Dresden Unit 2
Jet Pump 15/16 Riser Weld
Flaw Evaluation Report

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1.0 Executive Summary

During D2R15 refueling outage, In Vessel Visual Inspection (IVVI) of the ten Jet Pump Risers (JPRs) was performed in accordance with the guidelines delineated in BWR Vessel Internals Project (BWRVIP) inspection and evaluation document BWRVIP-41. The scope of the inspection consisted of enhanced visual inspection technique for five welds designated as RS-1 through 5 on each of the ten risers assemblies. These welds are considered high priority welds since their extensive degradation could lead to jet pump disassembly and could potentially reduce the ability to maintain 2/3 core coverage. One 1.5" indication was observed along the heat affected zone of the riser elbow at the RS-1 weld of the JPR 15/16.

The indication is typical of IGSCC in stainless steel and is treated as a through wall flaw. To demonstrate structural integrity, the observed indication was evaluated using ASME Section XI, Appendix C flaw evaluation methods with the bounding IGSCC growth rate of $5x10^{-5}$ inches per hour. The evaluation has determined that even after two fuel cycles; the projected RS-1weld flaw length is 3.8 times less than the structural integrity limit.

The impact of the flaw on peak cladding temperature (PCT) during the design basis LOCA in combination with a bounding single failure was also evaluated. This evaluation demonstrated that ECCS flow losses from the flaw would have no adverse impact on the PCT.

In addition to the above evaluations, bounding deterministic and probabilistic assessments were performed using the generic assessments presented in BWRVIP-28. The bounding deterministic assessment found that even with an assumed throughwall crack of approximately 96% of the JPR pipe circumference, adequate core cooling would be maintained under all design basis events. The worst case scenario (reactor recirculation suction line break combined with a JPR failure and JP disassembly) present an insignificant risk since its frequency is less than $1x10^{-06}$/year and is, therefore, beyond the design basis (NUREG-0800) frequency.

Failure of a JPR could potentially result in a loose part and debris within the vessel. The impact of loose parts and debris was evaluated and no safety concerns were identified.

Based on the evaluations presented in this report, it is determined that the observed flaw in the JPR 15/16 RS-1 weld does not pose a safety concern for the next two fuel cycles.

2.0 Introduction

The portion of the Dresden Unit 2 Recirculation piping addressed in this condition assessment is located inside of the reactor pressure vessel (RPV) next to the core shroud in the annulus region between the shroud and reactor vessel. It is a continuation of the recirculation "riser" piping from the recirculation pump discharge ring header between the reactor's N2 nozzles where this piping enters the reactor and where each riser turns back down at the top of the ten jet pump pairs. A typical representation of a riser and jet pump pair is illustrated in figure 2.1. The riser and weld designations as identified by the "BWR Jet Pump Assembly Inspection and Flaw Evaluation (BWRVIP-41) Guideline", reference 1, are illustrated in figure 2.2. The jet pump riser piping was fabricated from 10" nominal, schedule 40 piping components.

In March of 1998, Dresden Programs Engineering initiated the planned D2R15 In Vessel Visual Inspection (IVVI) of the Jet Pump riser piping. The D2R15 IVVI riser piping inspections were augmented examinations performed to meet the recommendations of SIL-605. The inspection plan was based on BWRVIP-41 recommendations in anticipation of NRC approval of BWRVIP-41. These welds have not been closely examined since their installation before commercial operation in 1970. The inspection scope consisted of an enhanced visual inspection technique capable of resolving a 0.0005" diameter wire for the five welds designated RS-1, 2, 3, 4 and 5 on each of the ten riser assemblies. These welds are designated as "high priority" by BWRVIP-41 since failure of the riser pipe at any of these locations could result in the disassembly of the Jet Pump and affecting the ability to maintain 2/3 core coverage. This could potentially reduce the effectiveness of the LPCI system in maintaining fuel peak cladding temperatures to below limits specified in the LOCA Analysis (reference 2). The RS-1 weld location is the location at which flaws have been identified at Peach Bottom and LaSalle.

The flaw evaluation approach used to justify continued operation is a limit load analysis as described in BWRVIP-41, reference 1. The evaluation considered crack growth from IGSCC and fatigue mechanisms to determine the flaw size at end of evaluation period. This report provides the assessment criteria, design inputs and results for the various evaluations performed to evaluate the impact of this flaw on the operation of the plant.

Section 3 of the report provides a summary of the methods and the extent of the examination, as well as, a detailed definition of the identified flaw. Section 4 provides the material evaluation with an assessment of the root cause and definition of material properties and the crack growth rate used in the evaluation. The definition of the applicable loads and load combinations used are provided in Section 5. The flaw structural integrity and leakage evaluations are provided in Sections 6 and 7 respectively. Section 8 provides a description of the impact of recirculation system leakage on the LOCA Analysis. Failure Assessments are given in Sections 9 and a loose parts evaluation is provided in Section 10. A summary of the results and conclusions are provided in Section 11.

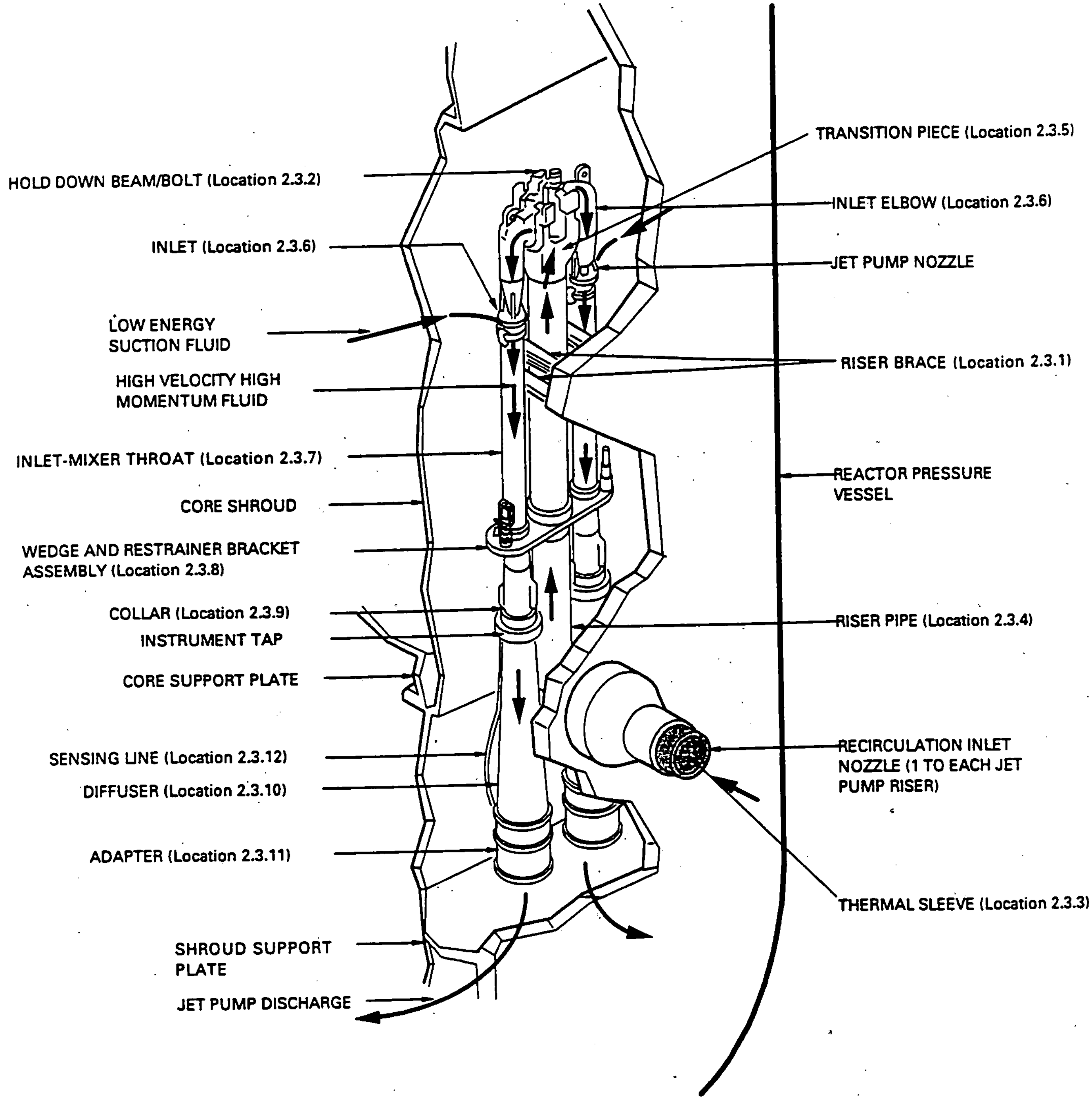

Figure 2.1: Typical Jet Pump Assembly

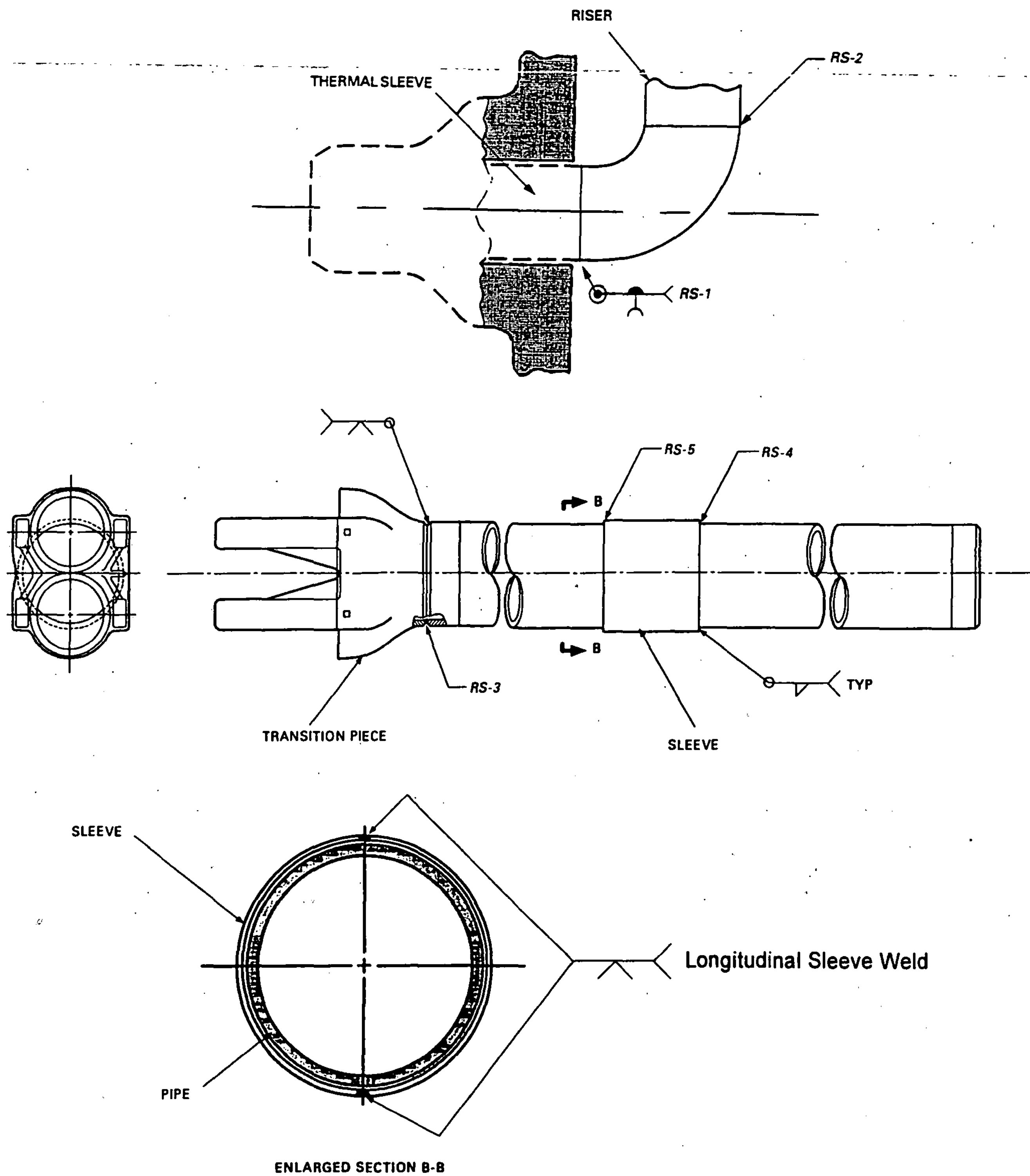

Figure 2.2: Riser Weld Designations

3.0 Description of Indications

The internal jet pump riser piping examinations conducted at Dresden during D2R15 refueling outage were performed in accordance with the "BWR Jet Pump Assembly Inspection and Flaw Evaluation (BWRVIP-41) Guideline", reference 1, and SIL-605.

Level II General Electric technicians using a RJ-2100 color video recording camera system performed the inspection. The inspection provided 100% coverage of all RS-1 welds. Resolution verification of a 0.0005" wire was performed per the EVT-1 method. A thorough soft nylon bristle brushing was necessary to remove a significant corrosion deposit layer. Access to some surfaces of the remaining welds was limited, in particular, access to the shroud side of some of the RS-3, -4 and -5 welds was restricted or not possible due to gusset plates on the restrainer brackets and the shroud repair hardware installed during D2R14.

The single identified flaw was observed along the heat affected zone (HAZ) of the riser elbow at the RS-1 weld between jet pumps number 15 and 16 (245° vessel location). The flaw is 1.5" in length centered at about the 270°(9:00 o'clock) position, or on the jet pump 16 side of the riser. This information is documented in the General Electric Inspection Services IVVI Report, reference 4.

Table 3.1 Summary of Flaw Lengths

Flaw Location	Weld	Measured Flaw Length (inches)[1]	Crack Growth Rate per Cycle (inches)[2,3]	Evaluated Flaw Length (inches)[4]
Riser between Jet Pumps 15/16	RS-1	1.5"	1.6"	4.7"

Notes:

1. Measured Flaw Lengths (MFL) were obtained with a flat scale held along the flaw and conservatively sized. Repositioning of the scale was not required. Measurement uncertainty is included in the stated length.

2. 5.00×10^{-5} inches per hour represents an upper bound limit for IGSCC crack growth in ductile materials (reference 4).

3. Crack growth per cycle (CG/C) is based on a two year fuel cycle at 100% availability minus a 60 day refuel outage (2 years x 335 days/year x 24 hours/day = 16,080 hours).

4. Two fuel cycles with growth at each end of the flaw, Evaluated Flaw Length, (EFL) = MFL + (2 cycles x 2Flaw Ends x CG/C), crack extension on both ends of flaw.

4.0 Materials Evaluation

4.1 Overview

The identified flaw was found in the Heat Affected Zone (HAZ) on the elbow side of the thermal sleeve to elbow weld at RS-1 on the Jet Pump Riser between Jet Pumps 15 and 16. This location is consistent with Intergranular Stress Corrosion Cracking (IGSCC) susceptibility requirements. This particular degradation mechanism is well documented for 300 series stainless steels exposed to high temperature BWR reactor water. Other BWR's have experienced IGSCC in this location including Peach Bottom and LaSalle.

4.2 Fabrication

General Electric drawings (GE#117C1475) indicate that riser components are fabricated of 10" NPS schedule 40, solution heat-treated type 304 piping products including an ASTM A-403 short radius elbow. The pipe is ASTM A-312 and the transition piece is cast. Welding was performed using the Gas Tungsten Arc (GTAW) process for "root" passes and the Shielded Metal Arc (SMAW) process for completion using E(ER)-308 filler. Tensile properties of the base material are used for the flaw analysis since the flaw originates in the HAZ of the base material and not within the weld.

4.3 Crack Growth Rate

The principle driving force propagating IGSCC flaws 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 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 5×10^{-5} inches/hour per NUREG-0313 (reference 4).

4.4 Material Behavior

The ductile or brittle response of the material of the jet pump riser piping components is evaluated with respect to initial characteristics and environmental degradation. All of the material used in the fabrication were austenitic stainless steels as indicated above. These materials do not undergo a phase transformation during thermal processing. The most significant material response to processing is carbide precipitation at the grain boundaries, which produces a zone next to the grains that is depleted of chromium. This condition is referred to as "sensitization" which is produced during welding. This condition influences the electro-chemical response of the material (increasing susceptibility to IGSCC), but does not alter the ductility or toughness of the material.

Exposure of the stainless steel material 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 $5x10^{20}$ n/cm^2. The neutron fluence in the area of the Jet Pump riser is less than the threshold limit, reference 1, therefore, no reduction in toughness or increased sensitivity to IASCC is expected.

4.5 Material Properties and Crack Growth Rate

In conclusion, the cracking observed in the Jet Pump riser piping is the result of IGSCC in austenitic stainless steel. The stresses driving the crack are self-relieving residual stresses indicating that the rate of crack growth will decrease as the cracking proceeds through wall. Therefore, the crack growth rate of $5x10^{-5}$inches /hour represents a conservative upper bound limit. In addition, the material properties of the jet pump riser piping will remain ductile throughout the life of the system.

5.0 Load Definitions and Load Combinations

5.1 Loads

The following loads were developed in the analysis performed by General Electric in GE-NE-523-B13-01869-054 (reference 5, Attachment A):

- Dead weight
- Hydraulic
- Seismic Inertia
- Fluid Drag
- Thermal Loads
- Vibration (flow induced)

5.2 Load Combinations

The bounding load combinations were considered and are consistent with the Dresden UFSAR. Load combinations were considered for the evaluation of the "Normal/Upset", the "Emergency/Faulted" and the "Beyond Design Basis" conditions. The "Beyond Design Basis" conditions combines both Safe Shutdown Earthquake and LOCA loading.

5.3 Combined Stress Summary

The calculated values of the membrane stress, P_m, bending stress, P_b, and secondary stress, P_e, at the RS-1 weld location are summarized in the following table for the governing Normal/Upset and Emergency/Faulted (including the beyond design basis) load combinations. See Attachment 'A' for additional details.

Table 5.1 Summary of Calculated Stresses

Load Case	P_m (psi)	P_b (psi)	P_e (psi)
Normal/Upset Combination	1607	1257	449
Emergency/Faulted Combination	1653	1347	449

6.0 Fracture Mechanics Evaluation

6.1 Load Limit Methodology

In this evaluation, the flaw in RS-1 is conservatively modeled as a throughwall flaw. The load limit methodology consistent with ASME Section XI, Appendix C, reference 6, was used in calculating the allowable flaw lengths. The welding processes used for fabrication of the RS-1 weld was a combination of Shielded Metal Arc (SMAW) and Gas Tungsten Arc (GTAW) welding. Since a non-flux process (GTAW) was specified for only part of the weld, it must be assumed that the weld is a flux weld and an appropriate "Z" factor was used to account for the reduced toughness of flux weld.

The stresses from the table in the preceding section were utilized to determine the acceptable throughwall flaw lengths. The acceptable flaw size was determined using the ASME Section XI, Appendix C, safety factor of 2.77 for the normal/upset condition and 1.39 for the emergency/faulted condition. The flow stress was taken as $3S_m$ (S_m=16.9 ksi for Type 304 stainless steel at 550°F). Specific equations for this evaluation and additional details can be found in Attachment A. The end of the evaluation period allowable flaw length without considering crack growth for the RS-1 weld is 17.9 inch.

6.2 Crack Growth Evaluation

Prior crack growth analyses performed for BWR shroud and core spray line indications have used a bounding IGSCC growth rate of 5×10^{-5} inch/hour of hot operating time. This crack growth rate was used for this evaluation. The evaluation period is defined as two fuel cycles of 16,080 hot operating hours each. Each end of the flaw will grow 0.8" during each fuel cycle for a total of 3.2" of crack growth during two fuel cycles.

Fatigue crack growth due to thermal cycling and flow induced vibration (FIV) was evaluated and determined to be insignificant for this weld. The FIV stress intensity factor range, ΔK_{FIV}, for the allowable flaw length is below the stress intensity factor threshold for fatigue crack growth, $\Delta K_{th} = 3$ ksi-$\sqrt{in}$, and consequently is not a concern for the significantly smaller RS-1 flaw. The fatigue crack growth from thermal cycling is negligible due to the small thermal stress intensities and limited number of thermal cycles, i.e. heatup and cooldown, during the evaluation period. See Attachment A for additional details of the fatigue crack growth evaluation.

The total crack growth, i.e. combining the crack growth at each end of the flaw, was determined to be 3.2" for the two fuel cycle evaluation period.

6.3 Flaw Evaluation Margin

The end of evaluation period flaw size is determined to be 4.7" for this evaluation. This flaw length is 3.8 times less than the allowable flaw length of 17.9" and therefore ensures the structural integrity of the weld for the evaluation period.

7.0 Flaw Leakage Calculation

The maximum leakage rate through the end of evaluation period flaw was conservatively estimated assuming an incompressible Bernoulli flow. The flow coefficient was assumed to be 1.0, which is very conservative for IGSCC flaws with rough, jagged surfaces. The leakage rate was based on the maximum riser ΔP of 190 psi which bounds the normal, upset and faulted conditions for the jet pump. The crack opening was conservatively idealized to be a rectangular slot with an area based on the end of evaluation period flaw length and the crack opening displacement. The crack opening displacement is calculated using $\delta = 4\sigma l/E$ from EPRI Report NP-2472, Vol. 2, where σ is the applied stress, l is one half the flaw length and E is Young's modulus. See Attachment A for additional details of the leakage rate calculation.

Figure 4 in Attachment A presents the leakage rate as a function of the flaw size. For the end of evaluation period flaw size of 4.7", the leakage rate is conservatively estimated to be 11 gpm.

8.0 Jet Pump Assembly LOCA Evaluation

8.1 Jet Pump Assembly Configuration and Function

The jet pumps are located in the annulus region between the core shroud and the vessel wall and provide core flow to control reactor power. Dresden Unit 2 has 10 pairs of jet pumps. Each jet pump assembly is composed of two jet pumps and a common jet pump riser pipe. Each of two Recirculation pumps provides "driving" flow to five risers (ten jet pumps). Additional "driven" flow is entrained in the Jet Pump inlet-mixer sections from the annulus region, which is then directed to the lower plenum region and into the bottom of the core. The driven to driving fluid ratio is about 2:1 for a total rated core flow of 98 x 10^6 pounds per hour.

The jet pump riser assembly is a 10" schedule 40 pipe, internal to the RPV, which connects the recirculation pump discharge line to the jet pump pair. A riser brace attaches the riser pipe to the vessel wall to provide lateral support. Each jet pump has an inlet-mixer assembly and a diffuser assembly. The inlet-mixer assembly consists of a 180° elbow, also known as a "rams head", a nozzle section with suction inlets, and a mixing section. The inlet-mixer assembly is clamped to the riser transition piece by the beam-bolt assembly, and fits into a slip joint at the top of the diffuser assembly. A restrainer bracket attached to the riser provides lateral support for each mixer section to increase the stiffness of the assembly and reduce the effects of vibration. The diffuser assembly consists of a gradual conical section terminating in a straight cylindrical section at the lower end, which is welded to the shroud support plate. Instrumentation monitors jet pump flow through the diffuser to ascertain individual and collective jet pump flow rates under various operating conditions.

The jet pump assemblies are non-ASME Code components and are classified as Safety Related. For post-accident core reflooding the jet pump assembly assures reflooding to 2/3 core height. An additional safety function of the jet pump assembly at Dresden Unit 2 is to provide a flow path for LPCI flow into the core. Assuming intact jet pump assemblies, there is no recirculation line break scenario which can prevent reflooding of the core to 2/3 core height, the height of the jet pump suction inlets.

8.2 Leakage Flow Evaluation

The bounding case for Dresden Unit 2 is the DBA-LOCA consisting of a reactor recirculation suction line break in combination with a single failure of the LPCI Injection Valve (SF-LPCI). The Peak Cladding Temperature (PCT) for the SF-LPCI case is 122°F greater than the closest limiting single failure event, i.e. Diesel Generator (SF-DG). The low pressure Emergency Core Cooling Systems (ECCS) available for the SF-LPCI event are 2 Core Spray (CS) pumps. The low pressure ECCSs available for the SF-DG event are 2 LPCI pumps and 1 CS pump. Since the jet pump riser is the injection path for LPCI, the leakage from the jet pump riser will affect only the SF-DG event and its results on PCT. For the SF-DG event, ECCS flow reduction is caused by the failure of the LPCI minimum flow valve to close and the new jet pump riser leakage (11 gpm). Per Reference 7, if the minimum flow valve stays open and there is additional leakage (11gpm) from the jet pump riser flaw, the degradation in the LPCI flow could potentially be 706 gpm. Based on the results of the LOCA analysis (Reference 2), the total ECCS flow for the SF-DG event (after reducing the flow by 706 gpm) is still greater than the total ECCS flow for the SF-LPCI event. The SF-DG event with the additional 11 gpm flow reduction is, therefore, not limiting. Per Reference 8, a LPCI flow decrease of 291 gpm for two pumps could result in an increase of 10°F in PCT for the SF-DG event. This increase is well below the PCT differential of 122° cited earlier between the SF-LPCI and SF-DG events.

The flow degradation due to jet pump riser 15/16 RS-1 weld flaw will affect the small break accident scenario. However, since the margin between the large break LOCA PCT is approximately 150°F greater than the small break LOCA PCT, the small break will not become limiting.

9.0 Bounding Failure Assessment

This bounding failure assessment is based on the generic evaluations performed by the BWRVIP in BWRVIP-28, Reference 9, currently under review by the NRC. This assessment utilizes both a deterministic and probabilistic approach for the worst case of a failed riser. Section 9.1 presents the details of the deterministic assessment and Section 9.2 presents the probabilistic assessment.

9.1.1 Extent of Cracking Required for Riser Failure

Per the BWRVIP-28 safety assessment, the extent of cracking required to lead to complete riser pipe severance is a throughwall crack of approximately 96% of the pipe circumference for the worst case LOCA loads. This degree of cracking has never been reported in any BWR piping system. The weld flaw, inclusive of calculated crack growth for the next two fuel cycles, in JPR 15/16 is approximately 15% of the pipe circumference.

9.1.2 Riser Failure During Normal Plant Operation

The failure of a jet pump riser, with or without jet pump disassembly, during normal plant operation poses no safety concern because the resulting reduction in core flow will reduce reactor power and increase the fuel thermal margin. Detection of the failure is possible due to these symptoms together with the change in other parameters, such as jet pump flow. Therefore, safe shutdown of the reactor is not affected by the postulated riser failure.

9.1.3 Riser Failure During Anticipated Transient

The failure of a jet pump riser, with or without jet pump disassembly, during an anticipated transient (e.g., turbine trip), does not result in a degradation of the fuel thermal margin. The consequences of riser failure are less significant that the effect of a recirculation pump trip initiated at many plants to reduce a power increase that may be caused by the transient. Recirculation flow decrease events are not likely to cause riser pipe failure due to the decreased load on the riser during this type of event. However, if riser severance were to occur, the event thermal margin would be slightly degraded. Flow decrease events are not limiting events. Therefore, satisfactory fuel thermal margins would be maintained during all transients, and safe shutdown of the reactor is not affected by the postulated riser failure.

9.1.4 Riser Failure during a Non-Recirculation Line LOCA or Seismic Event

The failure of a jet pump riser with or without jet pump disassembly, during a postulated non-recirculation line LOCA does not result in a significant degradation of the fuel thermal margin. Non-recirculation line LOCA does not result in a significant degradation of the fuel thermal margin. Non-recirculation line breaks are all above the top of the core and jet pump integrity is not required to ensure reflooding of the core. Similarly, for a seismic event, there is no LOCA so jet pump integrity is not required to ensure reflooding of the core. Therefore, safe shutdown of the reactor is not affected for either event by the postulated riser failure.

9.1.5 Riser Failure During a Small Recirculation Line LOCA

The failure of a jet pump riser, with or without jet pump disassembly, during a postulated small recirculation line LOCA does not result in a significant degradation of the fuel thermal margin. Small recirculation line breaks are not affected because the Emergency Core Cooling System (ECCS) capacity is sufficient to overcome the water lost out of the break and flood the vessel. Therefore, safe shutdown of the reactor is not affected by the postulated riser failure.

9.1.6 Riser Failure During a Large Recirculation Line LOCA

The failure of a jet pump riser, without jet pump disassembly, during a large recirculation line LOCA does not result in a degradation of fuel thermal margin. The existing LOCA analysis was evaluated for the potential JPR leakage due to a conservative crack size calculated for the two cycles of plant operation. Based on the evaluation documented in Section 8, it is concluded that the observed crack will have no impact on the PCT.

A probabilistic assessment was performed for the very selective combination of events involving jet pump riser failure, jet pump disassembly, and a large recirculation line LOCA in order to quantify this potential risk to continued plant safety. This evaluation is presented in Section 9.2.

9.2 Probabilistic Assessment

The extremely low potential for a jet pump riser cracking condition to pose a safety concern can be inferred from the discussion in the preceding deterministic evaluation. However, despite the evidence of the extremely selective conditions necessary to challenge plant safety, a probabilistic evaluation was performed to quantify this extremely low potential. The probabilistic assessment is based on the generic evaluation of BWRVIP-28 and plant specific large break LOCA frequency calculation. For the probabilistic assessment, it was conservatively assumed that:

a) A significant (approximately 96%) circumferential throughwall cracking at the riser elbow to nozzle thermal sleeve weld will occur during the next two fuel cycles. This is contrary to the projected flaw size of 4.7 inches (approximately 15% of the circumference).

b) Leakage from a severely cracked riser pipe is not detectable

c) Riser pipe failure causes jet pump disassembly

d) One core spray system will fail (single failure)

e) Operation of a single core spray system is not sufficient

f) LPCI is not effective

g) Linear heat generation rate is greater than the acceptable level

It was further assumed that a LOCA in either A or B recirculation loop will cause sufficient acoustic loading in the reactor vessel to cause the flawed jet pump riser to fail. Based on the above assumptions, probabilities were calculated for complete jet pump riser failure resulting from a large break LOCA.

9.2.1 Probability of a Recirculation Suction Line LOCA

The recirculation system suction line break is the only LOCA that produces acoustic loads that could result in a net increase in the loads tending to separate the riser elbow from the nozzle thermal sleeve. The frequency of such LOCA for Dresden Unit 2 is calculated in Reference 10 as 6.96 E-06 large break failures/year.

9.2.2 Probability of Riser Failure

Per reference 9, a 96% throughwall crack (less than 4% ligament) would exist for less than a day (i.e.6 hours) based on calculated crack growth rates. Reference 9 conservatively assumed the condition would exist for one day. Yielding a probability of the condition existing in a year as 1 day / 365 days.

9.2.3 Results of Probabilistic Evaluation

The probability per year of exceeding fuel cladding oxidation criteria as a result of the observed jet pump riser flaw at Dresden Unit 2 is determined by multiplying the probabilities calculated in Sections 9.21 and 9.2.2. The cumulative annual frequency is:

6.96 E-06/yr x 1/365 = 1.91 E-08/yr

This probability is less than 1.0 E-06/yr and is considered to be of minimal safety significance per the criteria found in NUREG 0800 Rev. 2.

10.0 Loose Parts Evaluation

As part of the evaluation of the cracked jet pump riser piping, a scenario has been postulated where as a consequence of the riser crack, jet pump assembly breaks apart and becomes loose parts inside the reactor vessel. These parts are assumed to fall into the vessel annulus region. An evaluation has been performed to address the safety concerns raised as a result of these potential loose parts.

10.1 Postulated Loose Parts

The postulated loose part is a broken fragment of the jet pump beam assembly. Based on the location of the observed cracking, if the subject weld (RS-1) were to fail, the associated jet pumps could disassemble. The Jet Pump riser, which is 10" schedule 40 piping, would probably fall down into and become lodged in the annulus. The Jet Pump beam assembly would fail as the transition piece fall away from the bottom of the inlet elbows, and backpressure from the remaining operating jet pumps would eject the mixers from their slip-fits in the diffusers. It is unlikely that a portion of the riser piping would break loose and become a loose part. However, there may be debris generated as a result of the rubbing and scraping of the riser piping on the other components in the annulus.

10.2 Safety and Operational Concerns

The safety and operational concerns associated with the postulated loose parts 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,1
- Potential for corrosion or chemical reaction with other reactor materials.

The riser piping, jet pump mixer, inlet elbows, transition pieces and beam assemblies are all postulated to break away and fall down onto the floor of the annulus area. This is reasonable since the openings at the top of the slip fit into the diffusers are very small, high above the annulus floor and is discharging flow from the lower plenum (an area of higher pressure as long as the recirculation pumps are operating). Also, this equipment normally occupies this area.

10.2.1 Potential for Fuel Bundle Flow Blockage and Consequent Fuel Damage

The riser elbow and jet pump piping assembly components are located in the annulus and due to their diameters, are too large to escape from the annulus region. The opening to the lower plenum at the top of the diffuser vacated by the ejected mixer is too small to permit passage of any piping components. Debris or a beam fragment, however, could be small enough to pass through a recirculation pump or the diffuser opening. A heavy beam fragment would then settle into the lower head drain area. Lighter pieces of debris, such as Jet Pump instrument tubing, may continue to be swept by flow velocities. Larger sizes of material would become entangled in the tie plates where no restriction to an even coolant flow to the fuel would result. Only very small and light material could continue up toward the orifices of the fuel support pieces where it would not significantly block flow. Therefore, no significant change in boiling transition effects would occur.

There is no significant concern for fuel bundle flow blockage due to postulated generation of debris caused by the failure of the Jet Pump riser piping.

10.2.2 Potential for Fretting Wear of Fuel Cladding

If a failure of a Jet Pump riser occurs and jet pumps disassemble, the reactor operators will diagnose the problem immediately when reactor power/flow anomalies become apparent. Operating procedures will require an immediate controlled shutdown.

If debris small enough to enter the fuel orifices is generated and does become entrapped in a fuel bundle spacer, it would require an extended time at flow to wear through the cladding. The result of such prolonged operation could be fuel cladding fretting and fuel leakage. Fuel leakage would be detected by the Off-Gas system, which monitors the plant's non-condensable gas effluent. Appropriate actions would be taken to maintain off-site release rates below acceptable limits.

10.2.3 Potential for Interference with Control Rod Operation

If debris is carried past the fuel support piece inlet orifices, to interfere with control rod motion, it must travel between and through the fuel bundle spacers, exit the fuel bundle past the upper guide and then reverse direction back into the area between the fuel channels. Since the channel surfaces are very smooth, there is no place for debris to lodge and block blade insertion motion. It must then drop down past the blade through the fuel support piece to blade clearance and into the control rod guide tube. Once in the guide tube, it must pass through the clearance between the blade velocity limiter and guide tube at the inside diameter of the guide tube, and then drift to the center of the guide tube to enter the drive's collet area. This is an extremely unlikely trajectory. If the loose part is small enough, it could enter the index tube ID between the spud fingers and settle in the inner filter. If the part travels down along the index tube OD, it would settle in the outer filter. This would all occur against CRD cooling flow and would seem to be very unlikely. If this should happen, the very small part would not have a sufficient mechanical strength to impair either the safety function (SCRAM) or normal control rod operation. Consequently, there is no concern for potential interference with CRD operation due to a postulated lost part.

10.2.4 Potential for Corrosion or Chemical Reaction with Other Reactor Parts

Since the postulated loose part is made of type 304 stainless steel, the predominant material of construction of reactor vessel internals, there is no concern for corrosion or chemical reactions with other reactor materials.

10.3 Loose Parts Monitoring

Dresden does not have a loose parts monitoring system. All reactor internals with threaded connections have thread-locking devices (keepers) to prevent disassembly. Hence, loose parts are not anticipated. Visual inspection to identify any loose or degraded components is performed on regularly scheduled intervals.

In the remote possibility that a part of the jet pump assembly becomes disassembled at power, the performance of the system is continuously monitored both indirectly by reactor power and directly by recirculation loop flow instrumentation. In the remote possibility of a loose part reaching and damaging fuel cladding, the Off-Gas system would detect the abnormal presence of fission products in the plant effluent. The Dresden Technical Specifications delineate the requirements for these monitoring systems, and station Operating procedures provide required actions when elevated release rates are indicated. Furthermore, the Main Steam Radiation Monitors will detect a large increase in fission products release rate (gross fuel damage) and will provide an automatic protective function to minimize the release of fission products. The Dresden Technical Specifications also delineate the instrumentation requirements and the actuation points. When actuated, the Main Steam Isolation Valves will close, and if the reactor is above 40% power, the reactor will automatically shutdown (SCRAM).

10.4 Conclusion of Loose Parts Evaluation

The loose parts evaluation for the postulated jet pump riser piping failure with resulting disassembly, and generation of debris has concluded that there is:

(1) no potential for significant fuel bundle coolant flow blockage,

(2) no safety concern for fuel cladding damage,

(3) no potential for interference with control rod operation, and

(4) no potential for chemical or adverse material reactions.

Thus, there are no significant safety concerns resulting from loose parts occurring during postulated jet pump riser failure and jet pump disassembly. Fuel cooling can be maintained and Control Rod motion can be achieved.

11.0 Summary and Conclusion

D2R15 IVVI for the jet pump risers identified a 1.5" long circumferential indication on the elbow side, heat affected zone of the RS-1 weld in the 15/16 jet pump riser. This indication was treated as a throughwall flaw and evaluated following the guidance in BWRVIP-41. All design basis loads acting on the RS-1 riser weld were determined, and the governing normal/upset and emergency/faulted load combinations were used to calculate the allowable flaw size. The 17.9" allowable flaw size was determined using the ASME Section XI, Appendix C, methodology for reduced fracture toughness of flux welds with the required safety factors of 2.77 for normal/upset and 1.39 for emergency and faulted loading conditions. Using the bounding IGSCC growth rate of 5×10^{-5} inches/hour, the throughwall flaw was determined to grow to 4.7" in length during an evaluation period of two fuel cycles, 16080 hot operating hours per fuel cycle. Fatigue crack growth from flow induced vibration and thermal cycling was determined to be insignificant for this flaw length. The result of these evaluations is that the allowable flaw size of 17.9" is 3.8 times larger than the projected flaw size of 4.7" after two fuel cycles.

Bounding flow loss from the flaw was conservatively determined to be 11 gpm using the maximum pressure differential and crack opening displacement for all operating conditions. The effect of the 11 gpm flow loss during LPCI injection was evaluated with other ECCS flow losses, and found no increase in the peak cladding temperature because this leakage is bound by the conservative leakage rates used in the current LOCA analysis.

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 safety assessments performed were based on the generic evaluations in BWRVIP-28. The deterministic assessments evaluated six different scenarios from normal, upset, emergency and faulted conditions. The deterministic assessment found that core cooling can be maintained by the existing ECCS systems and the reactor shutdown for all scenarios except for the very selective combination of the jet pump riser failure, jet pump disassembly and a large recirculation line LOCA. The probabilistic assessment was performed for this combination of events. Using the Dresden Unit 2 specific recirculation line LOCA frequency of 6.96×10^{-6} failures/yr, the probability of this very selective combination of events was determined to 1.9×10^{-8}/yr. This frequency is less than the 1.0×10^{-6}/yr criteria in NUREG 0800 Revision 2 and considered to be of minimal safety significance.

The potential effects of loose parts resulting from the flawed riser weld were evaluated. It postulated the failure of the riser weld and the disassembly of the jet pump and evaluated larger pieces falling into the annulus region and smaller debris migrating into the lower plenum through diffuser of the disassembled pump. 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,1

4. Potential for corrosion or chemical reaction with other reactor material.

The evaluation found no significant safety or operational concerns associated with the postulated loose part or debris.

The combination of the RS-1 weld structural integrity margin as well as the ECCS system functional capacity confirm the conclusion that sufficient margin exists to operate for two fuel cycles with the identified flaw. ComEd will continue to monitor the condition of the degraded jet pump risers by following the recommendations provided in BWRVIP-41 and reinspecting the jet pump 15/16 riser, RS-1 weld after two fuel cycles in refuel outage D2R17.

12.0 References

1. BWR Vessel and Internals Project: BWR Jet Pump Assembly Inspection and Flaw Evaluation Guidelines, BWRVIP-41, EPRI TR-108728, October 1997.

2. "Impact of Dresden Unit 2 Jet Pump Riser Cracks and Increased Core Spray Line Leakage on LOCA Analysis Results," NFS:BSA:98-035, April 7, 1998.

3. "InVessel Visual Inspection (IVVI) of Commonwealth Edison Company, Dresden Nuclear Power Plant (DRD) Unit 2 Internals During the 1998 D2R15 March Outage," DRD-98-1HDM9, GE Inspection Services, March 1998.

4. Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping, NUREG-0313, Revision 2, January 1988.

5. Jet Pump Riser Weld Flaw Evaluation Handbook for Dresden Unit 2 and Unit 3, GE-NE- 523-B13-01869-054, Revision 0, June 1997.

6. ASME Boiler and Pressure Vessel Code, Section XI, 1989 Edition.

7. "Updated Principal LOCA Analysis Parameters for Dresden Units 2 and 3," EMF-93-176, Revision 5, Siemens Power Corporation, May 1997.

8. "Dresden LOCA-ECCS MAPLHGR Limits for ATRIUM-9B and 9x9-2 FUEL," EMF-97-031(P), Revision 1, Siemens Power Corporation, May 1997.

9. BWR Vessel and Internals Project: Assessment of BWR Jet Pump Riser Elbow to Thermal Sleeve Weld Cracking, BWRVIP-28, EPRI TR-107667, December 1996.

10. "Dresden Station Frequency Estimate for Concurrent Loss of Coolant Accident and Failure of a Jet Pump Assembly," DRE98-0073, Revision 0, April 2, 1998.

Appendix A:

Jet Pump Riser Weld Flaw Evaluation Handbook for Dresden Unit 2 and Unit 3

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GE-NE-523-B13-01869-054, Rev. 0
June 1997

JET PUMP RISER WELD FLAW EVALUATION HANDBOOK FOR DRESDEN UNIT 2 AND UNIT 3

June 1997

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Attachment A

JET PUMP RISER WELD FLAW EVALUATION HANDBOOK FOR DRESDEN UNIT 2 AND UNIT 3

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Attachment A

TABLE OF CONTENTS

1. Purpose/Objective

The objective of this report is to document the results of a fracture mechanics evaluation of the Dresden Unit 2 and Unit 3 jet pump riser pipe circumferential welds. This evaluation results in the allowable end-of-cycle flaw lengths at the three riser piping circumferential welds. Figure 1 is a schematic showing the welds of interest which are labeled welds 1, 2, and 3.

The results presented in the flaw evaluation handbook can be used to disposition indications if found in the future in the jet pump riser pipes at Dresden Unit 2 and Unit 3.

2. Methods

This section presents the methodology and procedure used in performing the jet pump riser pipe weld flaw evaluation. Following are the steps used in the analysis.

1. Review of the reference drawings. The dimensional tolerances specified on the reference drawings are such that any variations within those values will have insignificant impact on the calculated stress values.

2. Determine the loading and load combinations.

3. Create a SAP4G07 (Reference 1) finite element model for the jet pump. Anchor connection points are the recirculation inlet nozzle, shroud support plate and riser brace.

4. Determine the membrane and bending stresses considering the load combinations.

5. Use the limit load methods of Paragraph IWB-3640, Section XI, ASME Code (see References 2 and 3) as a guide to determine the allowable flaw lengths. Section XI evaluation procedures are used as a guide, since the jet pump is not a part of the reactor pressure boundary.

6. Evaluate IGSCC and fatigue crack growth rate. Calculate IGSCC crack growth for two year (16,000 hrs) cycle based on a growth rate of $5x10^{-5}$ inch/hot hour. Determine if fatigue crack growth rate due to vibration is significant by calculating the stress intensity factor due to flow induced vibration and comparing with threshold stress intensity. If actual stress intensity is less than threshold, than given crack is acceptable.

7. Leakage curves are calculated which show leakage versus percent of Section XI allowable flaw during normal plant operation.

3. Assumptions

1. The jet pump geometry is as described in the reference drawings (Reference 4). The dimensional tolerances specified on the reference drawings are such that any variations within those values will have insignificant impact on the calculated stress values. It was also judged that any deviations between the as-built geometry and the geometry indicated in the reference drawings would not be significant in terms of stress analysis and the allowable flaw calculations.

2. The calculations are based on one flaw per riser. However, synergistic effects of multiple flaws in one riser are negligible and would not affect the results of this analysis. The reason is that even large flaws (180 degrees) do not significantly change the stiffness of the riser and therefore the response to input loadings does not change. Note that neglecting the synergistic effects is conservative because even a small decrease in stiffness due to flaws would reduce loads because of increased system compliance.

3. Fatigue due to thermal stresses is negligible due to very few transients experienced during one fuel cycle.

4. Design Inputs

The design inputs in this evaluation consisted of the geometry of the jet pump and the applied loads. The geometry of the jet pump was obtained from the drawings listed in Reference 4. A search of GE documentation for fabrication changes found no field deviations that will affect the results of this analysis.

The jet pump riser pipe is 10-inch schedule 40 and the material is Type 304 stainless steel (Reference 4). Figure 1 shows a schematic of the jet pump. For convenience of identification, the welds in Figure 1 have been arbitrarily numbered from 1 through 3. A finite element model was developed to determine the stresses from various design loads. Figure 2 shows a line plot of the finite element model. The SAP4G07V finite element program was used to perform the stress analysis.

4.1. Static Loads

The applied loads on the jet pump assembly consist of the following: deadweight, seismic inertia, hydraulic, fluid drag, and thermal expansion. Reference 5 lists the report number from which the loads were obtained. Each of these loads are briefly discussed next.

4.1.1. Deadweight

The deadweight loading consists of the weight of the jet pump. The stresses for this loading were calculated by applying one 'g' vertical acceleration in the finite element model of the jet pump assembly. For flaw evaluation purposes, the stress from this loading is treated as primary. The designation for this load is:

Deadweight: DW

4.1.2. Hydraulic Loads

The hydraulic loads acting on the jet pump are calculated by summing the fluid momentum and pressure forces in the vertical and horizontal directions. This load definition considers any pressure differences between the annulus and the jet pump. Two hydraulic force values are calculated and applied to the jet pump. The first value is the longitudinal force in the riser pipe which puts the riser welds in tension. The second value is the net vertical hydraulic load. The net vertical force is predominately caused by the pressure difference between the jet pump and annulus at the slip joint. Because the slip joint can not transmit a vertical load, the vertical load is carried through the riser pipe causing a bending moment on the riser. The following designations are used:

Hydraulic Load: F

The values of the hydraulic loads are presented in the following table.

Hydraulic Loads

Riser Longitudinal Load (lb)	Net Vertical Load (lb)
15112	7518

4.1.3. Seismic Inertia

The seismic inertia loading consists of horizontal and vertical inertia forces acting on the jet pump due to seismic excitation of the RPV (Reference 5). The locations where the seismic excitation is imparted to the jet pump are the vessel recirculation inlet nozzle, the shroud support plate and the riser brace. The following designations are used:

Operating Basis Earthquake Inertia: OBEI
Safe Shutdown (or Design Basis) Earthquake Inertia: SSEI

The natural frequency of the jet pump is high (>20 Hz) such that the zero period acceleration (ZPA) values of the acceleration may be used in a static analysis. The values used in the evaluation are shown in the following table.

Seismic Accelerations

	Horizontal	Vertical
OBEI	0.57	0.067
SSEI	1.13	0.133

4.1.4. Fluid Drag

The drag loads consist of the forces resulting from the fluid flowing in the annulus region past the jet pump. A postulated recirculation line break LOCA (suction side) subjects the jet pump to a drag force load in a tangential direction relative to vessel centerline. A previous TRACG analysis (Reference 6) performed for a similar size plant calculated the worst case flow velocities past the jet pump assembly. The worst case velocities correspond to a suction side recirculation line break LOCA. Other breaks do not affect the jet pump nearly as severely due to the other lines proximity or size. The velocities correspond to the jet pumps nearest to the suction nozzle. The horizontal drag loads on the jet pump were determined to be approximately 600 lb on the riser and 1600 lb on the diffuser. The following designation is used:

Drag Loads During LOCA Condition: DRG

4.1.5. Thermal Loads

The three anchor points of the jet pump (the recirculation inlet nozzle, riser braces on the vessel, and the shroud support plate) grow vertically and horizontally at different rates due to differences in the materials (low alloy steel for the vessel, versus stainless steel for the jet pump). The loads produced by the thermal expansion are treated as secondary. The following thermal displacements are considered:

Displacements during Normal Operation: NOD

The displacements are calculated at normal operating temperature which is 520°F for region B according to the reactor thermal cycle diagram (Reference 7).

4.2. Vibration

The flow induced vibration (FIV) loads are caused by turbulent flow in the piping exciting the natural frequencies of the jet pump assembly. The method of calculating the vibration stress from the test data can be summarized as follows:

1. Review the startup vibration data (Reference 8 & 9) to determine the primary modes of interest for the jet pump.
2. Using a finite element model of the jet pump, determine the natural frequencies, mode shapes, and modal stresses of all modes of interest.
3. Normalize the modal stresses such that the they are equal to the measured displacement data observed during startup testing.
4. Select the normalized modal stress at the weld location on the riser pipe for each mode. Add the modal stresses from each mode using SRSS. This is the FIV stress.

The following designation is used:

Flow Induced Vibration Stresses: FIV

5. Load Combinations And Stress Levels

This section describes the manner in which the various loads were combined for the purpose of obtaining stress levels for the flaw evaluation. The limiting stress levels at the welds are then summarized.

5.1. Load Combinations

The flaw evaluation methodology to be used makes the distinction between primary and secondary stresses by specifying different safety factors. The flaw evaluation methodology also makes the distinction between the normal/upset (Level A/B) condition loads, for which the factor of safety is 2.77, and the emergency/faulted (Level C/D) condition loads, for which the safety factor is 1.39. The load combinations are consistent with Dresden UFSAR.

The following set of load combinations were considered for the evaluation of normal/upset condition:

(1) DW(P) +F(P) + FIV(P) + NOD(S)
(2) DW(P) +F(P) + FIV(P) + OBEI(P) + NOD(S)

Note that the letter in the parenthesis indicates whether a load is primary or secondary as defined by the ASME Code. The set of load combinations used for the Faulted conditions are the following:

(3) DW(P) +F(P) + FIV(P) + SSEI(P) + DRG(P) + NOD(S)
(4) DW(P) +F(P) + FIV(P) + SSEI(P) + NOD(S)

5.2. Calculated Stress Levels

The forces and moments at various nodes in the model for all of the load sources were calculated using the SAP4G07V finite element code (Reference 1). These forces and moments were then combined to obtain the total forces and moments for a given load combination. Thus, for each load combination and each node, a set of forces and moments were obtained. Furthermore, within each set, the forces and moments from the displacement-controlled loadings were tabulated separately for the calculation of expansion stress. As described later, the flaw evaluation methodology uses the primary membrane (P_m), primary bending (P_b) and the expansion stress (P_e).

The calculated values of P_m, P_b and P_e stress levels at the circumferential weld locations are summarized in the following tables for the governing load combinations for Normal/Upset and Emergency/Faulted service levels.

Summary of Calculated Stress at Circumferential Welds For Normal/Upset Load Combinations

	Unit 2			Unit 3		
Weld ID (Figure 1)	P_m (psi)	P_b (psi)	P_e (psi)	P_m (psi)	P_b (psi)	P_e (psi)
1	1607	1257	449	1606	1220	390
2	2111	1395	138	2098	1376	271
3	2191	400	112	2191	392	119

The stress levels in the preceding table were used in the allowable flaw evaluations as described in the next section.

Summary of Calculated Stress at Circumferential Welds For Emergency/Faulted Load Combinations

	Unit 2			Unit 3		
Weld ID (Figure 1)	P_m (psi)	P_b (psi)	P_e (psi)	P_m (psi)	P_b (psi)	P_e (psi)
1	1653	1347	449	1651	1304	390
2	2131	1405	138	2118	1386	271
3	2215	538	112	2215	513	119

6. Fracture Mechanics Evaluation

The limit load methodology was used in calculating the allowable flaw lengths. This methodology is first described followed by the results of allowable flaw evaluations.

6.1. Limit Load Methodology

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

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

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

Where,

- t = pipe thickness, inches
- α = crack half-angle as shown in Figure 3
- d = crack depth
- R = pipe radius
- β = angle that defines the location of the neutral axis
- Z = weld type factor
- P_e = piping expansion stress
- P_m = primary membrane stress
- P_b = primary bending stress
- P_b' = failure bending stress
- $\sigma_f = 3S_m$, flow stress
- S_m = allowable stress

The safety factor (SF) is then incorporated as follows:

$$P_b' = Z^*SF\,(P_m + P_b + P_e/SF) - P_m \qquad (3)$$

The P_m and P_b are primary stresses. P_e is a secondary stress and includes stresses from all displacement-controlled loadings such as thermal expansion, seismic anchor motion, etc. All three quantities are calculated from the analysis of applied loading. The safety factor value is 2.77 for normal/upset conditions and 1.39 for emergency/faulted conditions. The crack angle (2α) is the value for which equation 2 is equal to equation 3.

Z Factor

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

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

where OD is the nominal pipe size (NPS) in inches. The procedures of Appendix C recommend the use of OD = 24 for pipe sizes less than 24-inches. This approach is very conservative and, therefore, the use of actual NPS (OD=10 inches) was made in calculating the 'Z' factor. This approach is considered reasonable as recent discussions in the Section XI Code Working Group on Pipe Flaw Evaluation indicate that for small diameter pipes, such as the 10-inch diameter jet pump riser pipe, the Z-factor may be close to or less than 1.0. The welding process used was a combination of shielded metal arc type (SMAW) or submerged arc welds (SAW) and gas tungsten arc weld type (GTAW). Since a non-flux process (SMAW) was specified for only part of the weld, it must be assumed that the welds are flux welds. The Z-factor is thus:

$$Z_{10\text{-inch}} = 1.15\,[1 + 0.013(10-4)] = 1.24$$

6.2. Allowable Flaw Length Calculation

The stresses from the table in the preceding section were utilized to determine the acceptable through-wall flaw lengths. The acceptable flaw size was determined by requiring a safety factor on stress. The flow stress was taken as $3S_m$ (S_m=16.9 ksi for Type 304 stainless steel at 550°F). As specified in Reference 2, safety factors of 2.77 for the normal/upset conditions and 1.39 for the emergency/faulted conditions, respectively, were used. The calculated values of the end-of-cycle allowable flaw lengths are tabulated in the following table.

End-of-Cycle Allowable Flaw Lengths Based on Outside Diameter

Weld	Flaw Length (inch)	
	Unit 2	Unit 3
1	17.9	18.0
2	17.0	16.9
3	18.5	18.5

These allowable values are the end of cycle values and they do not consider the crack growth due to IGSCC or fatigue. The crack growth is discussed in Section 6.3. The crack growth rate must be added to the existing crack length at each end to determine whether the projected end of cycle length is acceptable.

6.3. Crack Growth Evaluation

Prior crack growth analyses performed for BWR shroud and core spray line indications have used a IGSCC crack growth rate of $5x10^{-5}$ inch/hot hour. This crack growth rate translates into a crack length increase per two year cycle of approximately (16,000 hrs x $5x10^{-5}$) or 0.80 inch at each end of an indication. Thus, the projected length, l_f of any indication whose current length at the time of inspection is, l_p, would be (l_p +0.80x2) inches. A factor of 2 in the preceding parenthesis is to account for the growth at each end of the indication.

In addition to IGSCC growth, fatigue growth due to flow induced vibration (FIV) is discussed. The expected fatigue growth is a strong function of the crack size and orientation and cannot be determined until an indication is characterized. With a characterized crack, the stress intensity factor (ΔK) can be computed and compared to the threshold stress intensity factor (ΔK_{th}). The ΔK_{th} is the value at which fatigue crack growth for high cycle stress becomes significant for high cycle events and must be considered. At values below ΔK_{th}, fatigue growth can be neglected. For 304 stainless steel, Reference 10 reports a ΔK_{th} value of 3.0 ksi$\sqrt{in}$. The allowable crack size due to FIV was based on the stress intensity calculation method described in Reference 11. The smallest calculated EOC crack size for the three locations is 24.1 inches. This value is greater than the limit load method allowables, therefore, the limit load method is governing and fatigue is not a concern for Dresden.

Thermal expansions were also considered in evaluating fatigue. However, the fatigue crack growth due to thermal expansion stress cycling is negligible due to the limited number of cycles and low thermal stress intensities. The loading cycles are primarily the heatup/cooldown events. Total crack growth for thermal transients is insignificant.

7. Leakage Calculation

Leakage from postulated through-wall flaws with length equal to the allowable end of cycle (EOC) flaw size are calculated in this section. The leakage rate through an indication was estimated assuming incompressible Bernoulli flow:

$$Q = CA\sqrt{2g_c \Delta P / \rho} \qquad (5)$$

where, Q = Leakage
C = flow coefficient
A = area
ρ = mass density of fluid
ΔP = pressure difference across the pipe/vent

A conservative riser ΔP value of 190 psi is used which bounds the jet pump normal flow conditions for the plant. This is the design value for the steady state pressure difference during the jet pump operation for Dresden.

Leak rate from the through-wall indications in the riser pipe can be estimated using the preceding equation with the value of flow coefficient, C, assumed as 1.0. A key input needed is the crack opening area, A.

The approach used in this evaluation to calculate the value of A, was to calculate a conservative value of crack opening displacement, δ, and assume the crack opening configuration to be like a rectangular slot with one side being the crack length, 2a, and the other side as the crack opening displacement. The opening displacement is calculated using $\delta = 4\sigma l/E$ (Reference EPRI Report NP-2472, Vol. 2, D-2) where l is one half the crack length (allowables calculated in section 6.2), σ is the applied stress, and E is Young's modulus. Calculated crack openings are less than .5 mil. The crack opening area is then simply:

$$A = 2a\,(\delta) \qquad (6)$$

Figure 4 shows leakage rates versus the percentage of the allowable flaw size calculated in Section 6.2.

8. Summary & Conclusions

A flaw evaluation, consisting of stress and fracture mechanics analyses of the Dresden Unit 2 and Unit 3 jet pump circumferential riser welds was conducted to develop a flaw evaluation handbook. The procedures of Paragraph IWB-3640, ASME Section XI, were used as a guide in determining the allowable flaw lengths. End-of-cycle allowable flaw lengths were calculated at three circumferential weld locations. The methodology presented in this report can be used along with consideration of observed IGSCC and evaluation of fatigue crack growth rates to disposition any indications detected during future inspections of the jet pumps at Dresden Unit 2 and Unit 3.

The following table shows a summary of allowable beginning-of-cycle (BOC) flaw lengths for Unit 2 and Unit 3.

Summary of BOC Allowable Circumferential Flaw Sizes (1), (2)

Weld	Flaw Length (inch)	
	Unit 2	Unit 3
1	16.3	16.4
2	15.4	15.3
3	16.9	16.9

(1) Based on 16000 hours of hot operation.
(2) Based on IGSCC evaluation.

9. References

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[2] ASME Boiler and Pressure Vessel Code, Section XI, Rules for In-Service Inspection of Nuclear Power Plant Components, American Society of Mechanical Engineers, 1989 Edition, Paragraph IWB 3640.

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[4] Dresden Unit 2 and 3 Jet Pump Drawing No. 729E434
Dresden Unit 2 and 3 Jet Pump Riser Drawing No. 117C1645
Dresden Unit 2 and 3 Jet Pump Elbow Drawing No. 117C1475

[5] Dresden Unit 2 and 3 Seismic Analysis, GENE-771-84-1194

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[7] Dresden Unit 2 and 3 Reactor Thermal Cycles Drawing No. 921D265

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[11] Zahoor, A., Ductile Fracture Handbook, Prepared for Novetech Corporation and EPRI, EPRI Report Number NP-6301-D.

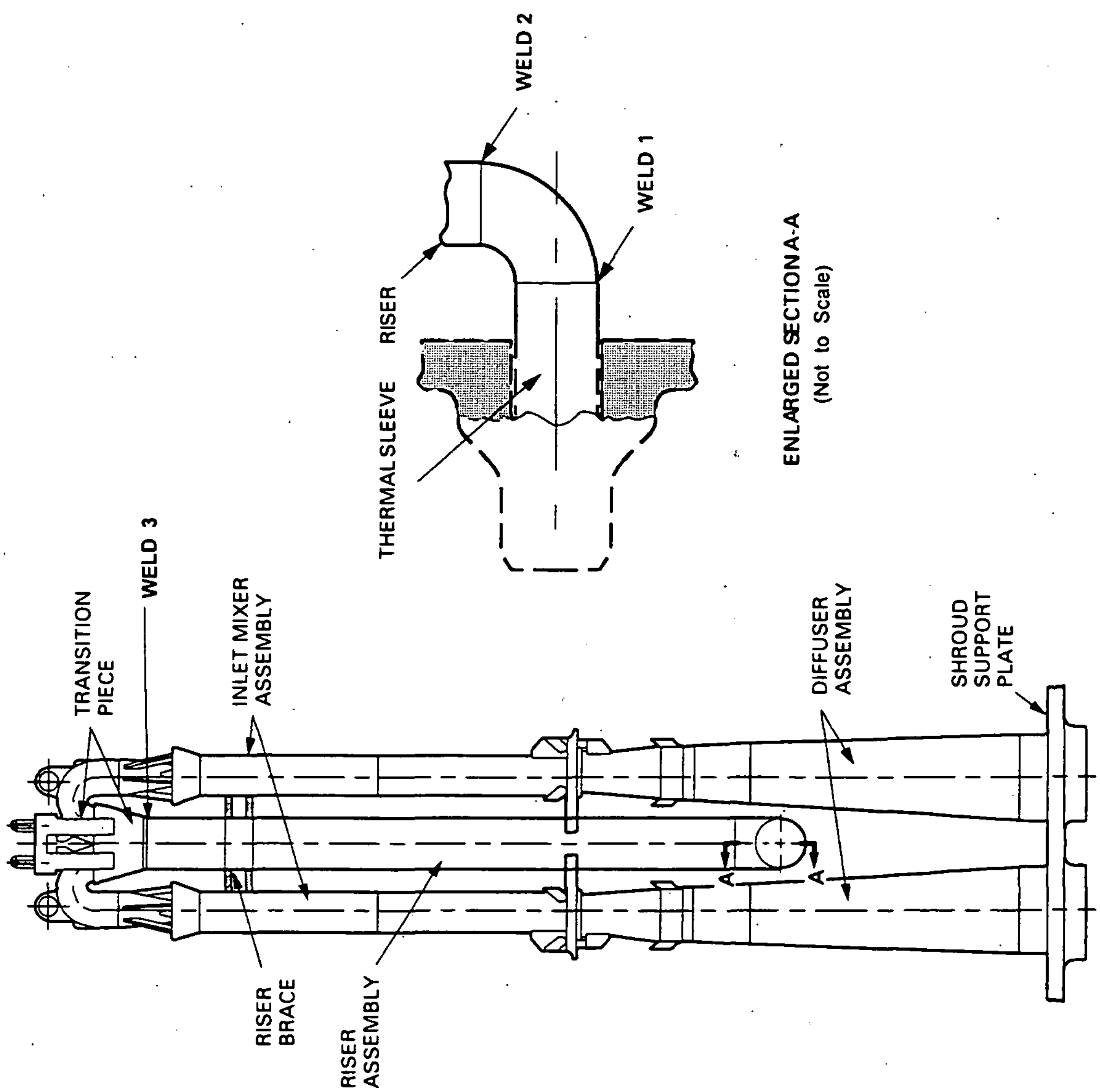

Figure 1. Weld Locations on Dresden Jet Pump

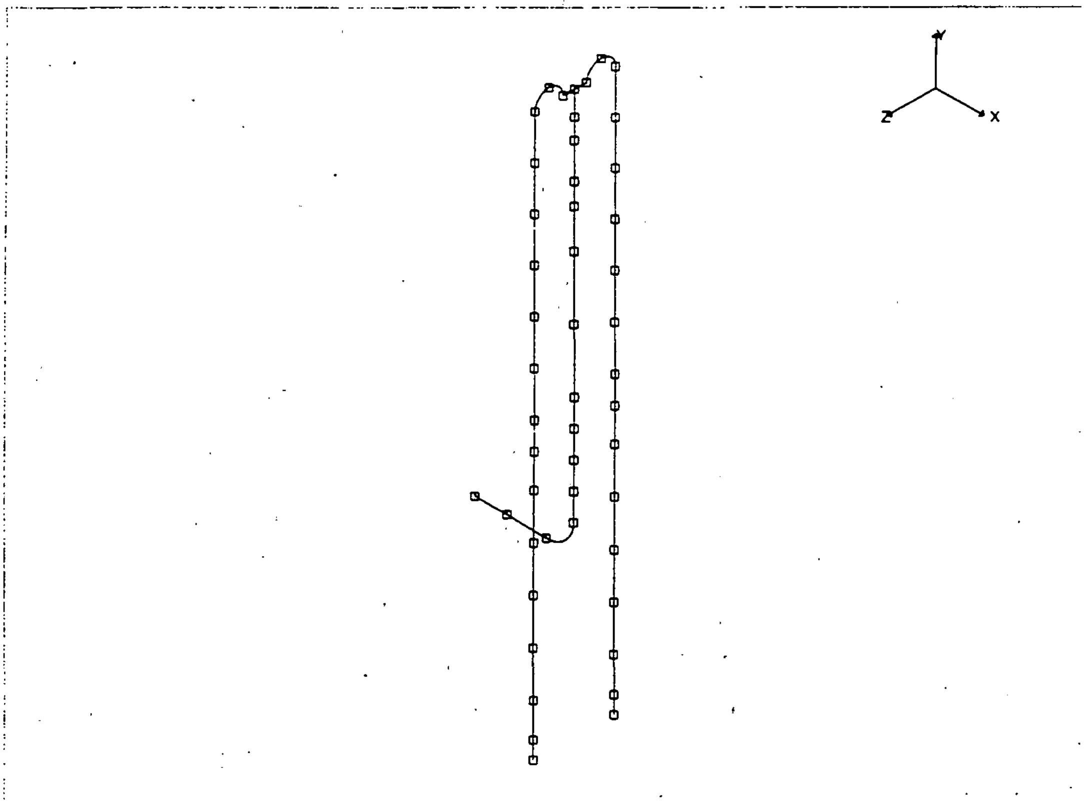

Figure 2. SAP Model of the Dresden Jet Pump

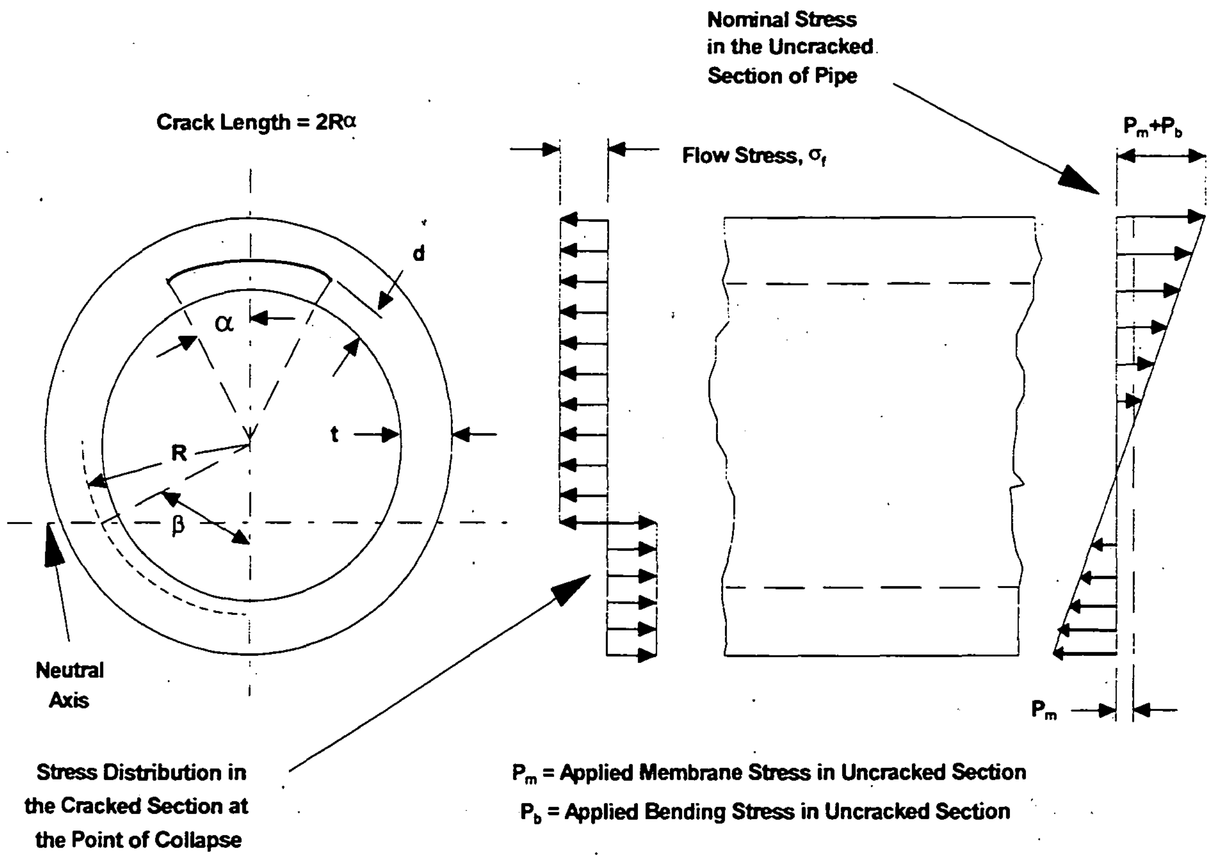

Figure 3. Stress Distribution in a Cracked Pipe at the Point of Collapse

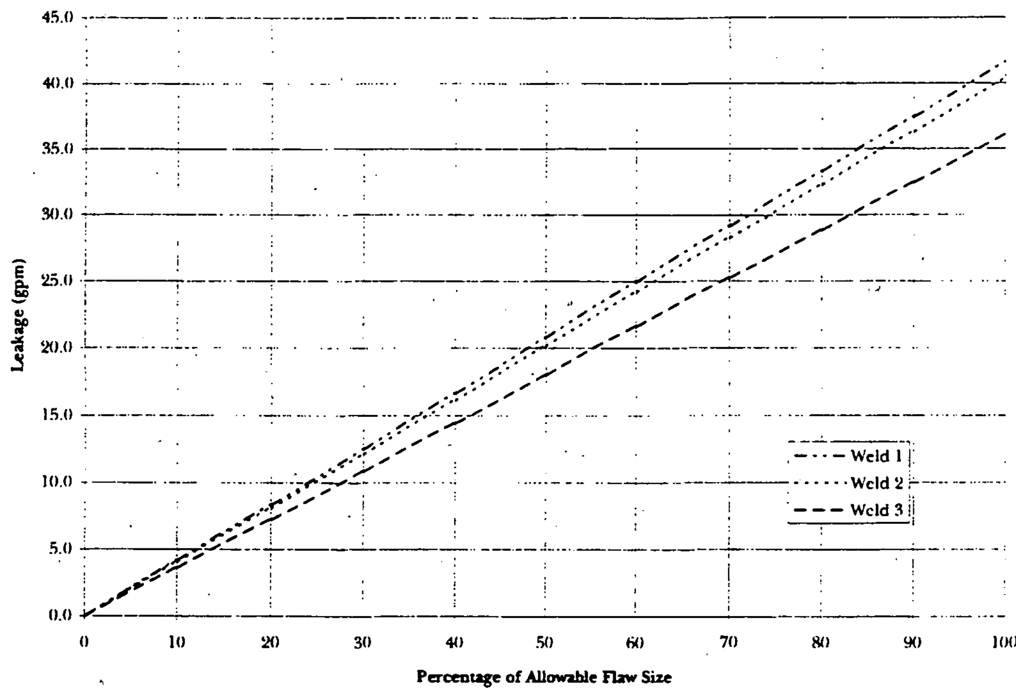

Figure 4. Leakage Versus Percentage of Allowable Flaw Size

Appendix on Susceptibility

10.Susceptibility

10.1. Overview of Susceptibility Factors

Within the jet pump assembly with emphasis on the jet pump riser pipe, there are several general factors that affect susceptibility. The materials/material product form, the water environment, the loading (both static and dynamic), and the fabrication stresses (attributable to manufacture) contribute to the susceptibility. The factors align with the key degradation factors that must be considered when analyzing the jet pump assembly sub-components: intergranular stress corrosion cracking (IGSCC) and fatigue. Each will be discussed separately in the following sections.

10.1.1. Intergranular Stress Corrosion Cracking (IGSCC)

The occurrence of IGSCC relies on the combined presence of an aggressive environment, a susceptible material, and tensile stress. The environment in the annulus region is highly oxidizing in all BWRs. Radiolysis model calculations predict that the environment has a significant concentration of H_2O_2. The inside of the jet pump assembly up to the inlet mixer is exposed to the coolant that has traveled around the recirculation circuit. The outside surfaces of the entire assembly are exposed to the coolant that is present in the annulus downcomer region. The piping below the inlet mixer is exposed to a combination of the two environments. Under normal water chemistry conditions, these environments should display little difference in the oxidant levels and the levels of H_2O_2.

Both the initiation and growth of cracks will be promoted by the high electro-chemical corrosion potential (ECP) which exists in the annulus region. With the introduction of hydrogen into the reactor water, the oxidant can be reduced. The beneficial effect of lowering the ECP will be seen first at the riser piping and the lower portions of the jet pump assembly. This factor is relevant to the assessment of differences between the two Dresden plants and will be discussed in greater detail.

From the material perspective, there are a large range of parameters that control the component's resistance to degradation, particularly from IGSCC. Some of these differences that may be present in the jet pump thermal sleeve/riser are:

- material

- material product form (wrought vs. casting)
- material condition (annealed and welded)
- heat chemistry (composition, e.g. carbon level)
- component form (seamless pipe, rolled vs. welded pipe)
- type of weld/weld design (fillet and groove)
- welding process (shop vs. field)

The susceptibility of materials is interrelated and dependent on the type of material, the actual product form, and the final condition. For the austenitic stainless steels, these factors are very important and play an important role in assessing susceptibility for the different jet pump assembly components. For type 304 stainless steel materials, the product form is critical in determining susceptibility. The cast product forms, similar to the weld metal that is used to manufacture the weld joints, have a high resistance to IGSCC in BWR core environments, due to the duplex microstructure which contains both ferrite and austenite phases. The wrought materials exhibit different susceptibility to IGSCC depending on the composition and is well documented. The high carbon level associated with the riser piping is the key factor in increasing the IGSCC susceptibility.

It is known that irradiation can affect the material properties of wrought stainless steel if the fluence is high. The fluence can lead to both hardening of the material as well as enhancing chromium depletion at the grain boundary locations. For the jet pump components, the use of type 304, which already makes the materials susceptible to chromium depletion, and the low to intermediate fluence range (with most locations below 10^{18} n/cm^2 cumulative fluence), the added contributions from irradiation would be very small.

The final key contributor to the susceptibility are the level of tensile stress present in the component, particularly in the vicinity of welds. The axial residual stress patterns for the small diameter piping has been shown to be near yield level around the pipe circumference. It has been shown that stress sign (i.e., tension or compression) also varies around the circumference, making full circumferential cracking very unlikely. For both field and shop welds these stresses are present and, therefore, both types of welds must be viewed as susceptible. However, for the plants where the jet pumps were installed in the field, the installation specification allowed the use of draw bead straightening to aid in the assembly process. This practice could contribute additional residual stresses to the assembly which in turn could accelerate crack initiation.

10.1.2. Fatigue

Fatigue is the term given to both crack initiation and subcritical crack growth under the influence of fluctuating or cyclic applied stresses. There are three sources of fatigue significant to the BWR: system cycling fatigue, rapid cycling fatigue, and vibration fatigue. System cycling refers to changes in the reactor system which cause variations in pressure and temperature at the component. Examples of system cycling are start-up, shutdown, SCRAM, and safety relief valve (SRV) blowdown. System cycling is generally accounted for in the initial design analysis. Rapid cycling (e.g., thermal mixing) is generally not an issue for jet pump components. This leaves high cycle fatigue due to vibration as the primary fatigue issue for the jet pump components.

Vibration fatigue has been seen in several jet pump assembly components. Two sources of vibration fatigue have been seen: flow-induced vibration (FIV) and pump resonance vibration. FIV is caused when coolant flowing past a component sheds vortices which create cyclic loads. These loads generally occur in a frequency range up to about 20 Hz, leading to the expectation that FIV cycles accumulate early in operation, probably during pre-operation tests. However, it is possible that some modes of FIV are associated with a particular operating mode which occurs infrequently.

Pump resonance vibration is an example of infrequent operating modes eventually causing vibration fatigue failure. As the recirculation pump operates at varying speeds during start-up, shutdown or other non-steady state conditions, the pump vanes send pressure pulses through the coolant at varying frequencies for short periods of time. When these pulses pass through the resonant frequency of a component, the amplitude of the vibration stress can become large, reducing the number of cycles required to cause failure. The amount of time spent at a component's resonant frequency determines whether vibration fatigue will cause failure, and if so, when the failure will occur. This type of fatigue is most relevant to the jet pump assembly.

The vibrational loads can also interact with cracks initiated through IGSCC. Once a crack is initiated by IGSCC, vibration driven fatigue can lead to extension of these cracks. The critical size beyond which fatigue crack growth occurs is dependent on the vibratory loads and the crack length. The critical parameter is the "threshold stress intensity" which is dependent on the material. The rate of growth per cycle is very slow and therefore any applicable growth would only be associated with high frequencies. At these frequencies, there would be no enhancement by the environmental corrosion on the fatigue process.

10.1.3. Embrittlement

The neutron fluence in the annulus region in not large enough to cause irradiation embrittlement of the austenitic jet pump assembly components. Therefore, thermal (aging) embrittlement of duplex casting materials is the only embrittlement mechanism which significantly affects the jet pump assembly.

For the jet pump assembly which is constructed using several cast components, thermal embrittlement is a potential degradation mechanism. Cast stainless steel is a duplex structure consisting of austenite and up to 25% ferrite. Precipitates form in the ferrite phase or at the grain boundaries at certain temperature ranges. Such precipitates are known to form at temperatures as low as 850°F, but there is concern that precipitation may occur at temperatures as low as 480°F over long periods of time. Such precipitation would cause a reduction in toughness. The degree of toughness loss can only be estimated based on current methodologies. However, the operating temperatures of the BWR are low enough that the potential for significant casting degradation exists only after 30 years of operation. It is important to note that thermal embrittlement does not in itself cause cracking to occur. It reduces the structural margin of a material in resisting propagation of cracks due to other initiators like IGSCC or fatigue.

Castings have high resistance to IGSCC. Fatigue initiation of cracks in castings cannot be ruled out, but because of the high stiffness, fatigue initiation due to vibration is unlikely. Therefore, thermal embrittlement mechanisms are of low concern.

Overall, the coolant environment in the jet pump assembly has the potential to promote IGSCC at all locations under normal water chemistry conditions. This is the most important degradation mechanism. The key factor in evaluating susceptibility to IGSCC is the material and its final processed condition. The specific considerations for the jet pump riser piping at the Dresden Nuclear Units will be discussed in the next section for each location.

10.2. Evaluation of Specific Locations in the Jet Pump Riser Piping

10.2.1. Nozzle Thermal Sleeve

Table 1 details the materials used in the construction of the thermal sleeves for the Dresden-2 and Dresden-3 plants. For both the plants, the thermal sleeves, from the original plant construction, are type 304 material.

Table 1. Thermal Sleeve Materials

Plant	Thermal Sleeve Material	Creviced?
Dresden-2	Type 304	Yes
Dresden-3	Type 304	Yes

The susceptibility of the thermal sleeve welds and the thermal sleeve to jet pump riser pipe is dictated by the sleeve material and the riser pipe material as well as the existence of any crevice. For the Dresden BWR/3 design plants, the thermal sleeves are type 304 and there exists a crevice design in the nozzle safe end vicinity. The butt weld locations associated with the thermal sleeve/riser piping are susceptible due just to the use of type 304 material.

10.2.2. Riser Pipe and Restrainer Bracket Fillet Welds

Table 2 details the materials used in the construction of the riser pipe for the Dresden plants. As for most of the plants, the piping is made of type 304 stainless steel. It was of either the rolled and welded (ASTM A-358) or seamless (ASTM A-312) product form. Figure 1 displays the entire jet pump assembly along with the standard restrainer bracket attachment to the riser pipe for these plants. For this configuration, the restrainer bracket is fillet welded over the entire 360° circumference.

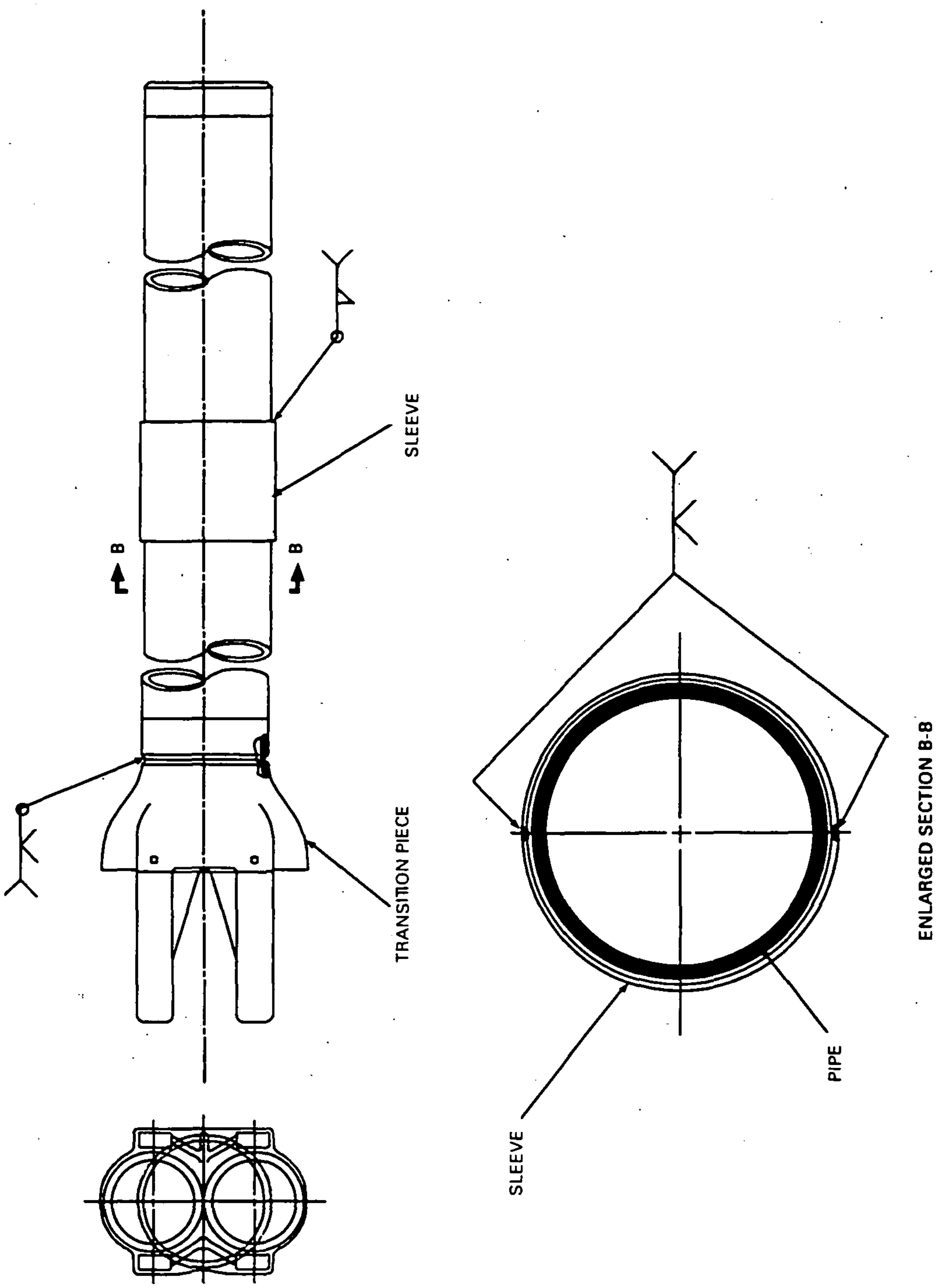

Figure 5. Typical BWR/3 Riser Throat

Table 2. Riser Materials

Plant	Riser Material Type and Pipe Schedule
Dresden-2	10" Sch. 40 Type 304
Dresden-3	10" Sch. 30 Type 304

The type 304 pipe and elbow materials were provided in a solution annealed condition. The specifications require that any longitudinal welds also be solution annealed. As discussed earlier, welding the high carbon type 304 materials lead to susceptibility to IGSCC in the normal water chemistry BWR environment. The higher carbon level is likely to produce sensitization of the material in the heat affected zone as part of the welding process. This sensitization can be produced using shop welding or field welding processes over a range of heat inputs. Therefore, all welds heat affected zones must be considered susceptible. The field weld joint, the joint between the thermal sleeve and the riser pipe elbow, can potentially have additional displacements required in the fit-up of the assembly. Field assembly allowed draw beads, as discussed earlier, which can superimpose additional stresses in the weld region. These additional stresses are expected to be coincident with the locations of the draw bead buildups.

All weld locations, susceptible to IGSCC could also be susceptible to fatigue crack growth if a crack were initiated as discussed in the overview section.

10.2.3. Transition Piece

Table 3 summarizes that the transition piece could be either a cast type 304 or a welded assembly. For those plants with welded pieces, Figure 5 displays the location of the weld attaching the jet pump "capture" arms to the transition body.

Table 3. Transition Piece Materials

Plant	Material	Heat Treatment
Dresden-2, 3	Type 304: casting or welded assy.	Welded assemblies: Solution annealed

For the Dresden plants, the transition could be welded and therefore should be considered to have the same susceptibility as the adjoining riser pipe with Figure 5 displaying potential sites for IGSCC.

10.3. Impact of Plant Operation on Susceptibility

Tables 4 and 5 detail the typical plant operational water chemistry for the 5th, 10th and 14th cycles for Dresden-2 and Dresden-3 respectively. The early years for the two plants are very similar. The water quality was one of high conductivity and high impurities as measured by the median conductivity and chloride contents. In the recent cycles, the water conductivity has greatly improved. In particularly, the concentrations of chloride and sulfate species are also much lower. These concentrations are on the order of 0.3 to 0.6 ppb Cl- and 1.5 to 2.1 ppb SO_4 respectively. These species which are known to affect IGSCC crack growth are at levels where the impact on the crack growth rates of any existing cracks will be positively affected. However, more important are the oxygen concentrations. For the Dresden-2 plant, the impact of hydrogen water chemistry (HWC) can be clearly seen, with the dissolved oxygen (DO) median value (typical of operation) to be ~12 ppb while for the Dresden-3 plant, the median is 242 ppb (typical of normal water chemistry (NWC)). Therefore, while IGSCC may have been promoted by the early years of operation, any future growth will be retarded or stopped by the HWC levels in place at Dresden-2.

10.4. Summary of Susceptibility Assessment for the Riser Pipe Locations.

Due to the age of the two plants and the higher water conductivity during the first several years of operation as well as the use of type 304 materials for the thermal sleeves, the jet pump riser pipe and the riser transition pieces, all welds must be considered susceptible to IGSCC. The plans for inspections to be performed must consider this. The riser elbow locations must be considered to be good indicators of the overall cracking characteristics of the jet pump riser piping welds. These elbow welds are full penetration butt welds made in the field thereby having all of the factors that could promote IGSCC.

It would be expected that for the Dresden-3 plant, the conditions for IGSCC will remain with future operation. The re-inspection planning needs to consider this. However, for the Dresden-2 plant, the implementation of HWC will slow down the initiation of any future cracks and slow down the growth of any existing cracks. For this plant, re-inspection planning can take advantage of these factors.

For the risers, the other potential degrading factors will have much less impact. Irradiation effects will have little additional effect because of the fluence levels and the already present susceptibility due to the use of type 304 material. Thermal embrittlement, relevant only to castings, is expected to have little effect due to the lower operating temperature of the BWR. The effects of fatigue only become important if an IGSCC crack has initiated and has achieved a significant length around the piping circumference.

Table 4. Dresden-2 Typical Water Chemistry: Cycles 5, 10, 14

Dresden 2	**Cycle 5**		**Cycle 10**		**Cycle 14**			
	COND (µS/cm)	CL (ppb)	COND (µS/cm)	CL (ppb)	COND (µS/cm)	DO (ppb)	CL (ppb)	SO_4 (ppb)
Mean	0.324	36.987	0.1595	20.99	0.0875	179.07	0.612	2.105
Median	0.32	34	0.0765	20	0.061	12.8	0.3	1.09
Minimum	0.11	30	0.055	20	0.056	0.02	0.273	0.3
Maximum	0.62	58	1.967	40	1.052	9550	36.7	94

Table 5. Dresden-3 Typical Water Chemistry: Cycles 5, 10, 14

Dresden 3	**Cycle 5**		**Cycle 10**		**Cycle 14**			
	COND (µS/cm)	CL (ppb)	COND (µS/cm)	CL (ppb)	COND (µS/cm)	DO (ppb)	CL (ppb)	SO_4 (ppb)
Mean	0.3525	41.3	0.1345	20	0.078	228.6	0.3361	1.549
Median	0.345	40.5	0.097	20	0.072	242	0.3	1.19
Minimum	0.059	30	0.069	20	0.059	1	0.2	0.3
Maximum	0.89	68	0.69	20	0.198	3940	3.3	13.6