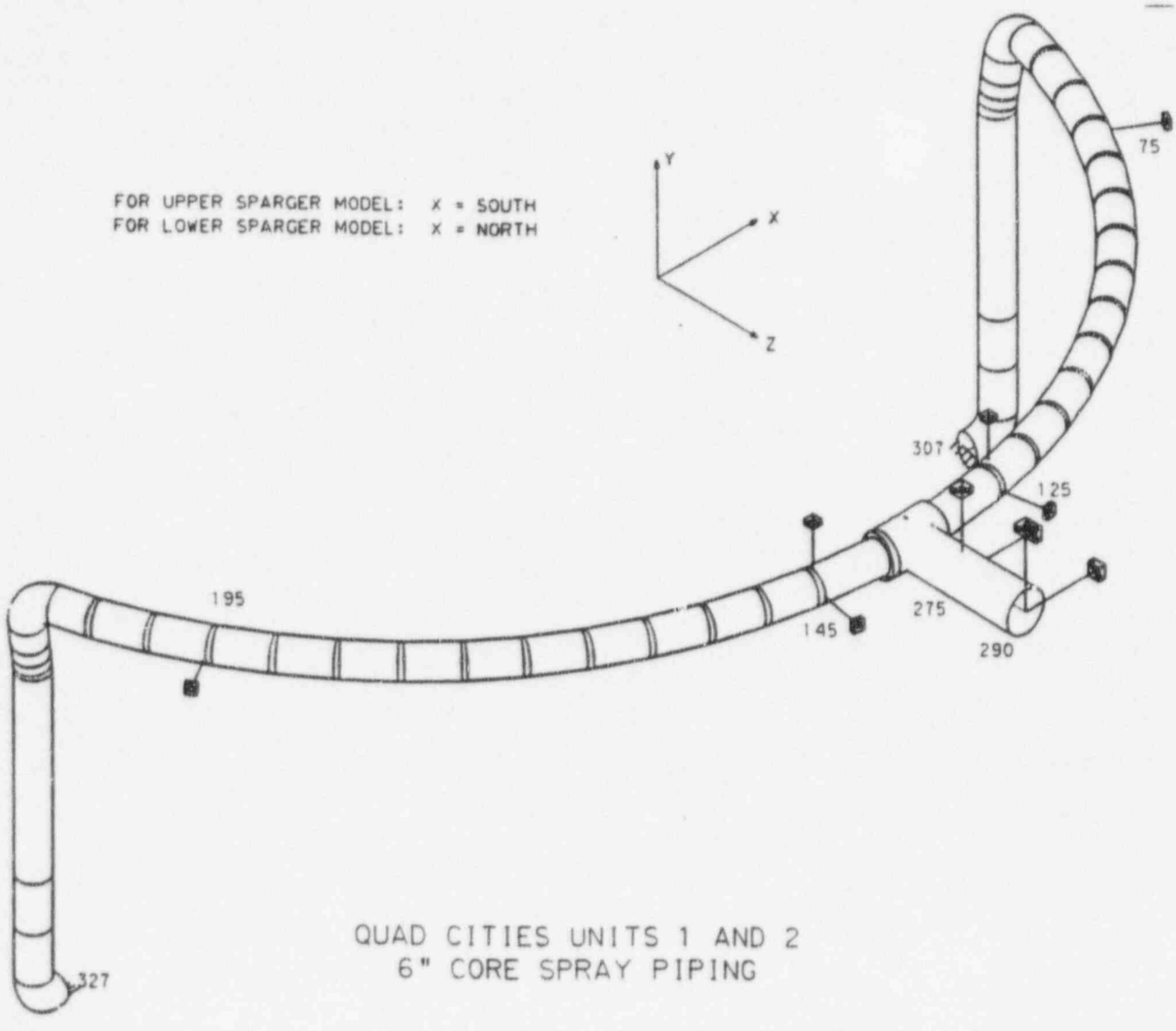


Figure 6.1 Core Spray Piping Analysis Model

6.2 PIPSYS Analyses Performed and Microfiche Index

Microfiche Run ID	Analysis Date	Description
UPPER Q1	11-17-95	Upper Sparger Model-Cracked Shroud
LOWER Q2	11-20-95	Lower Sparger Model-Cracked Shroud Rigid Model

7.0 FLAW EVALUATIONS

This section describes the methodology and details of the core spray line flaw assessments. The loading and stress analysis results as defined in the preceding sections serve as the primary inputs for these flaws evaluations. The flaw evaluations were performed using the ASME Section XI, Appendix C, limit load method for the flaws characterized in Section 3, and the material evaluation results presented in Section 4. The evaluation also includes an assessment of key analysis parameters and provides additional results based on the limits of these parameters.

7.1 Flaw Evaluation Methods

These flaws are evaluated using the limit load methodology of ASME B&PV Code Section XI, Appendix C, (Reference 15). This methodology assumes a plastic collapse failure mode of the flawed cross-section. Plastic collapse failure occurs when the remaining uncracked ligament is assumed to reach a plastic flow stress level and behaves as a hinge at failure (Reference 16). This failure mechanism is appropriate based on the inherent fracture toughness and ductility of type 304 austenitic stainless steels. As defined in ASME Section XI, Appendix C, the flow stress is defined as $3S_m$ at operating temperature. For these evaluations, the operating temperature is 550°F and the corresponding S_m is 16950.0 psi (Reference 17).

As previously stated the elbow flaws are located in the HAZ of a 100% GTAW weld (see Figures 3.1 and 3.2). Consequently, these flaws, which are located in the HAZ of a non-flux weld and base metal were evaluated using the base metal and GTAW evaluation formulas.

7.1.1 Flaw Characterization

As previously described in Section 3, the flaws being evaluated were found through a visual examination and corroborated by additional UT examinations. The UT examinations confirmed the existence and location of flaws that are believed to be connected to the inside surface of the elbows. Although the UT examination was capable of determining the existence of internal and external surface flaws, it was not able to determine whether or not they were connected through wall. For conservatism, these evaluations assume the flaws to be through wall. As previously defined in Section 3, the evaluation period has been defined as a 24 month hot operating period. The crack growth during this period is based on the conservative IGSCC rate of 5×10^{-5} in/hr as defined in

Section 4. The thermal transient and expansion loads associated with the start-up/shutdown and normal operation of the vessel are insignificant. During normal operation, the internal and external line pressure is equal. This eliminates any fatigue concerns associated with pipe line pressure fluctuations. Based on the low flow velocities and the horizontal rigidity (high fundamental frequency) of the core spray lines, flow induced vibrations will be negligible. Consequently, fatigue crack growth will not contribute significantly to crack extension and is not considered in the projected flaw length.

7.1.2 Flaw Evaluation Stress Inputs

The loads used in these evaluations were obtained from the piping model of the core spray lines documented in Reference 5. These models generated the axial forces and bending moments acting on these flaws for the following loads:

- Weight
- Thermal
- Seismic
- Operating Drag
- LOCA

The design basis load combinations were evaluated and the worst case normal/upset and emergency/faulted condition load combinations were used for these evaluations. Additional beyond design basis, faulted load combinations were also evaluated to assess the design margin for these extreme cases. The simultaneous occurrence of a seismic SSE event with the RRLB LOCA was postulated as the bounding beyond design basis load combination. Reference 14 has determined that the RRLB LOCA event produces loads which bound the MSLB LOCA loads for this piping. The loads used for the elbow flaw evaluation are taken directly from the piping analysis results reported in Section 6.

Table 7.1 presents the membrane and bending stress values for the bounding design basis load combinations. In addition to the design basis load combinations, the additional faulted load combination of SSE and RRLB LOCA was examined to calculate, "beyond design basis" margins.

Table 7.1 Flaw Evaluation Stress Values (psi)

Flow Location	Design Basis (1)		Beyond Design Basis	
	σ_m	σ_b	σ_m	σ_b
Upper Sparger 290° Elbow	417	639	417	1,008
Lower Sparger 260° Elbow	420	678	420	1,107

(1) Includes the bounding load combination for normal/upset as well as emergency/faulted conditions.

7.1.3 Flaw Limit Load Evaluations and Results

The allowable bending stress, P_B, for the limit load evaluation was calculated using equation 7-1.

$$P_B = 6\ \frac{S_m}{\pi}\ \{2\ \sin(\beta)\ -\ \frac{a}{t_n}\ \sin(\theta)\} \qquad (Eq.7\text{-}1)$$

$$with \qquad \beta = \frac{1}{2}\ \{\pi\ -\ \frac{a}{t_n}\ \theta\ -\ \pi\ \frac{P_m}{3S_m}\}$$

$$and \qquad \theta + \beta \leq \pi$$

Where θ is defined as the half angle as presented in Figure 7.1, and P_M is the membrane stress acting on the flaw. Because the flaws are assumed to be through-wall, the a/t_n ratio is equal to 1.

For these evaluations, the applied bending stress, P_{AB}, must be less than the allowable bending stress. The applied bending stress is calculated using equation 7-2.

$$P_{AB} = SF\ (P_m + P_b) - P_m \qquad (Eq.\ 7\text{-}2)$$

The code safety factor (SF) is 2.77 for normal/upset and 1.39 for emergency/faulted conditions. P_M and P_B are the applied membrane and bending stress, respectively.

The flaw evaluations were performed to determine the load margin for the end of evaluation period flaw size reported in Section 3. The load margin is defined as the ratio of the maximum permitted stress P_B, to the applied stress P_{AB}. This ratio represents the margin with respect to the applied load above the ASME Section XI safety factors. In addition to the load margins, the remaining months of operation were determined by calculating maximum flaw lengths which would meet the code required safety factors. The months of operation required to reach the critical flaw length were calculated using the bounding crack growth rate of 5×10^{-5} inches/hour. The results of these calculations are presented in Table 7.2.

Table 7.2 Flaw Evaluation Results

Flaw Location	Load Margin Factor at end of Evaluation Period[1]		Months of Operation to Reach Critical Flaw Length	
	Design Basis	Beyond Design Basis	Design Basis	Beyond Design Basis
Upper Sparger 290° Elbow	17	11	120	114
Lower Sparger 260° Elbow	22	14	139	131

(1) This is the margin on load above and beyond the ASME Code Safety Factors of 2.77 for Normal/Upset conditions and 1.39 for Emergency/Faulted Conditions

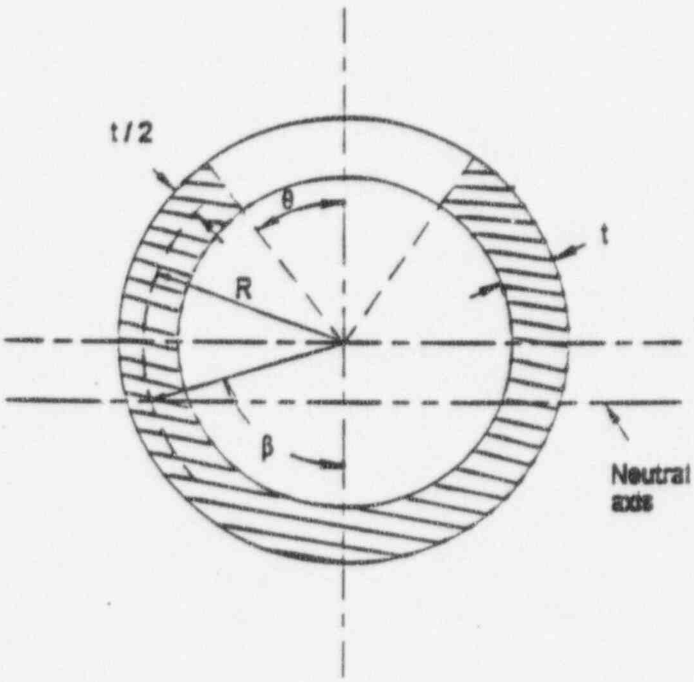


Figure 7.1 Cross Section of Flawed Pipe

7.2 Sensitivity Analysis

The most significant parameter influencing these flaw evaluations is the load acting on the flawed section. As previously discussed, the limit load method employed for this evaluation assumes a plastic collapse failure mechanism. Secondary or displacement controlled loads are relieved as the remaining ligament deforms plastically, thus the flaw evaluation is performed using only primary loads. The assumed plastic collapse failure mechanism is dependent on the material ductility and toughness, which is appropriate for type 304 austenitic stainless steels and non-flux welds. However, materials with reduced ductility and toughness such as flux welds, may exhibit ductile tearing with net section yielding, (i.e. an elastic-plastic failure mechanism). This sensitivity analysis examines the impact of secondary loads and ductile tearing on the flaw structural integrity and remaining life estimates. The elbow flaw is located in the HAZ of a non-flux weld therefore, in accordance with the test results reported in References 18 and 19, and as specified in Section XI, Appendix C of the ASME code the greater material toughness and ductility does not warrant an examination of the elastic-plastic failure mechanism. However, this sensitivity analysis examines the impact of the secondary loads on the elbow flaw structural integrity and remaining life estimates. The following evaluations determine the load margin for the end of evaluation period flaw size, from Section 3, and the remaining months of operation for the primary plus secondary loads.

The loads used in these sensitivity evaluations are defined in the same manner as described in section 7.1. Table 7.3 presents the membrane and bending stress values for the bounding design basis load combination as well as the "beyond design basis" load combination.

These evaluations were performed using the simplified elastic-plastic approach defined in Section XI, Appendix C of the ASME B&PV Code. This approach requires that secondary stresses be included in a limit load formulation which uses a reduction factor, Z_1, to conservatively approximate an elastic-plastic failure mechanism. The allowable bending stress, P_B, for these evaluations was calculated using equation 7-1. However, the applied bending stress is modified, as stated above, to include the Z_1 factor and the secondary stresses as presented in Equation 7-3.

$$P_{AB} = Z_1\ SF(P_m + P_b + P_e/SF) - P_M \qquad \text{(Eq. 7-3)}$$

Where Pe is the applied secondary load bending stress, and Z_1 is unity for GTAW weld and base metal.

The results of these sensitivity evaluations are presented in Table 7.4. It contains the load margins and remaining months of operation as defined in section 7.1.3. These results demonstrate that, for the limiting loads and material conditions, the structural integrity of the flaws is assured.

7.3 Flaw Evaluation Conclusions

Based on the results presented in Table 7.2, the minimum design basis load margin for the end of evaluation period flaw size is 17 and would require 120 months of operation to reach a critical flaw size. For the additional load combination of RRLB LOCA plus an SSE, which is beyond the design basis of the Quad Cities Station, the minimum load margin is 11 and would require 114 months of operation to reach a critical flaw size. These results demonstrate that the flaws, projected to grow at a conservative IGSCC crack growth rate of 5x 10^{-5} inches/hour for 17,280 hours, will remain structurally stable when subjected to design basis accident conditions. These results also demonstrate that reactor operation for more than 114 months can occur before the flaws are predicted to reach a critical length.

Table 7.3 Flaw Evaluation Sensitivity Analysis Stress Values (psi)

Flow Location	Load Type	Design Basis (1)		Beyond/Design Basis	
		σ_m	σ_b	σ_m	σ_b
Upper Sparger 290° Elbow	Primary	417	639	417	1008
	Secondary	139	5,029	160	5,836
Lower Sparger 260° Elbow	Primary	420	678	420	1,107
	Secondary	111	4,465	127	5,169

(1) Includes the bounding load combination for normal/upset as well as emergency/faulted conditions.

Table 7.4 Flaw Evaluation Sensitivity Analysis Results

Flaw Location	Load Margin Factor at end of Evaluation Period (1)		Months of Operation to Reach Critical Flaw Length	
	Design Basis	Beyond Design Basis	Design Basis	Beyond Design Basis
Upper Sparger 290° Elbow	2.85	2.35	78	71
Lower Sparger 260° Elbow	4.25	3.45	101	93

(1) This is the margin on load above and beyond the ASME Code Safety Factors of 2.77 for Normal/Upset conditions and 1.39 for Emergency/Faulted Conditions

8.0 LEAKAGE FLOW EVALUATIONS

This leakage flow evaluation determines the rate that water is lost from the elbow flaw in the upper sparger, during core spray injection. Upper sparger elbow flaw is longer than the flaw at the lower sparger elbow and this represents the bounding condition. The core spray system leakage is calculated for elbow flaw lengths at the end of the evaluation period, as reported in Section 3, and at the end of life.

8.1 Leakage Calculation Methodology

The elbow flaw leak rate is calculated using the PICEP program developed by EPRI for Leak-Before-Break applications, (Reference 20). This program uses elastic-plastic fracture mechanics to calculate the crack opening area of a through wall circumferential flaw. It calculates the leak rate based on "Henry's Homogeneous Nonequilibrium Critical Flow Model" (Reference 21). This evaluation is based on the combined membrane and bending stresses acting on the flaw from the combined loads which occur during the injection mode. The Ramberg-Osgood stress-strain parameters were obtained from Reference 22, the IPIRG Task 1.3 piping system tests database developed by Batelle, and are an average of type 304 base metal tests at 550°F and 70°F. Because the piping temperature cools very quickly during the LOCA event and after the initiation of the core spray flow at 120°F, the line temperature is reduced to an average temperature of 195°F for this leakage calculation. Interpolation of the stress-strain data for 550°F and 70°F to 195°F was used to establish the stress-strain input to the leakage calculations.

8.2 Leakage Calculation Applied Loads

During the core spray injection mode, the elbow flaw is subjected to the combined flow induced loads and differential thermal expansion loads. At approximately 60 seconds after a DBA LOCA, the core spray maximum differential pressure is 47 psid at a design basis rated flow of 4600 gpm (Reference 8). As the reactor vessel pressure continues to reduce to 0 psig, the maximum differential line pressure would reach 64 psid at the core spray pump flow rate of 5350 gpm and 71.4 psi at the pump runout flow rate of 5650 gpm. The leakage flow rate was calculated for both the 47 psid and 64 psid line pressure conditions. The maximum differential pressure of 71.4 psid is not a governing condition for leakage evaluations, but was used to determine bounding loads for the flaw evaluations. The thermal load acting during the injection mode is conservatively based on the core shroud and reactor vessel being hot while the core spray piping is cold, as described in Section 5.

8.3 Calculated Leakage

The leakage was calculated based on the previously described loads and material properties and is presented in Figures 8.1 and 8.2 for an injection pressure of 47psid and 64 psid respectively. From Figure 8.1, the leak rate for the end of evaluation flaw size is 5 gpm and at the end of life is 62 gpm. From Figure 8.2, the leak rate for the end of evaluation flaw size is 6 gpm and at the end of life is 92 gpm. The end of life flow rates calculated here are based the conservative thermal stresses generated from a rigid model neglecting the effects of the flexibility introduced by the flaw. The end of life flaw length will introduce significant flexibility in the system which would result in reduced bending stresses. The results of this leakage evaluation are compared to the system capacity in Section 9.0 of this report.

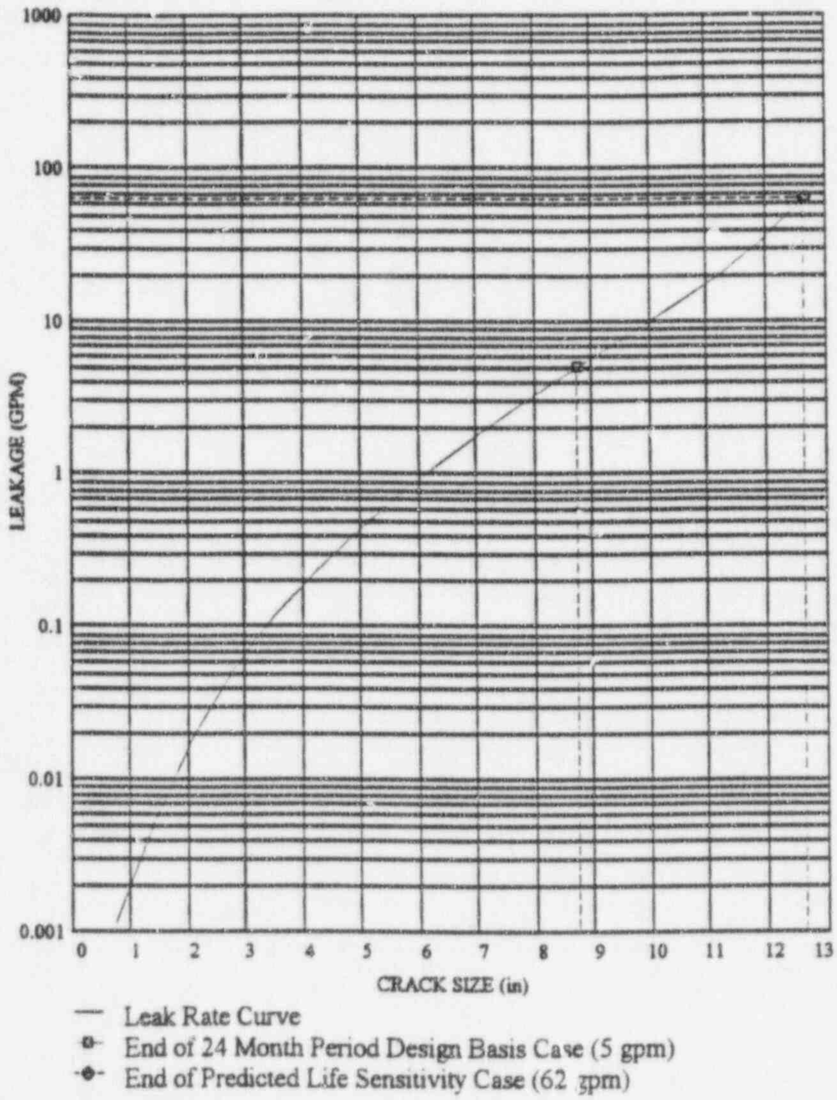


Figure 8.1 Leakage Rate at a Differential Pressure of 47psi

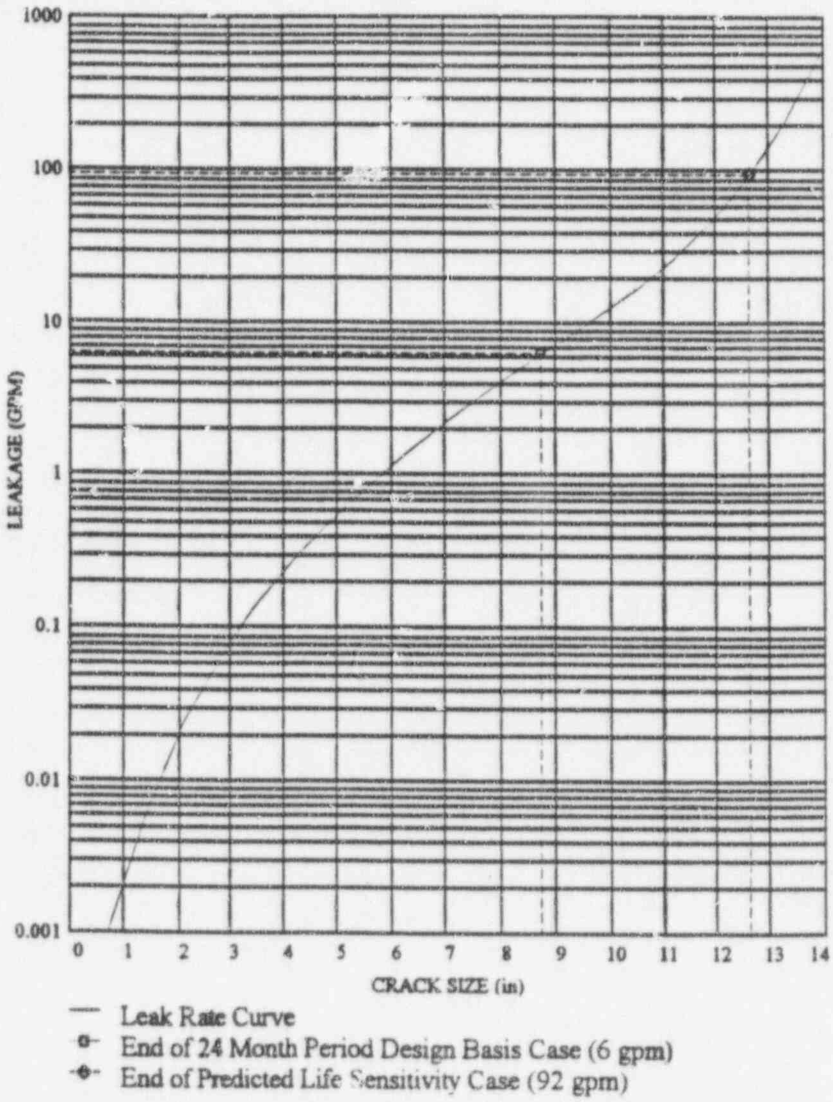


Figure 8.2 Leakage Rate at a Differential Pressure of 64psi

9.0 CORE SPRAY SYSTEM LOCA EVALUATION

9.1 Core Spray System Description

The core spray system along with High Pressure Coolant Injection (HPCI), Residual Heat Removal System (RHR) (Low Pressure Coolant Injection - LPCI) mode and Automatic Depressurization System (ADS) make up the ECCS for Quad Unit 1. The core spray system consists of two independent redundant loops each consisting of a pump, valves, piping and independent circular sparger ring inside the core shroud just above the core. The normal water source for pump suction is the suppression pool. Each core spray pump takes suction from a common ring header that has four suction lines. A fill system is used to ensure that the core spray discharge lines remain pressurized. This fill system consists of a pump which takes suction from the suppression pool via a core spray suction line and discharges to the core spray pump discharge lines. The power source for each core spray loop is located on an independent emergency bus. Each core spray loop is designed so that each component of the subsystem can be tested periodically.

9.2 Core Spray System Safety Function

Each core spray loop is designed to operate in conjunction with the RHR subsystem and either the ADS or HPCI subsystems to provide adequate core cooling over the entire spectrum of liquid or steam pipe break sizes. For the small line break accident, the ADS or HPCI subsystems are used to depressurize the vessel to a point where the core spray and RHR systems can be initiated in time to ensure adequate core cooling. For the large break LOCA, the depressurization assistance from HPCI or ADS is not required. For the full range of LOCA break sizes, the current design basis requires that core cooling be provided by both core spray loops operating together with ADS and HPCI, or by one core spray loop operating with two RHR pumps (one RHR subsystem) and ADS. The core spray loops can be powered from either offsite or onsite sources.

9.3 Leakage Flow Evaluation

The bounding case for core spray with respect to PCT is the DBA-LOCA consisting of a reactor recirculation suction line break in combination with a single failure of a battery (Table 6.3-12 Reference 12). This requires core spray to cool and reflood the core with assistance from the LPCI mode of RHR.

The critical DBA-LOCA leakage is based on a pressure of 47 psid which corresponds to the maximum core spray flow of 4600 gpm and is 5 gpm through the elbow flaw. This is based on a flaw length developed after 24 months of operation with crack opening based on the design basis load combinations. A bounding leakage of 62 gpm was determined based on the end of life flaw size and using the design basis load combinations.

During the blowdown phase of the DBA-LOCA, any core spray flow due to leakage in the annulus piping will be lost through the break. This volume of water loss can be directly subtracted from the core spray flow assumed in the current DBA-LOCA calculations. This would cause a decrease in liquid flow to the lower plenum during the reflood phase of the DBA-LOCA and a subsequent increase in the time required to quench the "hot node".

Core Spray T-Box weld flaw repair, core spray line weld flaws in the T-Box, core spray slip fit thermal sleeve-to nozzle safe end, vent hole in core spray line T-Box, and purge hole in nozzle thermal sleeve were determined to have a total leakage of less than 277 GPM in the design basis LOCA scenario (Reference 23). The additional leakage of 5 gpm due to the elbow flaw is well within the 400 gpm analyzed reduction in the core spray flaw rate. Core spray was conservatively analyzed with a total reduction in flow rate of 400 gpm so that 4,100 gpm of core spray was delivered to the top of the core. Each of these leakages were incorporated into the current design basis LOCA analysis (Reference 23).

Flow distribution in the upper plenum, as well as leakage not available to the fuel rods, were taken into account in establishing the original design basis flow requirements. However, with the introduction of SAFER/GESTR LOCA methodology by GE, credit for the flow distribution was conservatively not assumed. Only the reflood effect was credited and required for adequate core cooling in the current design basis LOCA analysis. All of the documented existing leakage locations in the core spray system were considered and an evaluation was performed for 4100 GPM of core spray delivered to the top of the core and the impact has been assessed to increase the fuel's peak clad temperature (PCT). The PCT increase for this 400 gpm reduction is 40°F, and when added to the existing PCT of 1725°F, it is less than the regulatory limit per 10CFR50.46.

Based on this evaluation, the postulated leakage is not significant since the PCT would remain below 2200° F for the GE 8x8 fuel which is presently installed. Leakage resulting from the elbow flaw would only have an impact on the PCT for the postulated bounding case of a recirculation line break with concurrent with a battery failure.

10.0 BOUNDING FAILURE ASSESSMENT

Based on the results of the flaw evaluation in conjunction with the visual and UT inspections, the potential of developing a 360° circumferential failure in the downcomer elbow containing the flaw is not credible. This bounding beyond-design-basis failure assessment was performed as a means of assessing design margin. This assessment utilizes both a deterministic and probabilistic approach. The bounding failure postulates a 360° circumferential failure in any one of the four core spray downcomers that feed the spargers located inside the shroud. There are two such downcomers per core spray subsystem. Section 10.1 discusses the details of the deterministic assessment and Section 10.2 discusses the probabilistic assessment.

10.1 Deterministic Assessment

The deterministic investigation consists of an evaluation of three scenarios, each concurrent with the postulated 360° failure of any one of the four core spray downcomers. The three scenarios evaluated are:

- The DBA-LOCA of the instantaneous failure of a coolant or reactor recirculation pipe,
- Safe Shutdown Earthquake (SSE),
- The DBA-LOCA with the single failure (most probable single failure being the failure of the LPCI injection path).

The evaluation consists of postulating each scenario and demonstrating that, for each scenario adequate core cooling is provided.

10.1.1 Postulated Failure with DBA-LOCA

The DBA-LOCA is the instantaneous mechanical failure of a pipe equal in size to the largest coolant/recirculation system pipe. The bounding DBA-LOCA for demand on the core spray system is a reactor recirculation suction line break. Adequate core cooling can be provided even if one core spray loop is disabled due to failure of a core spray downcomer elbow in conjunction with the DBA-LOCA, since one core spray loop and one LPCI loop will remain available and can provide the required core cooling.

10.1.2 Postulated Failure with a SSE Event

The SSE is the earthquake which produces the maximum vibratory ground motion for which certain structures systems and components are designed to remain functional. The reactor vessel pressure boundary would be maintained during and after a SSE event. Should core spray be required, it would only be required to re-flood the vessel and not spray on top of the core. Thus, the postulated failure of the core spray downcomer elbow would only affect flow to the top of the core and the reactor coolant pressure boundary provides for capability of both core spray loops to reflood the core to assure adequate core cooling.

10.1.3 Postulated Failure with DBA-LOCA and Single Failure

This scenario combines the same DBA-LOCA discussed in Section 10.1.1 with a single failure. The single failure was postulated to be failure of the LPCI injection path. This single failure was postulated because it is the most probable single failure. The original design basis for Quad Cities for a DBA-LOCA is that one core spray loop was sufficient to cool the core. Due to changes in 10CFR50.46 and Appendix K of 10CFR50 in the mid 1970's, the current design basis requires at least one core spray loop and two LPCI pumps or two core spray loops to be operational to cool the core following a DBA-LOCA.

General Electric (GE) issued a Licensing Topical Report (Reference 25) in December 1988. This report was developed to identify and evaluate changes to Technical Specifications associated with Emergency Core Cooling systems (ECCS). This report states that any one low pressure ECCS pump or loop with at least 4600 gpm capacity and the operation of at least two Safety Relief Valves (SRVs) is sufficient to provide adequate core cooling for a BWR 3/4 plant so that the parameters and/or criteria of 10CFR50.46 are met. The results of this report are based on GE's realistic (SAFE and CHASTE computer code) LOCA model which was previously reviewed and approved by the NRC for technical specification methodology.

This results of this Licensing Topical Report (Reference 25) apply to Quad Cities Unit 1. Quad Cities Unit 1 is a BWR 3 design, with a tested flow rate for one core spray loop of 4700 gpm. Core spray pump flow is periodically tested at a flow rate of 4700 gpm to ensure that the minimum rated flow of 4500 gpm is available should the need arise.

There are other parameters and conditions in the GE evaluation (Reference 25) that are different from those existing at Quad Cities Unit 1. However, as discussed below the conclusions of this report are applicable to Quad Cities Unit 1.

- The rated core spray flow of 4500 gpm is based on a vessel pressure of 90 psig. However, as the vessel continues to depressurize following the DBA-LOCA, the core spray flow will continue to increase until the equilibrium is reached between the vessel and drywell or until system maximum flow is reached (See Section 9 for the effect of the increased flow).

- The current DBA-LOCA evaluation uses conservative estimates for other "known" leakages (i.e. through the plenum access holes, core shroud, bottom head drain line, T-Box repair, etc).

- This evaluation assumes that there will be no flow to the spargers through the failed core spray loop. Since only one of the two downcomers will contain the postulated 360° circumferential failure, some flow will be delivered through the intact downcomer, as well as the downcomer with the postulated break.

- The above mentioned repair is based on GE's 8x8 fuel. For the next operating cycle, Quad Cities Unit 1 will utilize GE8, GE9 and GE10 fuel. The GE9 and GE10 fuel have improved refueling characteristics with a "flatter" peak temperature per pin and lower stored energy than 8x8 fuel.

Thus, based on the GE Licensing Topical Report and the discussion above, for the postulated beyond design basis scenario with failure of one core spray loop due to the postulated break in the core spray downcomer elbow, core cooling could still be provided by the intact core spray loop.

10.2 Probabilistic Safety Assessment

A probabilistic evaluation was made for two scenarios (Reference 24). The first scenario is a reactor recirculation suction line break followed by failure of LPCI injection. The second scenario is a SSE occurring concurrently with the events in the first scenario. The probability of structural failure of a degraded core spray line was conservatively neglected. This approach is conservative because if such a structural failure had been included in the events postulated for the scenarios, then the scenario frequencies calculated below would have been multiplied by a structural failure probability estimate, thus giving a lower event frequency.

10.2.1 Frequency Estimate for Scenario 1

The first scenario postulates a reactor recirculation suction line break followed by failure of LPCI injection. This scenario was chosen because it is within the unit's design basis, and represents the most critical case with respect to peak cladding temperature calculations. For this scenario:

Frequency of Event = Line Break Frequency x LPCI Failure Probability

The frequency of a reactor recirculation suction line break was previously estimated as 5.6×10^{-6}/year. In the Quad Cities PRA model for a large LOCA, LPCI failure is dominated by failure of LPCI injection. The model for LPCI injection includes the loop injection valves, loop injection check valves, loop selection logic and other supporting equipment. For a large LOCA (including a reactor recirculation suction line break), the Quad Cities Individual Plant Evaluation gives a LPCI injection failure probability of 2.89×10^{-3}. This value is used for the LPCI failure probability. Thus, the frequency of the postulated scenario is:

Frequency of Scenario 1 = 5.6×10^{-6}/yr x 2.89×10^{-3} = 1.6×10^{-8}/yr.

10.2.2 Frequency Estimate for Scenario 2

The second scenario postulates a SSE concurrent with the reactor recirculation line break and failure of LPCI injection. This postulated scenario is outside the original plant design basis.

Following the approach previously used for other reactor internal evaluations, a concurrent (but independent) SSE is postulated to occur within 24 hours of the event in Scenario 1. Thus, the frequency of this scenario is:

Frequency of Scenario 2 = 1 day x (SSE Frequency)/365 days/yr x (Frequency of Scenario 1)

The frequency of a seismic event exceeding the SSE is 2.2×10^{-5}/yr (Reference 24). Using this value and the frequency estimate for Scenario 1 gives:

Frequency of Scenario 2 = (2.2×10^{-5}/yr)/365/yr x 1.6×10^{-8}/yr = 1×10^{-15}/yr.

Note that this frequency estimate is for concurrent but independent events. This estimate should not be interpreted as applying to a seismic-induced LOCA.

10.2.3 Conclusions of Probabilistic Safety Assessment

Based on the low values of the calculated frequencies for the two scenarios, it can be concluded that the likelihood of the occurrence of either scenario is very small, and neither scenario is risk significant.

11.0 LOOSE PARTS EVALUATION

As part of the evaluation of the cracked core spray sparger, a scenario has been postulated where an elbow of the lower sparger inlet piping breaks off. This section of piping is assumed to fall into the vessel annulus region. An evaluation has been performed to address the safety concerns raised as a result of this loose piece.

11.1 Postulated Loose Part

The postulate loose part is a curved, stainless steel elbow. Based on the location of the observed cracks in core spray loop "B" (the elbow to the thermal sleeve weld at the elbow penetration into the core shroud) the entire elbow is the most likely part to break loose. There may also be debris created as a result of rubbing and scraping of the elbow on vessel internal components.

11.2 Safety and Operational Concerns

The safety and operational concerns associated with this postulated loose part are:

- Potential for fuel bundle flow blockage and consequent fuel damage,
- Potential for fretting wear of the fuel cladding,
- Potential for interference with control rod operation,
- Potential for corrosion or chemical reaction with other reactor materials.

The elbow is postulated to break away from the core spray piping and fall into the downcomer region. This is reasonable since it is part of the piping in the annulus region outside the shroud.

11.2.1 Potential for the Fuel Bundle Flow Blockage and Consequent Fuel Damage

The elbow is located in the annulus region. Because of its size it will be unable to leave the annulus region. The jet pump throat is too small to pass the elbow and the jet pump nozzle is far too small to pass the part into the lower plenum. Therefore, the elbow itself cannot create a fuel bundle flow blockage. Debris created by the falling part is small enough to enter the lower plenum. Once in the lower plenum, the flow velocities are sufficiently large that the debris will be carried toward the fuel support inlet orifice. Because of its size the debris will not restrict the flow through the fuel support inlet orifice.

Depending upon the size of the debris, it may or may not pass through the lower tie plate openings. Even if it becomes trapped in the lower tie plate, the flow blockage would be quite small and distributed throughout the fuel assemblies. Therefore, no boiling transition would occur.

There is no concern for fuel bundle flow blockage due to the postulated lost.

11.2.2 Potential for Fretting Wear of Fuel Cladding

If debris is created by the elbow rubbing on vessel internal parts, it could be small enough to be carried upward past the lower tie plate openings. It may become trapped at a fuel bundle spacer. This may cause the debris to rub over a small surface of a fuel rod. Prolonged operation may lead to fretting wear and leaks in the fuel rod. Any fuel cladding leaks would be detected by the off-gas system so that appropriate action can be taken to maintain the offsite radiation release within acceptable limits. Any such cladding damage would be an operational or economic concern, not a safety concern.

11.2.3 Potential for Interference with Control Rod Operation

If debris is carried past the lower tie plate it would have to travel through the fuel bundle spacers, exit the fuel channel through the upper tie plate, reverse direction, and travel downward so that it could enter the control rod guide tube. This is an extremely unlikely trajectory. Once in the control rod guide tube, the debris would have to pass through the clearance between the velocity limiter and the guide tube wall and continue to fall. Once past the velocity limiter, it is very likely that piece would drop to the outer edge of the guide tube bottom. Once resting there, the debris is not likely to be lifted because there is no upward flow velocity in the outer edge of the guide tube bottom. Even if debris were lifted from the bottom, it would have to rise above the ridge surrounding the annulus between the index tube and the guide tube bottom, move over the annulus opening, orient itself in such a way as to enable travel through the very small gap and then fall into the control rod drive (CRD) mechanism. This would all occur against CRD cooling flow. This is considered highly unlikely. Even if this should happen, the debris would not have sufficient mechanical strength to impair either the safety function (scram) or normal control rod drive operation. Consequently, there is no concern for potential interference with the CRD operation due to the postulated lost part.

11.2.4 Potential for Corrosion or Chemical Reaction with Other Reactor Materials

Since the postulated loose part is made of stainless steel, a material approved for in reactor use, there is no concern for corrosion or chemical reaction with other reactor materials.

11.3 Loose Parts Monitoring

Quad Cities does not have a loose-parts monitoring system. All reactor internals components and repair hardware are designed to have all pieces locked in place with mechanical devices. Hence loose parts are not anticipated. Visual inspection to identify any loose or degraded components is performed at regularly scheduled intervals.

In the remote possibility that a part of the core spray system does become loose, it would fall and rest on the jet pump support plate. The possibility of a loose part reaching the reactor fuel is even more remote. If fretting of the fuel clad did occur due to a small loose part/piece (i.e., 1/2 inch in diameter or less), the Off Gas Radiation Monitors would detect the increase in fission product release (radiation). The Quad Cities Technical Specifications delineate the instrumentation requirements for these monitors. Station operating procedures provide required actions when these monitors indicate elevated release rates, in order to minimize the release of fission products.

The Main Steam Line Radiation Monitors are designed to detect large changes in fission product release (gross fuel failure), and provide automatic protective functions to minimize the release of fission products.

This protective function will actuate when a predetermined and preset radiation level in the main steam is reached. The Quad Cities Technical Specifications delineate the instrumentation requirements and setpoint for these monitors. When the setpoint is reached, an automatic action is initiated to close the main steam line isolation valves and SCRAM the reactor.

11.4 Conclusion of the Loose Parts Evaluation

The safety evaluation conducted for the postulated core spray sparger elbow and debris has concluded that there is no potential for significant fuel bundle flow blockage, no safety concern due to cladding wear, no potential for interference with control rod operation and no potential for corrosion or adverse chemical reaction with other reactor materials. Thus, there are no safety concerns raised by the postulated break of the elbow of the core spray lower sparger inlet piping and fuel cooling throughout the core as well as control rod operation can be maintained.

12.0 SUMMARY AND CONCLUSIONS

Crack indications were identified at two locations on the core spray downcomers during the Q1R14 in vessel visual inspections. This core spray line inspection was planned and implemented as part of the in-vessel visual inspection of the reactor internals. The approach used to define and evaluate the flaws in the Quad Cities Unit 1 core spray downcomers was complete and thorough and addressed all relevant parameters. The approach was to fully utilize all of the latest industry and plant specific information to plan and execute the inspections as well as the engineering evaluations. This is reflected in the thorough and detailed visual inspections that were performed along with the use of ultrasonic testing to corroborate and clarify the inspection results. The stress analysis and flaw evaluations were performed using verified design inputs for all key analysis parameters. Where the analysis parameters were determined to have a significant impact on the analysis or evaluation, a conservative bounding value was selected and a sensitivity study was performed. Provided below is a summary description of the analyses and evaluations performed along with the conclusions reached.

The details of the visual and ultrasonic examination results are defined in Section 3 of this report. The critical flaw identified was a 7 inch long crack in the A-loop inlet elbow at 290°. This crack was conservatively assumed to be through wall and was extended using a bounding IGSCC crack growth rate of 5.0 X 10^{-5} for a 24 month operating cycle to an evaluated flaw length of 8.72". The UT methodology developed and utilized as part of the flaw characterization was prequalified and independently verified by industry experts. The approach and methods used represent the best available in the industry and provide an accurate basis for performing a flaw evaluation.

The materials evaluation included a detailed assessment of the inspection records, the fabrication details, the key material behavior characteristics as well as a review of relevant industry information. The review of the inspection results and pertinent industry experience indicates that the flaws are the result of IGSCC. The fabrication records were reviewed as part of the determination of the cause of the cracking as well as to identify the appropriate material properties for the flaw evaluations. The review of the material behavior and other aspects provided corroboration of the conclusion that the flaws were IGSCC and thus a conservative crack growth rate was selected for the flaw evaluations.

The flaw evaluation was supported by a thorough and complete review of the applicable loads and load combinations for the affected piping. The latest design basis information regarding RRLB, MSLB and seismic loads were incorporated into the loads definition. A detailed piping analysis was performed for the defined loading conditions. The piping modelling included such details as the rotational stiffness properties of the penetration assemblies and the gap type supports. The results of the piping analysis represent an accurate and complete definition of the critical flaw section stresses under design basis and beyond design basis load combinations. The key analysis parameters associated with the loadings, material properties and system operating conditions were reviewed and enveloped by the analyses performed.

The flaw evaluations and sensitivity study were performed using the ASME Section XI, Appendix C limit load methods. The evaluations performed include an assessment of the key analysis parameters and provides results based on the limits of these parameters. The critical elbow flaw has a load margin under design basis load combinations of 17 times the ASME code factor of safety. The sensitivity study concluded that even with consideration of all of the upper bound limits of the analysis parameters a load margin of 2.85 times the ASME code safety factor exists for design basis load combinations. This load margin corresponds to an operating cycle length of 78 months with the upper bound crack growth rate prior to meeting the code specified factors of safety. These results clearly corroborate the conclusion that the core spray piping is very flaw tolerant and has sufficient margin to perform it's design basis function for the next operating cycle.

The leakage flow was calculated using the end of the operating cycle crack lengths in conjunction with the bounding flaw section stresses. The estimated leakage of 5 gpm for the system operating flow rate of 4500 gpm results in no significant increase in the peak cladding temperature (PCT). The leakage associated with the end of life crack size is 32 gpm per loop and is well within the analyzed leakage for core spray flow. The effect of these core spray flaw leakages along with the other cumulative leakages were assessed with respect to the PCT. The results of the analysis show that the PCT is well within the 10CFR50.46 limits.

A bounding failure assessment was performed to verify that adequate design margin exists. This assessment was performed using both a deterministic and probabilistic approach. The deterministic approach evaluated three scenarios: 1) reactor recirculation line break, 2) SSE and 3) reactor recirculation line break with single failure of the LPCI injection. In each of the scenarios, core cooling can be maintained with existing ECCS systems. The probabilistic approach postulated two scenarios: 1) reactor recirculation line break in combination with a failure of LPCI injection and 2) reactor recirculation line break in combination with a failure of the LPCI injection and concurrent SSE. The frequency of these events was calculated to be 1.6×10^{-8}/year and 1.0×10^{-15}/year, respectively. Thus, both scenarios can be concluded to be non risk significant. The potential effects of a loose part resulting from the cracked core spray sparger was evaluated. It was postulated that an elbow of the lower core spray sparger inlet piping breaks off and falls into the reactor vessel annulus region and that debris is created as a result of the rubbing and scrapping of the elbow on internal vessel components. Four safety and operational concerns associated with the postulated loose part and debris were evaluated: 1) potential for fuel bundle flow blockage and consequent fuel damage, 2) potential for fretting wear of the fuel cladding, 3) potential for interference with control rod operation and 4) potential for corrosion or chemical reaction with other reactor materials. The evaluation found no safety or operational concerns associated with the postulated loose part or debris. The combined assessment of the system structural margin as well as core spray system functional capacity confirm the conclusion that sufficient margin exists to operate for one 24 month cycle with the identified flaws. ComEd will continue to monitor the condition of the degraded core spray welds by following the BWRVIP guidance for the next scheduled refueling outage currently scheduled for the Spring of 1998.

13.0 REFERENCES

1. GE letter dated February 25, 1996, from Robert Turpin (GENE) to Tom Wojcik (ComEd), "260° and 290° Core Spray Downcomers".

2. August 9, 1995 Letter from G. Selby of EPRI NDE Center, to J. Whitman of ComEd, "UT Technique for Dresden Core Spray Piping".

3. UT Examination Summary Sheet, Reports No. R-200, R-201 and R-202 "Quad Cities Unit 1 RPV Core Spray Piping", March 16, 1996.

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

5. Sargent & Lundy Calculation, QC-NPD-0038, Rev. 0

6. GE Drawing 921D265, Revision 1, "Reactor Pressure Vessel Thermal Cycles".

7. Quad Cities UFSAR.

8. GE Drawing 161F312, Revision 1, "Process Diagram - Core Spray System", Quad Cities Units 1 and 2.

9. GE Drawing 104R921, Revision 3, "Assembly - Reactor Nuclear Boiler".

10. GE Drawing 104R861, Revision 10, "Assembly - Reactor Nuclear Boiler".

11. Willamette Iron & Steel Co. Drawing E861 Sheet 1, Revision J, "Quad Cities 1 & 2 Core Structure, Shroud".

12. Willamette Iron & Steel Co. Drawing E861 Sheet 2, Revision B, Quad Cities 1 & 2 Core Structure, Shroud".

13. GE Drawing 885D485 Sheet 4, Revision 3, Quad Cities Reactor Vessel.

14. Sargent & Lundy Calculation 9389-64-DQ.

15. ASME B&PV Section XI Appendix C, 1989.

16. "Evaluation of Flaws in Austenitic Steel Piping," EPRI NP-4690-SR, July 1986.

17. ASME B&PV Section III, Appendix I, 1989

18. "Instability Predictions for Circumferentially Cracked Type - 304 Stainless Under Dynamic Loading," EPRI, NP-2347, April 1982.

19. "Degraded Piping Program Phase II, Semi Annual Report, October 1984 - March 1985," NUREG/CR-4082, BMI-2120, Vol. 2, July, 1985.

20. "PICEP: Pipe Crack Evaluation Program (Revision 1)," EPRI NP-3596-SR, Revision 1, December 1987

21. "The Two-Phase Critical Discharge of Initially Saturated or Subcooled Liquid," Nuclear Science and Engineering, 41, 1970

22. "Pipe Fracture Database," from Douglas A. Scarth, Task Group Chairman, ASME Section XI Task Group on Pipe Flaw Evaluation, October 21, 1992

23. GE-NE-B1301725-01, Revision 3, "Safety Evaluation for Quad Cities Unit 1 Core Spray Line Repair", January 1996.

24. March 22, 1996, ComEd memorandum, "PRA Group Frequency Estimates for Two Failure Scenarios", from R. H. Johnson to K. R. Sturtecky.

25. NEDC-30936P-A, "BWR Owner's Group Technical Specification Improvement Methodology" (with Demonstration for BWR ECCS Actuation Instrumentation) Part 1, December 1988.

26. SIA Calculation CECO-42Q-301, Rev. 0, "Compliance Study of Flawed Pipe".

c:\tjb\sl5060.wpf