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July 12, 1994

Mr. William T. Russell, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attn: Document Control Desk

Subject: Dresden Nuclear Power Station Units 2 and 3
Quad Cities Nuclear Power Station Units 1 and 2
Response to NRC Request for Information on
Recirculation Pipe Break
NRC Docket Nos. 50-237/249 and 50-254/265

References: (a) Commonwealth Edison (ComEd) teleconference with NRC staff, dated July 7, 1994.

(b) Commonwealth Edison teleconference with NRC staff, dated July 11, 1994.

Dear Mr. Russell:

In the Reference (a) teleconference, the NRC staff requested additional information regarding the probability of a large recirculation pipe break at Dresden and Quad Cities Stations. The response to that question is included as an attachment to this letter.

In the Reference (b) teleconference, the NRC staff requested ComEd document the revised schedule for delivery of results for Dresden and Quad Cities from the TRAC-G 3D model that GE is developing. The final report for Dresden and Quad Cities is currently scheduled to be provided to the NRC staff for your information by August 19, 1994.

To the best of my knowledge and belief, the statements contained in this response are true and correct. In some respects, these statements are not based on my personal knowledge, but obtained information furnished by other Commonwealth Edison employees, contractor employees, and consultants. Such information has been reviewed in accordance with company practice and I believe it to be reliable.

DCD Rec'd 7/26/95 for processing

ZZZZ

	Ltr	Encl
J Stang	1	1
R. Pulsifer	1	1
File Center	1	1
NRC PDR	1	1

Please direct any questions you may have concerning this response to this office.

Sincerely,

Peter L. Piet
Nuclear Licensing Administrator

Attachment: Dresden and Quad Cities Station Safety Assessment for Reactor Recirculation Piping

cc: J. B. Martin, Regional Administrator - RIII
C. Miller, Senior Resident Inspector - Quad Cities
M. Leach, Senior Resident Inspector - Dresden
C. Patel, Project Manager - Quad Cities
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DRESDEN AND QUAD CITIES STATION
SAFETY ASSESSMENT FOR REACTOR RECIRCULATION PIPING

SECTION I. EXECUTIVE SUMMARY

The purpose of this safety assessment is to evaluate the potential for a Double-Ended Guillotine Break (DEGB) of the recirculation system piping. This event is analyzed in the UFSAR. If this event occurs, the core shroud and the recirculation jet pumps maintain a floodable volume up to two-thirds core height. Therefore, the core shroud mitigates the consequences of this Design Basis Accident (DBA).

Although a rupture of the recirculation system piping is a postulated failure, it does not mean that this failure is anticipated. The concept of a deterministic failure of the largest diameter high pressure pipe was originated by the U.S. Atomic Energy Commission. Requiring BWRs to analyze this event assures that the Containment and the Emergency Core Cooling Systems (ECCS) are conservatively sized. The recirculation piping systems have been analyzed using appropriate codes and standards to limit applied stress. Materials were selected to provide adequate ductility and toughness.

The likelihood of a failure of the recirculation piping must be estimated since there is no experience base in BWRs of any recirculation pipe breaks, much less a double-ended guillotine break. In addition, industry experience indicates that high energy pipes develop leaks long before pipe failure occurs.

If a pipe crack develops, the crack will grow to a critical crack length before it ruptures. This concept is referred to as
Leak-Before-Break (LBB). The critical crack length is larger for larger diameter pipes; therefore, the recirculation piping would develop detectable leakage for a significant period of time before experiencing rupture.

A recirculation pipe leak would be detected by the drywell temperature monitors, drywell pressure monitors and drywell floor drain sump monitoring systems. The Technical Specifications require a unit shutdown when unidentified drywell leakage exceeds 5 gallons per minute. The leakage from a recirculation pipe through-wall crack would exceed 250 gallons per minute before approaching the critical crack length.

If a recirculation pipe leak continued after being detected, it would eventually increase the drywell pressure above 2 lbs. At this point, the Reactor Protection System (RPS) would initiate a reactor SCRAM and start all Emergency Core Cooling Systems (ECCS). Operators would follow the normal and Emergency Operating Procedures (EOPs) to assure the reactor is maintained in a safe condition.

The recirculation piping systems at Dresden and Quad Cities were made of a type of stainless steel which is susceptible to InterGranular Stress Corrosion Cracking (IGSCC). IGSCC has been found in all four of these units. Augmented InService Inspection (ISI) Programs have been implemented in accordance with Generic Letter 88-01 to detect and monitor IGSCC.

Cracking which was identified in the Dresden Unit 3 recirculation piping resulted in its replacement in 1987. Since that time, no recirculation pipe cracking has been identified on this Unit. Hydrogen Water Chemistry (HWC) has been implemented at Dresden Unit 2 and Quad Cities Units 1 and 2 to mitigate IGSCC. When cracks are identified in the recirculation piping, the Station implements a weld overlay, stress improvement, or other mitigating actions as appropriate.

It is difficult to determine a definitive recirculation pipe break probability. NUREG/CR-4792 determined this probability to be in the range from 1.0×10^{-12} to 3.82×10^{-12} per reactor year. BWROG-93149 estimated an upper bound frequency of 7.51×10^{-6} per reactor year based on operating experience. This estimate will decrease with each additional year of reactor operation without experiencing recirculation pipe breaks.

ComEd used IDCOR Technical Report 86.3B1 to conservatively estimate failure frequency of the recirculation large bore piping at Dresden and Quad Cities Station. The upper bound for frequency is estimated to be on the order of 1×10^{-5} per reactor-year (Attachment A).

In conclusion, this assessment identifies that the failure of the recirculation piping is very unlikely. If cracks were to occur and grow, they would produce through-wall leakage before pipe rupture. The leak detection systems would identify this leakage in time to safely shut down the reactor prior to piping rupture.

DRESDEN AND QUAD CITIES STATION
SAFETY ASSESSMENT FOR REACTOR RECIRCULATION PIPING

SECTION II: SAFETY ASSESSMENT

A. INDUSTRY EXPERIENCE WITH RECIRCULATION LINE BREAKS

The probability of a small pipe break (2 inches in diameter) is generally recognized to be an order of magnitude greater than a large pipe break.

There has not been a leak or break in any BWR recirculation piping system of any nuclear power plant to date. Therefore, operating experience does not provide data with which a failure rate can be determined. All probabalistic line break frequencies are based on assumptions. As a result, the probabalistic values vary depending on the assumptions used.

B. LIKELIHOOD OF A RECIRCULATION PIPING LEAK

1. Pipe Stress Levels

Although the plant's design basis, as discussed in the Dresden and Quad Cities UFSAR, includes the evaluation of a loss of coolant accident resulting from a postulated recirculation pipe break, considerable effort goes into designing piping and safe end systems to assure that such a break will not occur.

The Dresden Unit 2 and Quad Cities Units 1 and 2 recirculation piping systems were designed and analyzed in accordance with the USAS B31.1.0-1967 "Power piping" code. Dresden Unit 3 recirculation piping was designed and analyzed in accordance with ASME Section III 1980 Edition through summer 1982 addenda. Materials were selected to provide adequate ductility and toughness. These codes also provide implicit margins for material fatigue.

The recirculation piping systems have been seismically analyzed and supported in accordance with code requirements. The piping system fabrication and installation invoked substantial Quality Assurance measures and procedures to assure quality for construction and repair activities.

2. Recirculation Piping Augmented Inspections

The Dresden and Quad Cities reactor recirculation piping is fabricated from austenitic stainless steel. This piping system is susceptible to InterGranular Stress Corrosion Cracking (IGSCC). Because of IGSCC, the Dresden Unit 3 recirculation system piping was replaced.

This piping is the subject of Generic Letter 88-01. The GL 88-01 large bore recirculation piping program scope is summarized in Attachment B.

A summary of the ISI inspection results for Dresden and Quad Cities Stations is provided in Attachment B.

3. IGSCC Mitigation

a. Dresden Unit 3 Recirculation Piping Replacement

IGSCC has been an industry concern for several years. When IGSCC was identified at Dresden Unit 3, it was determined that piping replacement would be the most cost effective mitigating option. The new piping materials were selected for their resistance to IGSCC. The piping was pre-treated prior to installation to further reduce its IGSCC susceptibility.

b. Hydrogen Water Chemistry

The Dresden Unit 2 Hydrogen Addition System was installed in 1983. Hydrogen was added to the feedwater system at an increasing rate until the electrochemical potential was reduced below the point where IGSCC can propagate.

The Dresden Hydrogen Addition System design was enhanced, then installed at Quad Cities Units 1 and 2.

Hydrogen Water Chemistry can mitigate IGSCC, particularly for the recirculation suction piping which sees the highest concentration of hydrogen in the reactor vessel.

C. EFFECTS OF A RECIRCULATION PIPE LEAK

1. Leak Detection Systems

The recirculation piping is completely contained within the drywell. A through wall crack in this piping system would result in saturated steam and water at 1000 psig exhausting to the drywell. The drywell leak detection systems are capable of detecting a leak from a through wall crack before the crack grows to a critical length.

These monitoring systems are equipped with safety related, environmentally qualified sensors that alarm in the control room.
Drywell Pressure Sensors:

The drywell is maintained at a pressure of approximately 1 psig. The Reactor Protection System (RPS) initiates a reactor SCRAM if drywell pressure exceeds 2 psig (2 1/2 psig at Quad Cities). Drywell pressure is very sensitive to steam leaks. Drywell pressure would be an early indication of a small steam leak.

Drywell Floor Drain Sump:

Leakage into the floor drain sumps is termed unidentified leakage and is monitored in the control room. If unidentified leakage increases it would be investigated immediately. If unidentified leakage reaches 5 gallons per minute, the Technical Specifications require a reactor shutdown.

An analysis of critical crack lengths (Reference 1) identified that a 16 inch diameter pipe would develop a crack of 23.1 inches before rupture occurs. This crack would result in a leak rate of 262 gpm at operating reactor pressure. The recirculation piping of concern is 28 inches in diameter. The larger piping diameter results in longer critical crack length and greater leakage prior to rupture.

Drywell Temperature Monitoring:

A steam leak in the drywell will increase drywell temperatures. There are several temperature sensors in the drywell that have individual alarm set points. These sensors alarm in the main control room.

D. Analysis of a Recirculation Pipe Crack Growth

1. Leak-Before-Break

The Leak-Before-Break (LBB) concept is based on the fact that reactor piping is fabricated from tough, ductile materials which can tolerate large through-wall cracks without fracture under service loadings. By monitoring the leak rate from through-wall cracks, and setting conservative limits on acceptable leakage, cracks in piping can be detected well before the margin to rupture is challenged.

In NUREG 1061, Volume 3, the NRC Piping Review Committee outlined the limitations and general technical guidance on LBB analysis to justify mechanistically that breaks in high energy fluid system piping need not be postulated. In a recent modification to General Design Criterion 4, the NRC has formalized the use of the LBB approach to justify the elimination of pipe whip restraint and jet impingement barriers as design requirements for hypothetical Double Ended Guillotine Break (DEGB) in high energy reactor piping systems.

A key parameter in the LBB evaluation is the critical crack length at which pipe rupture is predicted. The focus in the LBB evaluation is on the through-wall circumferential cracks because they could eventually lead to a DEGB.

Leakage from a through-wall crack with a length approaching the critical crack length, would be large enough to be readily detected. Thus, isolation can be achieved well before the crack grows to critical length and well below design basis flows and pressures are established.

Critical crack length and leak rate calculations for typical BWR piping geometries have been documented (Reference 1). The calculations use methods described in References 2, 3, and 4.

Table 1 lists the values of parameters used in the critical crack length and leak rate calculations. The results of the calculations of representative pipe sizes are summarized in Table 2. A limit load approach with a conservative value of flow stress equal to 2.4 Sm (where Sm is the value of the material design stress intensity given in the ASME code), was used in calculating critical crack lengths. When based on test data, the flow stress for four inch diameter pipes was assumed to be 2.7 Sm. The leak rate calculation methods used for both water and steam lines are outlined in Reference 3. Table 3 list the line sixes for the recirculation pump suction and discharge piping.

The calculated leak rate at critical crack length is a strong function of pipe diameter. Nevertheless, even for the four inch diameter water lines, the predicted leak rate is 25 gpm at close to the critical crack length. Therefore, the recirculation piping is expected to develop a detectable leak prior to reaching the point of incipient rupture. Thus, a DEGB in these lines is highly unlikely.

TABLE 1 **VALUES OF PARAMETERS USED IN CRITICAL CRACK LENGTH AND LEAK RATE CALCULATIONS** **(Reference 7)**	
Pipe Thickness	Schedule 80
Pipe Internal Pressure	1050 psi
Temperature	528 F
Normal Operation Bending Stresses	4 ksi
Material	Stainless Steel

TABLE 2

CRITICAL CRACK LENGTHS AND LEAK RATES FOR VARIOUS DIAMETER PIPES
(Reference 7)

Pipe Diameter (in.)	Critical Crack (in.)	Leak Rate at Critical Crack Length (gpm)	
		Water	Steam
4	7.1	25	15
6	9.8	41	27
12	18.5	166	108
16	23.1	262	170

TABLE 3

SYSTEM PIPE SIZES FOR DRESDEN and QUAD CITIES

System	Nominal Pipe Diameter
Recirculation Suction	28 inches
Recirculation Discharge	28 inches
Ring Headers	22 inches

2. Effects of a Line Break

UFSAR Section 15.6.5.1 identifies the Recirculation Suction Line break as a limiting fault ie., and event that is not expected to occur but is postulated because the consequences may result in the release of significant amounts of radioactive material.

A line break of the recirculation piping will immediately produce low reactor water level and high containment pressure. This will cause a reactor SCRAM and automatically initiate ECCS. This line break will also create high drywell radiation and drywell temperatures. All of these parameters are entry conditions to the Emergency Operating Procedures (EOPs).

The UFSAR assumes loss of off-site power and an additional single active failure occurs at the same time as the Recirculation suction line break.

The coolant lost through the rupture is condensed by the pressure suppression pool, thus reducing primary containment pressure. Energy is removed from the pressure suppression pool by the Containment Cooling System.

During the early phase of the LOCA depressurization transient, core cooling is provided by the existing coolant inventory. In the latter stage of system depressurization and after depressurization has been achieved, the ECCS provides core cooling and supplies liquid to refill the lower portion of the reactor vessel and reflood the Core. The reflood process provides sufficient heat removal to terminate the core temperature transient.

Emergency operating procedures direct the operators to verify that all control rods are inserted and the required automatic actions have occurred.

E. CONCLUSIONS

In conclusion, this assessment identifies that the failure of the recirculation piping is very unlikely. If cracks were to occur and grow, they would produce through-wall leakage before pipe rupture. The leak detection systems would identify this leakage in time to safely shut down the reactor prior to piping rupture.

F. REFERENCES

Reference 1: GESSAR II, 238 Nuclear Island, Section 5.2.5, GE Document No. 22A7007, Rev. 0.

Reference 2: S. Ranganath and H.S. Mehta, "Engineering Methods for Assessment of Ductile Fracture Margin in Nuclear Power Plant Piping," ASTM STP 803, 1983, pp. II-309 to II-330.

Reference 3: A. Zahoor, R.M. Gamble, H.S. Mehta, S. Yukawa and S. Ranganath, "Evaluation of Flaws in Carbon Steel Piping: Appendices A and B", EPRI Report No. NP-4824, October 1986.

Reference 4: H.S. Mehta, "Determination of Crack Leakage Rates in BWRs", Attachment 2 in Letter dated April 22, 1985, from Jack Fox, Chairman, ANS-58.2 Working Group to K. Wichman of NRC.

Reference 5: Title 10 Code of Federal Regulations Part 50 Appendix A General Design Criteria

Reference 6: Dresden Units 2 and 3 Updated Final Safety Analysis

Reference 7: Report of the U.S. Nuclear Regulatory Commission Piping Review Committee, NUREG-1061,Volumes 1 through 5, 1984.

Reference 8: Federal Register, Volume 52, p. 41288, Final Rule Modifying General Design Criterion 4 in 10CFR50 Appendix A.

Reference 9: C.W. Schroeder letter to T. Kovach (CWS LTR #92-585), Dresden Station Response to Generic Letter 92-04, Resolution of the Issues Related to Reactor Vessel Water Level Instrumentation in Boiling Water Reactors Pursuant to 10CFR 50.54(F), dated September 28, 1992.

ATTACHMENT A

Subject: Estimate of Large Recirculation Pipe Break Frequencies for Dresden 2&3 and Quad Cities 1&2

References:

1. Delian Corporation, "IDCOR Technical Report 86.3B1, Individual Plant Evaluation Methodology for Boiling Water Reactors - Volume I," April 1987.

2. BWR Owners Group report "Pipe Break Probabilities in Boiling Water Reactors," November 1993, as transmitted to the USNRC via letter BWROG-93149, "Response to NRC Request for Information on Pipe Break Frequencies," dated December 8, 1993.

As requested, the NETS PRA Group has estimated the large recirculation pipe break frequencies. A large break could potentially impose large loads on the core shroud.

Piping Rupture Rates

The BWR IPE methodology (Ref. 1) based on WASH-1400 specifies a rupture failure rate of 8.6E-11 per hour-section and 1.5E-10 per hour-component for piping and components. These values were used in the Dresden and Quad Cities IPEs for piping and components "subject to intense scrutiny." These values are applicable to the recirculation system because of station implementation of the ISI program, leak detection, and other special measures. The values are discussed in Ref. 1 as applying to ruptures rather than external leaks.

Reference 2 is a recent industry review of the pipe break frequencies and includes the following clarification of the meaning of a pipe section:

> "A 'pipe section' is defined as a segment of piping, between major discontinuities such as valves, pumps, reducers, tees, etc. This definition is taken from WASH-1400. A pipe section is typically 10 to 100 feet long, and contains from four to eight welds. Each section can also contain several elbows and flanges. Instrumentation connections are not considered 'major discontinuities'."

Piping Considered

The ComEd IPEs for these units use the large LOCA frequency estimate from WASH-1400, with a large LOCA defined as a pipe 6" or larger. The current concern, however, is only with the very large piping in the recirculation systems. Therefore, the analysis below considers the 28" suction piping and discharge piping up to the ring header. Smaller attached piping such as the 22" ring headers and smaller risers and LPCI/SDC/RHR lines were excluded.

The ring headers were excluded because of their smaller size and location. A ring header break would have an area approximately 40% less than a pump discharge or suction line break. Also, due to the close proximity of the ring headers to the jet pumps and the distance from the recirculation suction nozzles, potential loads on the core shroud would be greatly reduced. This would be due to the distribution of flow from ten jet pumps and the reduced flow out the suction nozzle due to pressure losses through the piping, pump, and valves.

For Dresden 2&3 and Quad Cities 1&2, the "major discontinuities" for the large recirculation piping are as follows:

Recirculation Pumps (2 per unit)

Recirculation Pump Suction and Discharge Valves (4 per unit)

LPCI/SDC/RHR Tees (4 per unit)

Pump Discharge Line/Recirculation Ring Header Tees (2 per unit)

A total of 12 large recirculation pipe sections per unit is obtained by using these major discontinuities. For the total recirculation pipe break frequency estimate below, 6 components per unit were used; only the pumps and valves were considered as "components," consistent with the approach used in the Dresden and Quad Cities IPEs.

Total Recirculation Pipe Break Frequency Estimate

For Dresden 2&3 and Quad Cities 1&2:

Large Recirculation Pipe Break Frequency = (12 sections x 8.6E-11/hr-section + 6 components x 1.5E-10/hr-component) x 8760 hrs/yr

= 1.7E-5 per reactor-year.

As discussed above, this estimate uses the methodology employed in the Dresden and Quad Cities IPEs, summaries of which were submitted to the NRC in 1993. The BWR IPE methodology (Ref. 1) was developed in the 1980s and based its rupture rates for piping and components on the WASH-1400 study completed in the mid-1970s. As a consequence, this methodology does not take credit for industry experience of no large recirculation pipe breaks in the approximately 20 years since the WASH-1400 study.

Comparison with Recent Industry Estimates

Reference 2 gives bounding estimates for a 6" break (or larger) LOCA in BWR recirculation piping based, in part, on recent studies sponsored by the NRC. Reference 2 gives an "upper bound" value of 7.51E-6 per reactor-year based on the NUREG/CR-4407 statistical approach updated by EPRI (using the operational experience of no BWR recirculation system pipe breaks through September 1993). Reference 2 discusses concerns with a double-ended guillotine break (DEGB) and Intergranular Stress Corrosion Cracking (IGSCC) and includes the following conclusion:

> "The actual large break frequency for BWR recirculation system piping is most likely substantially (several orders of magnitude or more) below the upper bound calculated based on currently available operational experience. Application of the analytically derived relative probabilities of small and large breaks to the experience based probability of a small break would result in an estimated large break frequency several orders of magnitude lower than 7.51E-6. The NUREG/CR-4792 analysis calculated large pipe DEGB frequencies on the order of 1E-12 per reactor year, exclusive of IGSCC. With the effective actions taken by the industry to mitigate IGSCC, the actual large break failure rate lies at some intermediate point between 7.51E-6 and 1.0E-12 per reactor year."

This industry conclusion is judged to be applicable to the Dresden and Quad Cities units due to ComEd implementation of IGSCC mitigating actions. Although the estimate of 1.7E-5 per reactor-year for the Dresden and Quad Cities units is above the upper bound given in Reference 2, the ComEd IPE methodology is conservative and does not take credit for recent favorable industry experience with BWR recirculation systems.

Recirculation Suction Pipe Break Frequency Estimate

In addition to considering large pipe breaks anywhere in the large recirculation pipe sections, a specific review focused on the suction line between recirculation suction nozzle at the reactor vessel and the pump suction valve. Structural integrity of the core shroud is necessary to maintain two-thirds core height coverage upon reflood following a unisolatable break in this suction line or a rupture of a recirculation pump suction valve (structural integrity of the core shroud would be maintained if no gross failure occurs; minor leakage due to through-wall shroud cracking would not constitute gross failure). Considering both loops, the suction line upstream of the valve consists of 4 pipe sections and 2 components (the valves) per unit.

Structural integrity of the core shroud is not necessary to maintain two-thirds core height coverage upon reflood for a recirculation pipe break downstream of the pump suction valve, provided that the suction or discharge valves would succeed in isolating the break from the recirculation suction nozzle. Continued operability of a recirculation pump's suction and discharge valve should a pipe break occur nearby is a complex question that is beyond the scope of this review; therefore, the NETS PRA Group has not estimated the success probability for the pump suction or discharge valves.

For Dresden 2&3 and Quad Cities 1&2:

Recirculation Suction Pipe Break Frequency = (4 sections x 8.6E-11/hr-section + 2 components x 1.5E-10/hr-component) x 8760 hrs/yr

= 5.6E-6 per reactor-year.

Conclusion

The large recirculation pipe break frequency estimate using the ComEd IPE methodology for BWRs is 1.7E-5 per reactor-year. This methodology, used in the Dresden and Quad Cities IPEs, also gives a recirculation suction pipe break frequency estimate of 5.6E-6 per reactor-year.

A recent BWROG analysis (Ref. 2), based on studies subsequent to the development of the ComEd IPE methodology for BWRs, indicates that these rupture rate estimates are conservative. The BWROG report uses operational experience through late 1993 and, due to ComEd implementation of IGSCC mitigating actions, is judged to be applicable to the Dresden and Quad Cities units. The BWROG report gives 7.51E-6 per reactor-year as an upper bound for a 6" (or larger) recirculation pipe break frequency for BWRs.

Based on this review, an upper bound for the frequency of large recirculation pipe breaks for the Dresden and Quad Cities units is estimated to be on the order of 1E-5 per reactor-year.

ATTACHMENT B

DRESDEN UNITS 2 AND 3
QUAD CITIES UNITS 1 AND 2
REACTOR RECIRCULATION PIPING

Summary of welds which include all full penetration girth welds 22" or larger in diameter which make up the recirculation system. IGSCC categories are as defined in NUREG 0313 Rev. 2.

DRESDEN

FOR DRESDEN UNIT 2 IGSCC CATEGORIES:

A = NO EXAMINATIONS TO BE PERFORMED IN NEXT 10 YEARS
C = 13 WELDS TO BE EXAMINED WITHIN THE NEXT 10 YEARS
D = 24 WELDS EXAMINED EVERY OTHER OUTAGE (WHICH WILL BE R14, R16, R18, R20, . . . ETC.)
E = 4 WELDS SAME AS ABOVE (D)
F = 2 WELDS EXAMINED EVERY OUTAGE
G = 1 WELD WILL RECEIVE A WELD OVERLAY D2R14 AND THEN BE PLACED INTO CATEGORY E

FOR DRESDEN UNIT 3 IGSCC CATEGORY:

ALL WELDS ARE CATEGORY A. WELDS TO BE EXAMINED ARE CHOSEN STRICTLY BY ASME SECTION XI REQUIREMENTS. THERE WILL BE 9 WELDS SELECTED THIS INTERVAL AND INSPECTIONS WILL BE COMPLETED BY 2/28/02.

THERE WERE 5 WELDS EXAMINED DURING D3R13.

QUAD CITIES

QUAD CITIES UNIT 1 IGSCC CATEGORIES:

A = NO EXAMINATIONS TO BE PERFORMED IN THE NEXT 10 YEARS
C = 36 WELDS TO BE EXAMINED WITHIN THE NEXT 10 YEARS
D = 1 WELD EXAMINED EVERY OTHER OUTAGE (WHICH WILL BE R14, R16, R18, R20, . . . ETC.)
E = 5 WELDS SAME AS ABOVE (D)

THERE WERE NO WELDS EXAMINED FOR IGSCC DURING Q1R13.

QUAD CITIES UNIT 2 IGSCC CATEGORIES:

A = NO EXAMINATIONS TO BE PERFORMED IN THE NEXT 10 YEARS
C = 28 WELDS TO BE EXAMINED WITHIN THE NEXT 10 YEARS
D = 0 WELDS EXAMINED EVERY OTHER OUTAGE (WHICH WILL BE R14, R16, R18, R20, . . . ETC.)
E = 13 WELDS SAME AS ABOVE (D)

DRESDEN UNIT 2 WELD LOCATIONS/CATEGORIES

"A" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 2-202-4A:

C = 5, D = 2 TOTAL: 7 WELDS

FROM OUTBOARD WELD OF M.O.V. 2-202-4A TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 1 D = 2 TOTAL: 4 WELDS

FROM 28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

D = 5, E = 1, F = 1 TOTAL: 7 WELDS

22" RING HEADER WELDS:

C = 1, D = 3, E = 1 TOTAL: 5 WELDS

"B" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 2-202-4B:

C = 2, D = 3 TOTAL: 5 WELDS

FROM OUTBOARD WELD OF M.O.V. 2-202-4B TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

D = 2, F = 1, G = 1 TOTAL: 4 WELDS

FROM 28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

C = 4, D = 3 TOTAL: 7 WELDS

22" RING HEADER WELDS:

C = 1, D = 3, E = 1 TOTAL: 5 WELDS

DRESDEN UNIT 3 WELD LOCATIONS/CATEGORIES

"A" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 3-202-4A:

A = 4 (1 INSPECTION LEFT THIS INTERVAL) TOTAL: 4 WELDS

FROM OUTBOARD WELD OF M.O.V. 3-202-4A TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 3 TOTAL: 3 WELDS

28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

A = 5 (NO WELDS INSPECTED, NONE PLANNED THIS INTERVAL)
TOTAL: 5 WELDS

22" RING HEADER WELDS:

A = 2 (NO WELDS INSPECTED, 1 PLANNED THIS INTERVAL)
TOTAL: 2 WELDS

"B" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 3-202-4B:

A = 4 TOTAL: 4 WELDS

FROM OUTBOARD WELD OF M.O.V. 3-202-4B TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 3 TOTAL: 3 WELDS

FROM 28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

A = 5 (NO WELDS INSPECTED, 1 PLANNED THIS INTERVAL)
TOTAL: 5 WELDS

22" RING HEADER:

A = 2 (NO WELDS INSPECTED, 1 PLANNED THIS INTERVAL)
TOTAL: 2 WELDS

QUAD CITIES UNIT 1 WELD LOCATIONS/CATEGORIES

"A" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 1-202-4A:

A = 1, C = 6 TOTAL: 7 WELDS

FROM OUTBOARD WELD OF M.O.V. 1-202-4A TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 1, C = 3 TOTAL: 4 WELDS

FROM 28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

C = 6, D = 1 TOTAL: 7 WELDS

22" RING HEADER:

C = 5 TOTAL: 5 WELDS

"B" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 1-202-4B:

A = 1, C = 4, E = 1 TOTAL: 6 WELDS

FROM OUTBOARD WELD OF M.O.V. 1-202-4B TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 1, C = 3, E = 1 TOTAL: 5 WELDS

28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

C = 7, TOTAL: 7 WELDS

22" RING HEADER:

C = 2, E = 3 TOTAL: 5 WELDS

QUAD CITIES UNIT 2 WELD LOCATIONS/CATEGORIES

"A" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 2-202-4A:

A = 1, C = 5, E = 1 TOTAL: 7 WELDS

FROM OUTBOARD WELD OF M.O.V. 2-202-4A TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 1, E = 3 TOTAL: 4 WELDS

28" PUMP DISCHARGE TO 22" RING HEADER CROSS:

C = 5, E = 2 TOTAL: 7 WELDS

22" RING HEADER:

C = 4, A = 1 TOTAL: 5 WELDS

"B" RECIRC LOOP

FROM NOZZLE-TO-SAFE END WELD TO INBOARD WELD OF M.O.V. 2-202-4B:

A = 1, C = 3, E = 2 TOTAL: 6 WELDS

FROM OUTBOARD WELD OF M.O.V. 2-202-4B TO RECIRC PUMP/90 DEGREE ELBOW SUCTION SIDE:

A = 1, C = 2, E = 2 TOTAL: 5 WELDS

28" PUMP DISCHARGE TO 22" RING HEADER:

C = 5, E = 2 TOTAL: 7 WELDS

22" RING HEADER:

C = 4, E = 1 TOTAL: 5 WELDS