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January 29, 1990
JPN-90-012

U.S. Nuclear Regulatory Commission
Attn: Document Control Desk
Mail Stop P1-137
Washington, D.C. 20555

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
IGSCC Inspection Plans for 1990 Refuel Outage

References:

1. NRC Generic Letter 88-01, dated January 25, 1988, which transmitted NUREG-0313 Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

2. NYPA letter, J. C. Brons to NRC, dated July 26, 1989 (JPN-89-053), provided response to request for additional information on Generic Letter 88-01 response.

Dear Sirs:

In Reference 2, the Authority committed to prepare and submit a pre-outage Intergranular Stress Corrosion Cracking (IGSCC) inspection plan for the upcoming 1990 refueling outage. A copy of this plan is included as Attachment 1.

As permitted by Generic Letter 88-01 (Reference 1), the plan takes credit for the FitzPatrick plant's Hydrogen Water Chemistry Program and reduces the number of weld inspections required. This reduction in weld inspections will apply only to inspections scheduled for the 1990 refueling, as guidelines for inservice inspection credit under hydrogen water chemistry have not been established. Occupational radiation exposures are expected to be cut by approximately 16 person-rem as a direct result of the reduced number of welds requiring inspection.

Attachment 2 summarizes the Authority's experiences with Hydrogen Water Chemistry during the first cycle of operation.

Attachment 3 provides additional information on the 1988 weld inspections and supplements prior Authority reports. Details on both the shrinkage stress analysis for weld overlays and as-built data for overlays performed during the 1988 outage are included in this attachment.

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Since Generic Letter 88-01 and Revision 2 of NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," no longer requires pre-outage inspection plans, the Authority will not routinely submit pre-outage plans in the future.

Should you or your staff have any questions regarding this matter, please contact Ms. Sofia Toth of my staff.

Very truly yours,

John C. Brons
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ATTACHMENT 1

A total of thirty-one welds will be inspected during the 1990 refueling outage. The Authority had originally scheduled a total of 41 welds for inspection. Due to the implementation of a Hydrogen Water Chemistry Program and the modified inspection frequency allowed by NUREG-0313 Revision 2, Paragraph 2.3.1, the number of welds to be inspected has been reduced by 10. Occupational radiation exposures are expected to be cut by approximately 16 person rems as a direct result of the reduced number of welds.

Although only one of the four welds with IGSCC crack indications requires inspection under the guidelines of Generic Letter 88-01, three of the four welds will be inspected. The two extra weld inspections included as part of the plan will provide additional data on the effectiveness of the FitzPatrick plant's Hydrogen Water Chemistry Program.

Table 1-1 details the summary of welds per category that are susceptible to Intergranular Stress Corrosion Cracking (IGSCC) at the FitzPatrick plant and establishes the basis for inspecting 31 welds during the 1990 refueling outage.

It is noteworthy that 92 welds out of a weld population of 151 were inspected in the 1988 refueling outage. During the 1989 maintenance outage, as detailed in Reference 11, two welds that had IGSCC were inspected as part of a mid-cycle inspection required by References 6, 7, and 9. These inspections revealed no significant IGSCC growth.

Table 1-1 IGSCC Inspection Summary

NUREG Category	No. of Welds in Category	No. of Welds Scheduled for Inspection	No. of Welds using Modified HWC Inspect. Frequency	Notes
A	26	2	2	Only one weld inspection is required
C	63	12	6	Note 1
C-3	3	0	0	Note 1
C*	2	0	0	Note 1
D	32	16	13	Note 2
E	20	7	5	Note 3
F	2	1	2	Note 4
G	3	3	3	
	151	41	31	

A - Long seams are included in Category A, but are not noted as such. The long seam welds are inspected when the circumferential weld is inspected.

C - All Category C welds were inspected in 1987 or 1988. The welds are scheduled on a basis of 100% in 10 years.

C-3 - Welds with high stress. IHSI has been performed and the three welds were inspected in the 1988 refueling outage. See Reference 10 for the Authority's position on these welds.

C* - Welds with Resistance Heating Stress Improvement performed as noted in Reference 5. Both welds were inspected in the 1988 refueling outage.

E - Total overlaid welds in Category E are eighteen. The other two welds contain IGSCC and have been IHSI treated

NOTES TO TABLE 1-1

NOTE 1 - All welds in this category have been mitigated by Inductive Heating Stress Improvement (IHSI). Hydrogen water chemistry is also an added mitigation factor in reducing susceptibility to IGSCC. The Authority requests that the weld inspection frequency be decreased by a factor of two for this refueling outage as allowed by NUREG-0313 Rev. 2, paragraph 2.3.1.

This will decrease weld inspections from 12 to 6 welds. All welds in this category have been inspected in either 1987 or 1988 (56 welds in 1988; 8 welds in 1987).

NOTE 2 - This category includes 12 Reactor Recirculation System safe end to nozzle welds which contain Inconel 182 weld metal and are currently protected by hydrogen water chemistry. Six of these welds are scheduled for inspection during this outage. The Authority requests that the weld inspection frequency be decreased by a factor of two for this refueling outage as allowed by NUREG-0313, Rev. 2, paragraph 2.3.1.

Two other nozzle to safe end welds, Core Spray "A" and Jet Pump "B," will be inspected during this outage. Thus five safe end to nozzle welds will be inspected. All these welds were inspected in 1987 or 1988 with no IGSCC noted. Granting of this relief by the NRC would result in an inspection decrease from 8 to 5 nozzle to safe end welds. Significant ALARA savings are anticipated as the vessel nozzle inspections are labor intensive due to the vessel bio-shield doors and insulation system located on the nozzles.

In addition to the 5 nozzle to safe end welds, the Authority will inspect 8 more Category D welds. A total of 13 Category D welds in all will be inspected.

NOTE 3 - Category E welds include welds that have been weld overlaid (18 total) and welds with IGSCC that have been IHSI treated (2 total) and comply with NUREG-0313 Rev. 2, paragraph 4.5.

As detailed in Reference 12, three weld overlays on the Core Spray "B" loop were surface finished and inspected during the 1989 fall maintenance outage. In order to consolidate the Core Spray System overlay inspections on a refueling outage basis due to ALARA considerations, the Authority plans to inspect these welds on an alternating refueling outage basis, i. e. 1991, 1995 etc., starting with the 1991 refueling outage.

Currently, NUREG-0313 Rev 2, requires inspection of weld overlays at a frequency of 100% in two refueling outages with 50% required in the first refueling outage after installation. The remaining 15 weld overlays were inspected in the 1987 refueling outage or after installation during the 1988 refueling outage. Thirteen of the weld overlays are on the Reactor Recirculation System and are mitigated by hydrogen water chemistry. The two remaining Reactor Recirculation System overlays are on the jet pump instrument nozzle assembly. A hydrogen water chemistry injection system was installed in September 1989 to protect the jet pump instrument nozzle assembly welds.

The two welds in this category (12-4 and 28-33) that contain IGSCC were inspected during the 1988 refueling outage. One was re-inspected during the 1989 maintenance outage with no significant crack growth noted. These welds do not require inspection, according to NUREG-0313 Rev. 2, until the 1991 refueling outage.

The Authority proposes that the inspection cycle of weld overlays be decreased by a factor of two for the 15 weld overlays on the Reactor Recirculation System and the jet pump instrument nozzle assembly based on the mitigating effects of hydrogen water chemistry. The Authority will inspect the two cracked welds as discussed above to demonstrate the mitigating effects of hydrogen water chemistry even though these welds do not require inspection. This will also allow the Authority to implement an inspection plan for the 15 weld overlays on the Recirculation System on a schedule of 3/4/4/4 - 15 total on a refueling outage basis.

This requires relief from the Category E requirement of inspection of weld overlays from 100% in two outages to 100% in four outages. The weld overlay metal, 308L with ferrite > 7.5FN, is extremely resistant to IGSCC. The compressive residual stress of the weld overlay process has also been shown to stop crack growth in laboratory tests. Finally, the hydrogen water chemistry effect provides added protection from IGSCC.

NOTE 4 - This category contains two welds (28-53, 28-112). Both welds were inspected during the 1988 refueling outage. Weld 28-112 was inspected during the 1989 fall maintenance outage. This weld was determined not to contain IGSCC by two independent examiners (EBASCO and NYPA). The Authority will continue to classify this weld as Category F until the next scheduled inspection (1991 refueling outage). If it is determined that no IGSCC exists at this time, this weld may be upgraded to Category C.

Only one weld requires inspection during this outage, but if the HWC relief is approved as detailed in the previous notes, both welds will be inspected.

ATTACHMENT 2

During Cycle 9, the Crack Arrest Verification (CAV) system was operated in both normal water chemistry (NWC) and hydrogen water chemistry (HWC) almost continuously . Cycle 9 began December 21, 1988 and is scheduled to end in March 1990. A planned two week maintenance outage was held starting September 15, 1989.

Hydrogen water chemistry operated from January 5, 1989 until the September 15, 1989 shutdown. In addition, General Electric's Zinc Injection Process (GEZIP) also began operation on January 12, 1989 with no observed effects on electrochemical potential (ECP) or crack growth rates. The only effect of GEZIP on HWC was the suppression of chromate which causes conductivity spiking during the shutdown of the hydrogen addition system.

The CAV system was operated from December 21, 1988 until July 23, 1989 when a solenoid valve, on the supply of the recirculation sample to the CAV, failed and the system isolated. During this period (July 24, 1989 - September 15, 1989), chemistry data such as conductivity, dissolved hydrogen, dissolved oxygen, pH, ECP and crack growth on the recirculation sample line could not be obtained. For the remainder of the cycle, HWC was controlled by monitoring main steam line radiation and Reactor Water Cleanup (RWC) dissolved oxygen.

A review of the HWC data collected shows compliance with the EPRI "HWC Guidelines" for achievable values of ECP and conductivity > 80% of the time for the entire reporting period from December 21, 1988 - September 15, 1989. The Authority plans to maintain at least 80% Hydrogen Water Chemistry when the plant is above 80% power from the time of the September 1989 maintenance outage to the refueling outage planned in March 1990.

Chemistry Results

Figures 2-1 through 2-4 show plant power and the chemistry of the recirculation system compared with the EPRI "HWC Guidelines" for conductivity, Cl and SO_4 for the reporting period of January 1989 to September 1989. These parameters comply with the EPRI Achievable values of > 90%. Some transients are indicated and were attributed to condenser leaks which were found and repaired during several power reductions or the 1989 fall maintenance outage. The power reductions are indicated on Figure 2-1. Chlorides and sulfates paralleled the conductivity transients, but the corresponding ECP and crack growth data showed no significant changes. Conductivity transients during the first few months with no corresponding Cl or SO_4 changes are attributed to chromate spikes from lowering or isolating the hydrogen addition system.

Chlorides and sulfates are normally measured from the RWC System. The RWC measurement is a combination of recirculation and bottom head drain lines. One would expect the bottom plenum area to be more oxidizing and hence have a higher dissolved oxygen concentration than the Recirculation System. A comparison of Figures 2-5 and 2-6 demonstrate this point. Figure 2-5 is the recirculation dissolved oxygen which ranged from 0.5 ppb to 1.0 ppb. Transients occur during periods of high makeup flow or placement of new condensate demineralizer beds in service. In comparison, Figure 2-6 shows the RWC inlet dissolved oxygen which typically operated at 1.5 ppb before the CAV isolated and 2.0 ppb after.

Figure 2-7 plots the average of the four main steam line radiation monitors (MSLRMs). When this value is > 1300 mr/hr, the ECP was generally < -230 mV SHE. After the CAV isolated (July 23, 1989), there was approximately a 10 day period when the MSLRM's were slightly below this value. During this period, the power supply to the hydrogen flow control system was causing a false indication of H_2 flow transients. There was a concern that if the H_2 flow spike indications were true, the potential existed for a plant scram due to high steam line radiation. To prevent a possible plant scram, hydrogen flow was limited. Additionally, there were some hydrogen flow measurement problems limiting the effective amount of hydrogen in the two feedwater trains. Even during this period, the estimated ECP ranged from -200 mV to -230 mV. Other transients below -100 mV SHE were caused by plant power reductions but did not affect crack growth.

Crack Growth Results

Table 2-1 shows the specimens in the CAV autoclave and relevant parameters.

Table 2-2 shows the crack growth rates at several time periods using linear regression for best fit lines with a 95% confidence level. As shown on the table, there is about a 10-fold reduction in crack growth rates from NWC (region 1) to HWC.

Table 2-3 again shows the overall crack growth of the CAV specimens. Total crack extension not including CAV flow interruption periods is given.

Figure 2-8 shows the crack growth of the 304 SS specimen. The step-like graph is an effect of unloading and loading the CAV during system shutdowns as shown in Figures 2-9 and 2-10. Figures 2-9 and 2-10 are 8 month plots of CAV inlet flow rate and temperature. The cycling seen by the CAV is not representative of the load cycle on plant components. Figure 2-11 is the crack growth rate of the Alloy 182 which is more sensitive to flow, temperature, and loading/unloading variations.

ECP Measurements

The recirculation ECP and conductivity are the key parameters used in controlling the HWC environments. The CAV includes the following electrodes: 304 stainless steel, Alloy 182, Alloy 600, copper-copper oxide (Cu-Cu2O) and platinum (Pt). The 304 stainless steel ECP versus the Pt electrode is used to control ECP for the EPRI HWC Guidelines, but can not be used during periods when the recirculation dissolved oxygen is > 50 ppb because it defaults to -999mV. Data from the Cu-Cu2O electrode is used for the plots and statistics shown in this attachment, as it is accurate in a broader range of operating modes.

Figure 2-12 shows the 304 stainless steel ECP values for the entire 7 month report period. As reflected in Table 2-4 and Figure 2-11, operators routinely lowered the hydrogen flow during the first three months for ALARA reasons during surveillance in the feedwater heater and condenser bays causing several transients ranging from -230 mV to -50 mV. Subsequently, this practice was modified, and fewer transients occurred in the remainder of the report period. ECP transients of +100 mV to +200 mV were caused by shutdowns of the Hydrogen Addition System. Despite all the transients, ECP was maintained in compliance with the EPRI HWC Guidelines as shown in Table 2-4.

TABLE 2-1

FITZPATRICK CAVS TEST SPECIMENS

Specimen Number	Material	Initial Stress Intensity Level (ksi in)
SS-122	Type 304 Stainless Steel, Sensitized	24
INC-71	Alloy 182	28.5
INC-73	Alloy 600	27.5

TABLE 2-2

CRACK GROWTH DATA

Region	304 SS Crack Growth Rate (mil/yr)	Alloy 182 Crack Growth Rate (mil/yr)	Environment
1	28.6 ± 4.3	76.2 ± 2.8	NWC
2	2.1 ± 0.5	12.9 ± 1.5	HWC
3	2.0 ± 0.4	6.4 ± 0.4	HWC
4	1.5 ± 0.7	4.7 ± 1.3	HWC
5	0 ± 0.9	7.9 ± 1.3	HWC
6	1.8 ± 0.4	0 ± 2.4	HWC
7	0.9 ± 0.3	8.5 ± 0.8	HWC
8	3.0 ± 0.4	10.8 ± 2.3	HWC
9	--------------	11.4 ± 0.5	HWC

Table 2-3

FITZPATRICK CAV PERFORMANCE

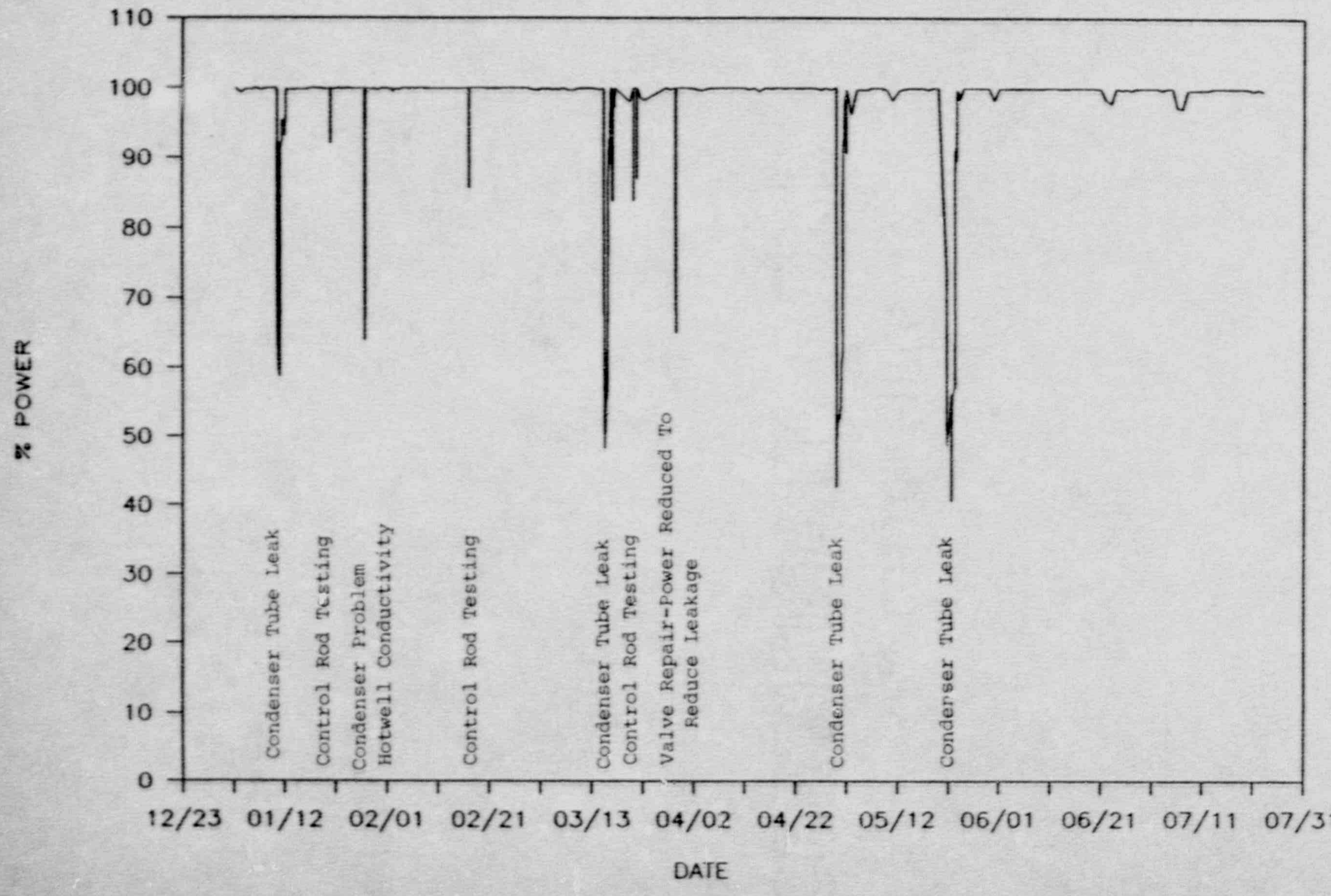
Material	Crack Extension Under HWC (mils)	Annualized Crack Extension Under HWC (mil/yr)	EPRI Guideline Value (mil/yr)
304 SS	0.7	1.5	5
Alloy 182	2.6	8.5	---

Table 2-4

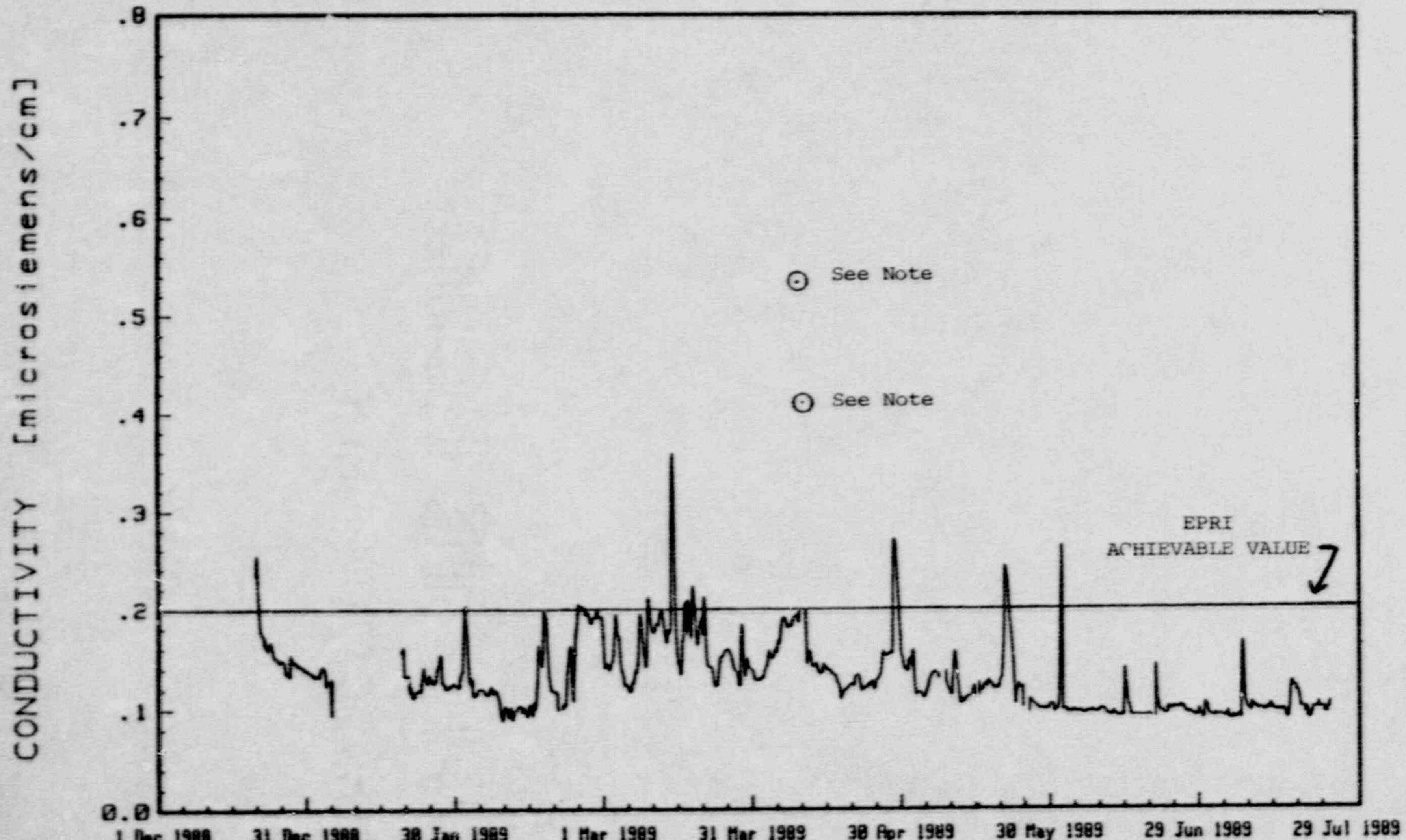
PLANT OPERATION UNDER HWC CONDITIONS

Month	Action Level 1 % < -230 mV SHE	Achievable % < -250 mV SHE
January	77.2	66.7
February	92.4	88.5
March	89.6	83.4
April	95.5	95.3
May	92.5	92.0
June	100.0	100.0
July	92.5	85.4
Jan.-July	91.4	87.3

JAFNPP
FIGURE 2-1
REACTOR POWER

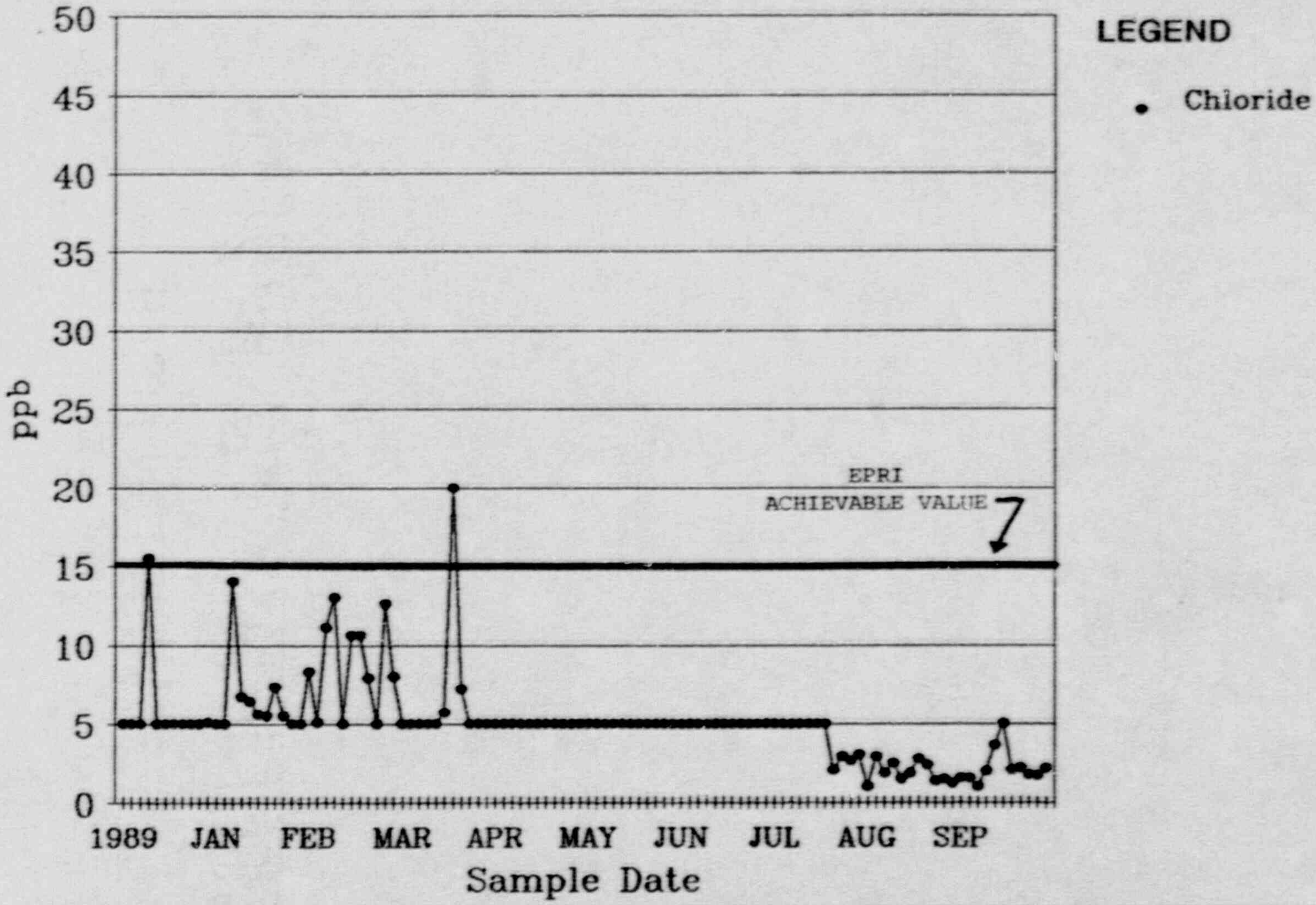


JAFNPP
FIGURE 2-2
REACTOR WATER CONDUCTIVITY



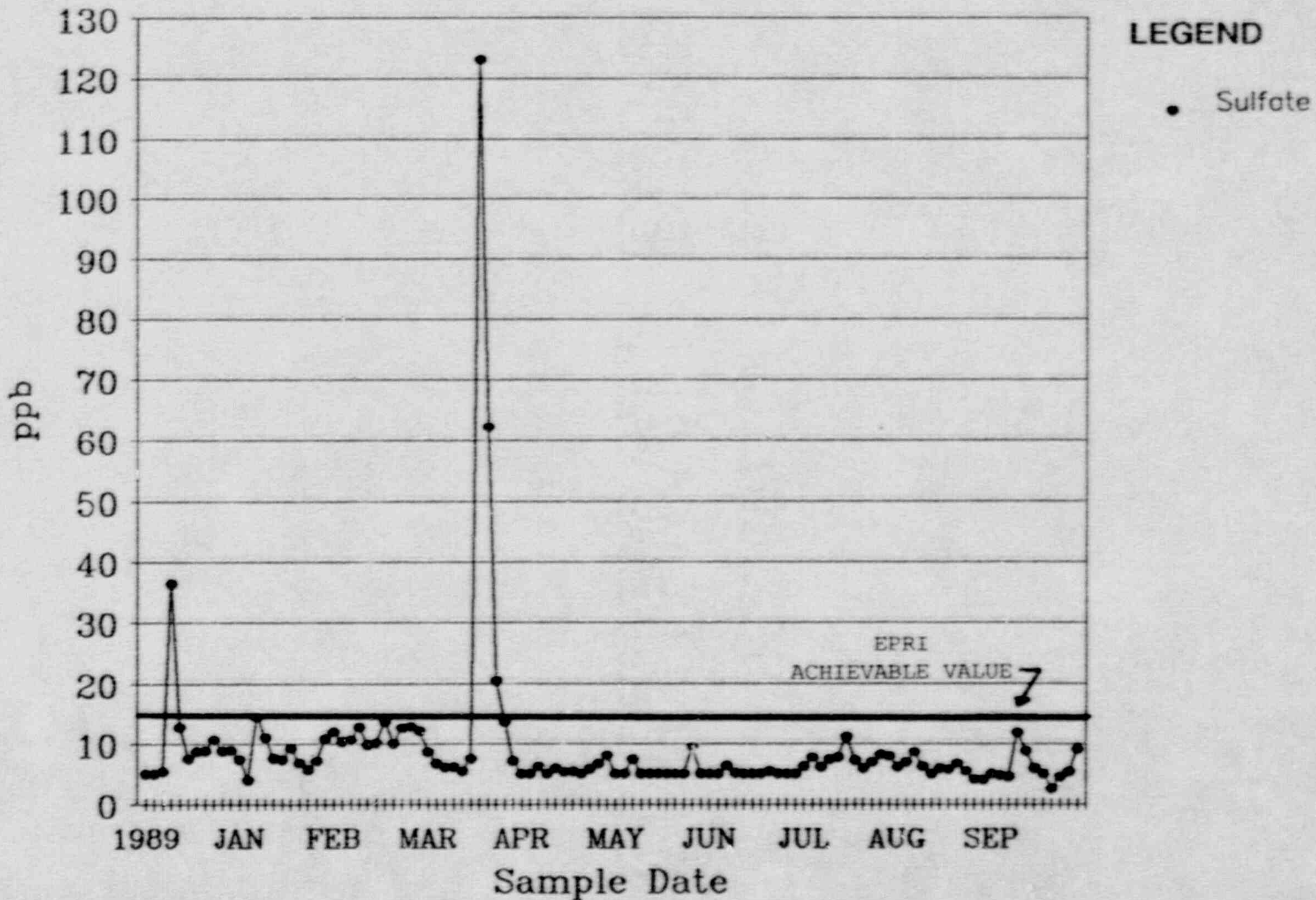
Note: Grab Samples were taken due to CAV shutdown.

JAFNPP
FIGURE 2-3
REACTOR WATER CHEMISTRY - CHLORIDE

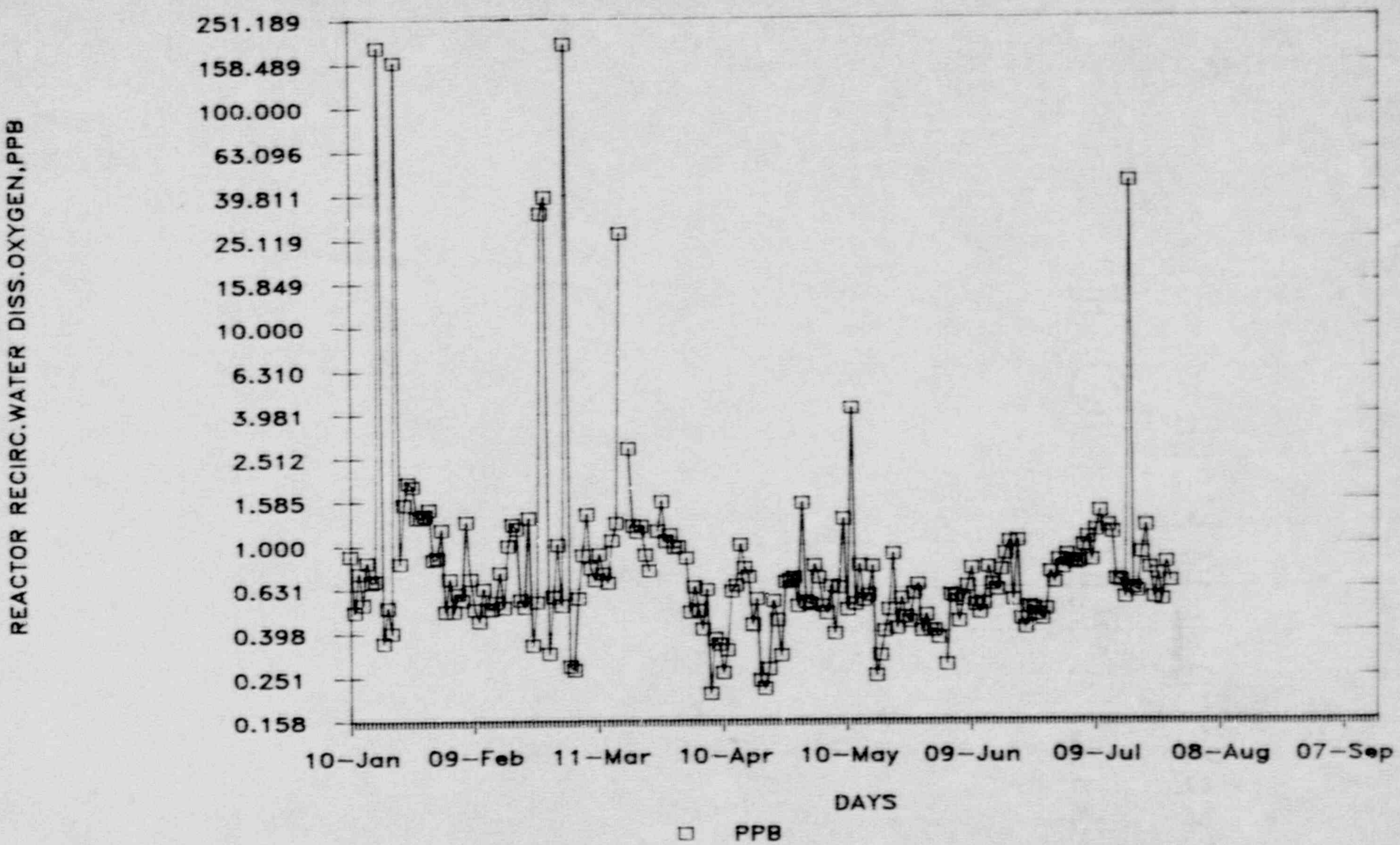


Note: 5 ppb was lowest level of detection through July, 1989.

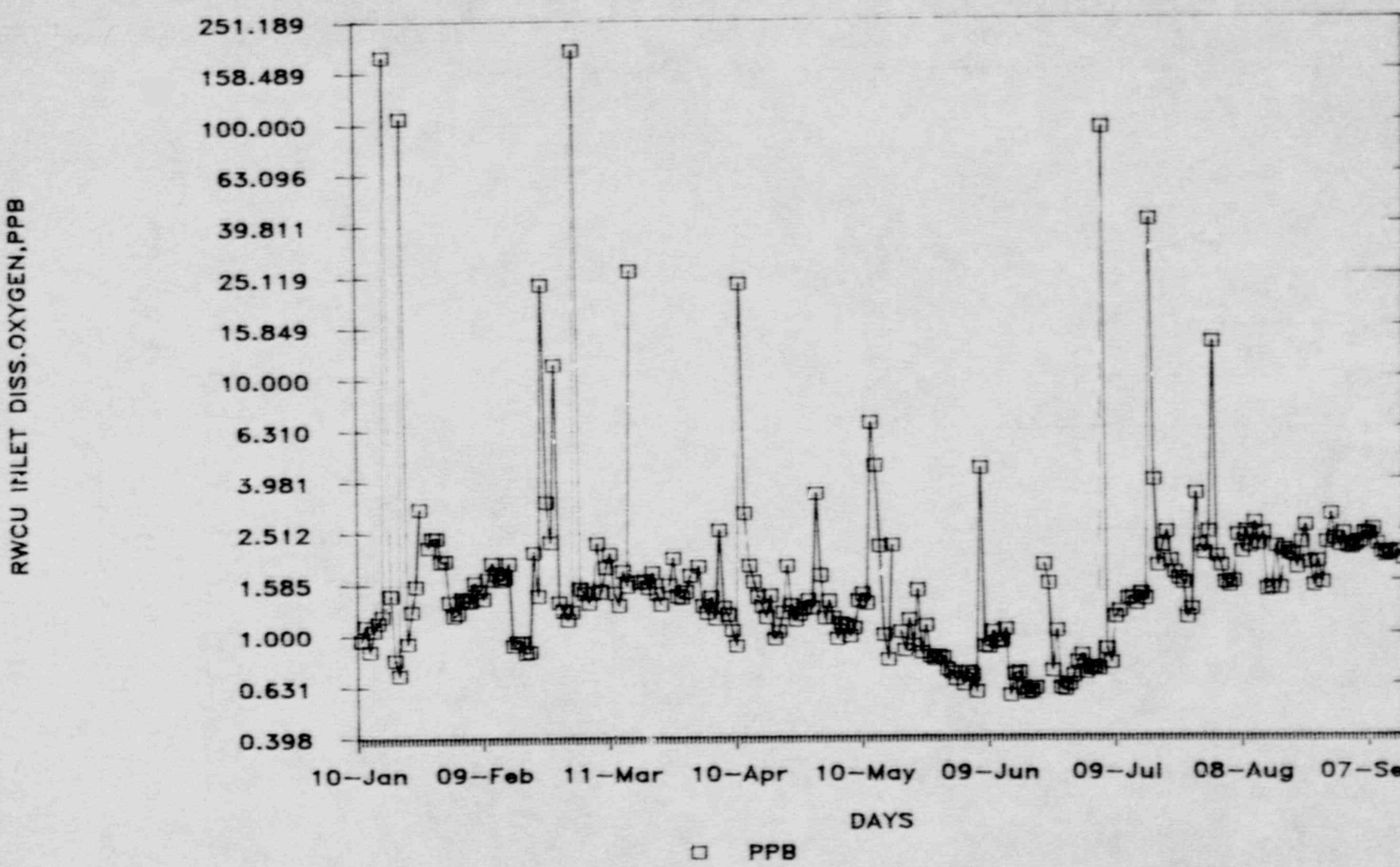
JAFNPP
FIGURE 2-4
REACTOR WATER CHEMISTRY - SULFATE



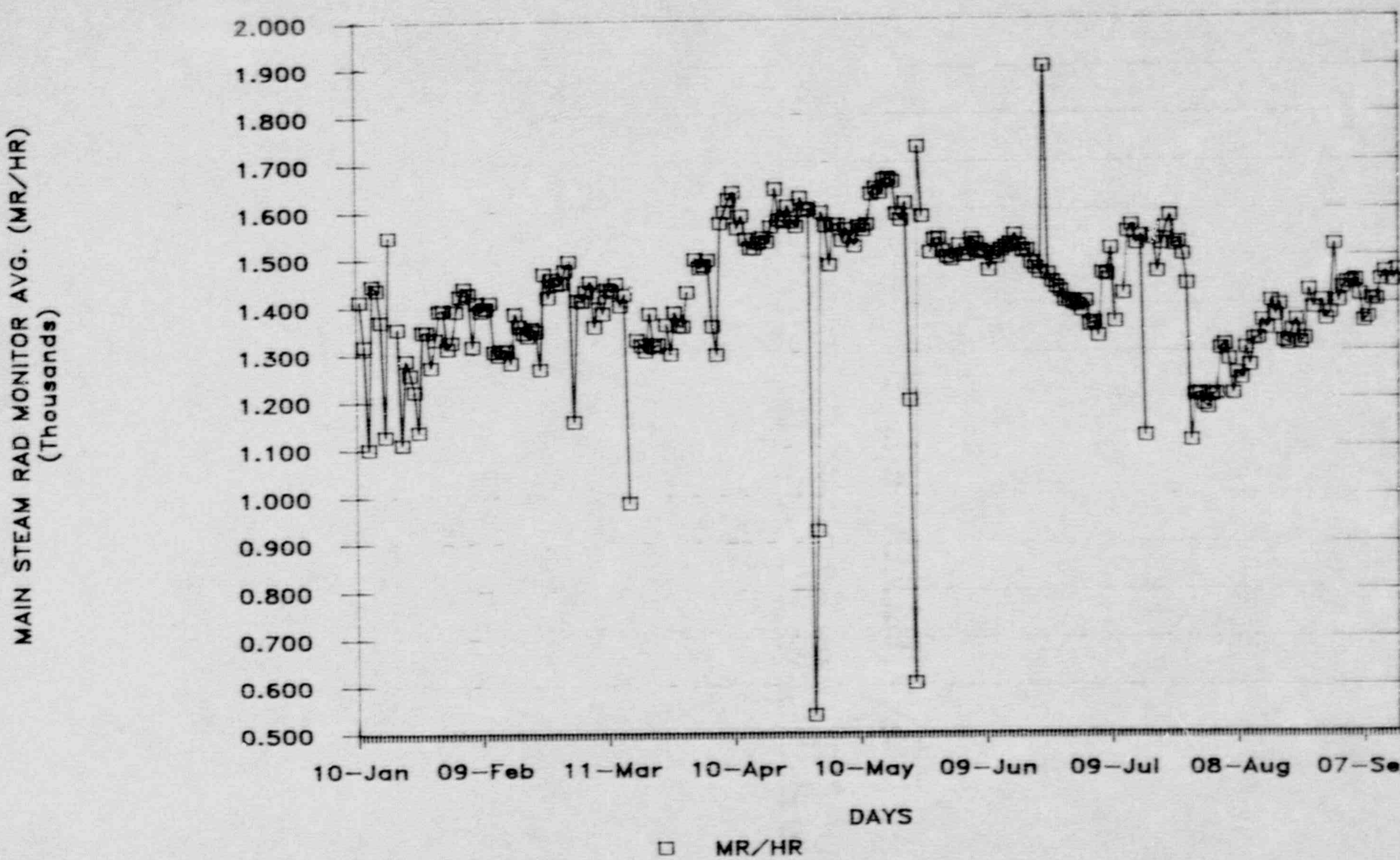
JAFNPP
FIGURE 2-5
RECIRCULATION WATER DISSOLVED OXYGEN



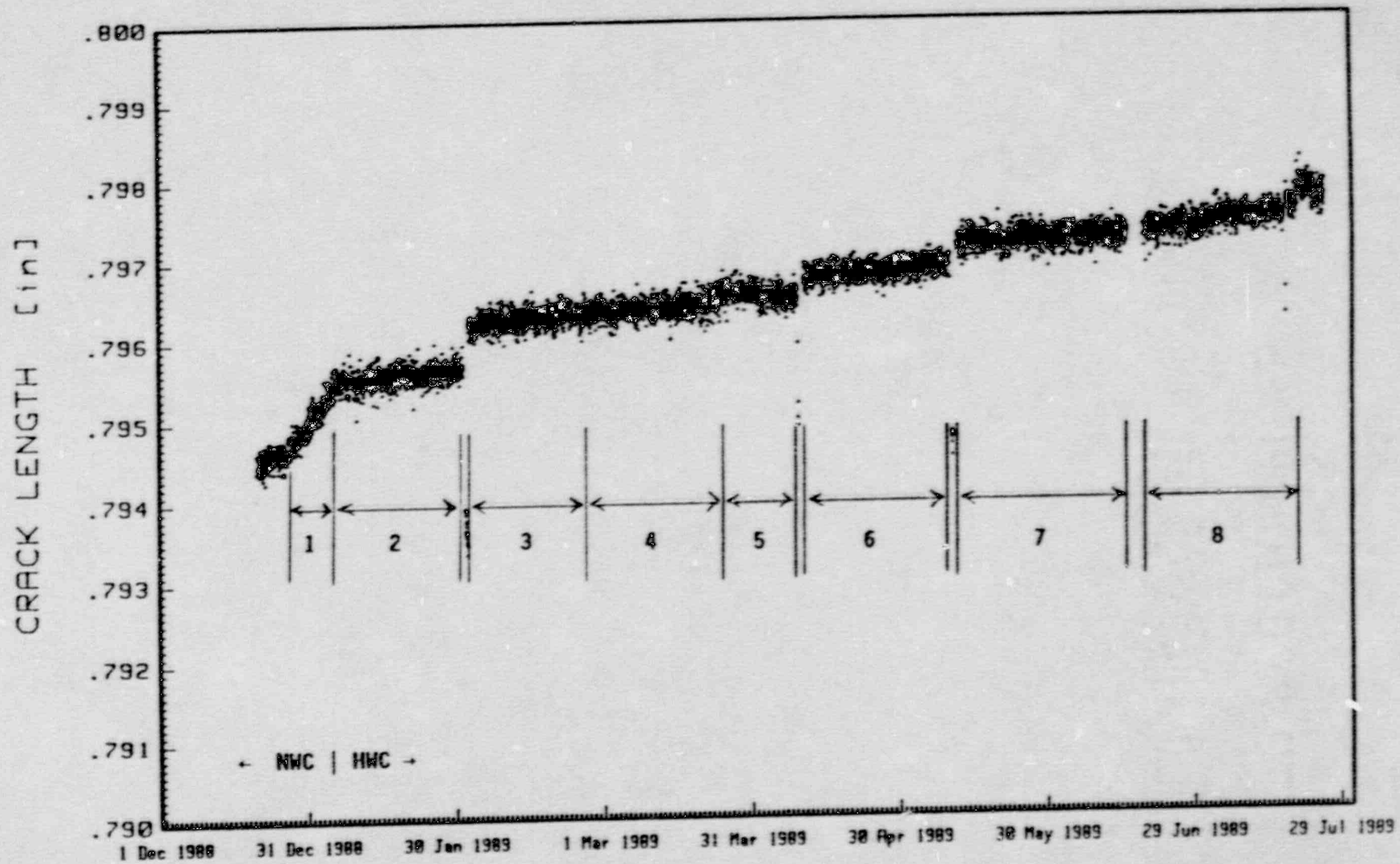
JAFNPP
FIGURE 2-6
RWCU INLET DISSOLVED OXYGEN



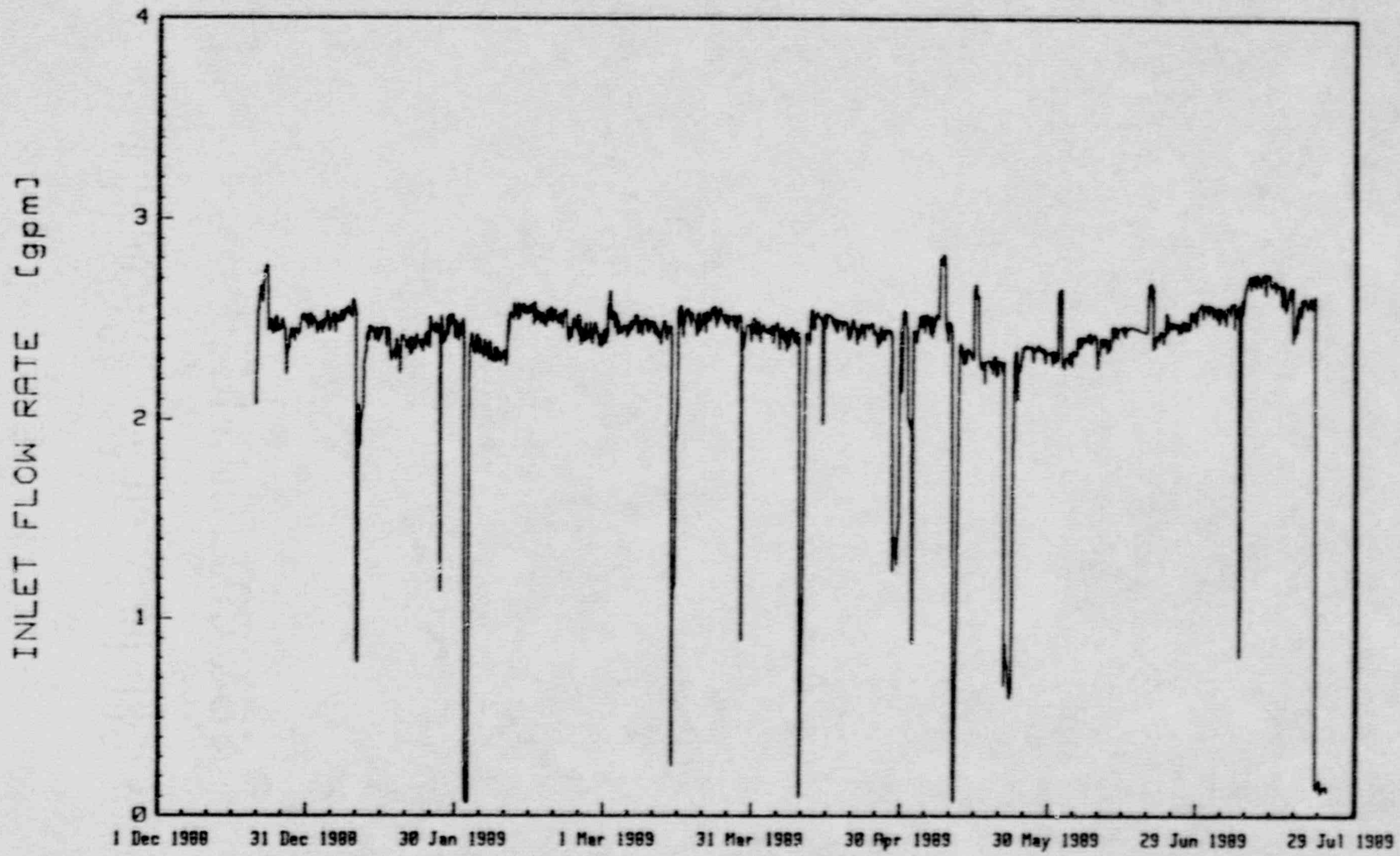
JAFNPP
FIGURE 2-7
MAIN STEAM RAD MONITOR AVERAGE



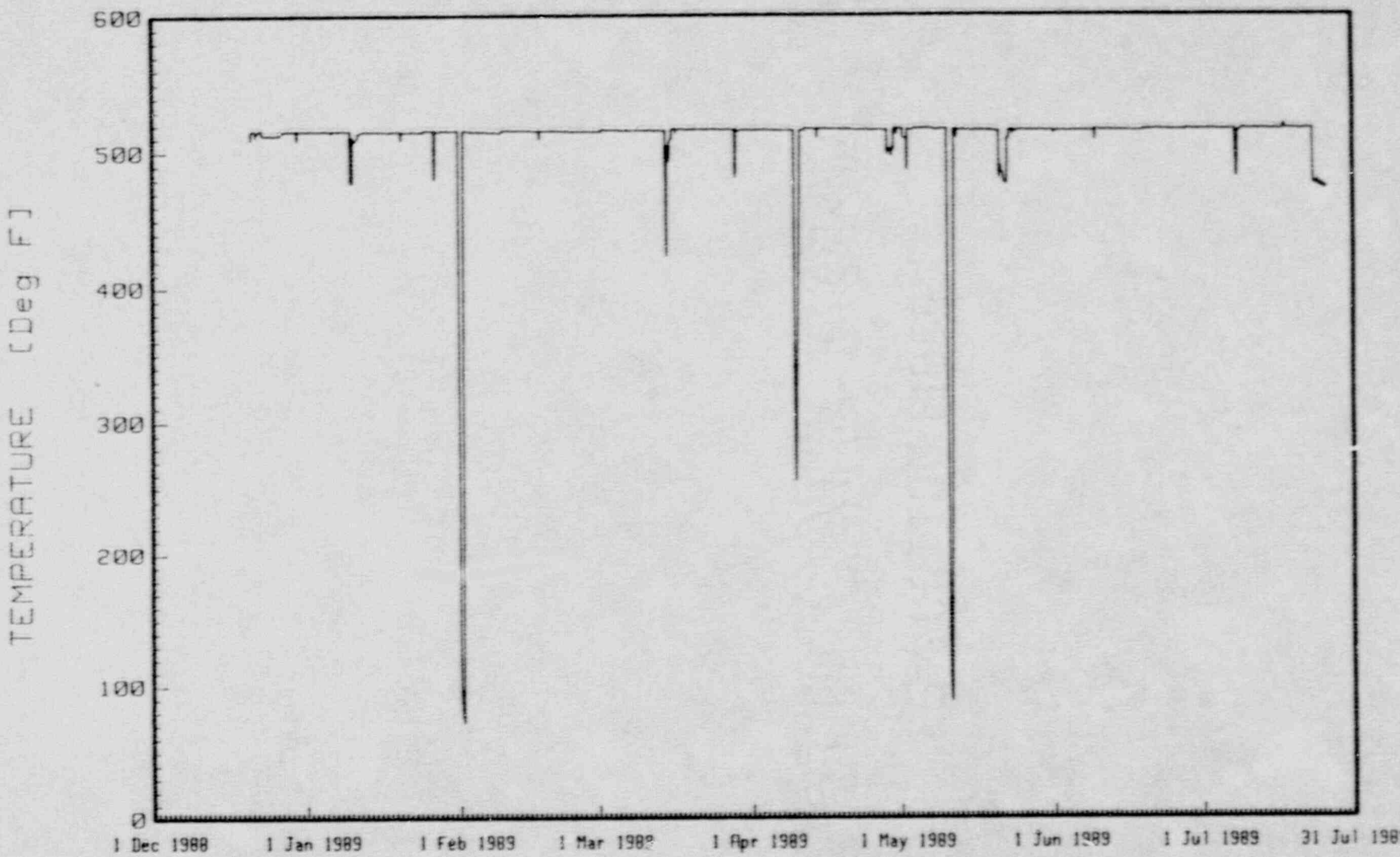
JAFNPP
FIGURE 2-8
CRACK GROWTH IN 304 SS SPECIMEN



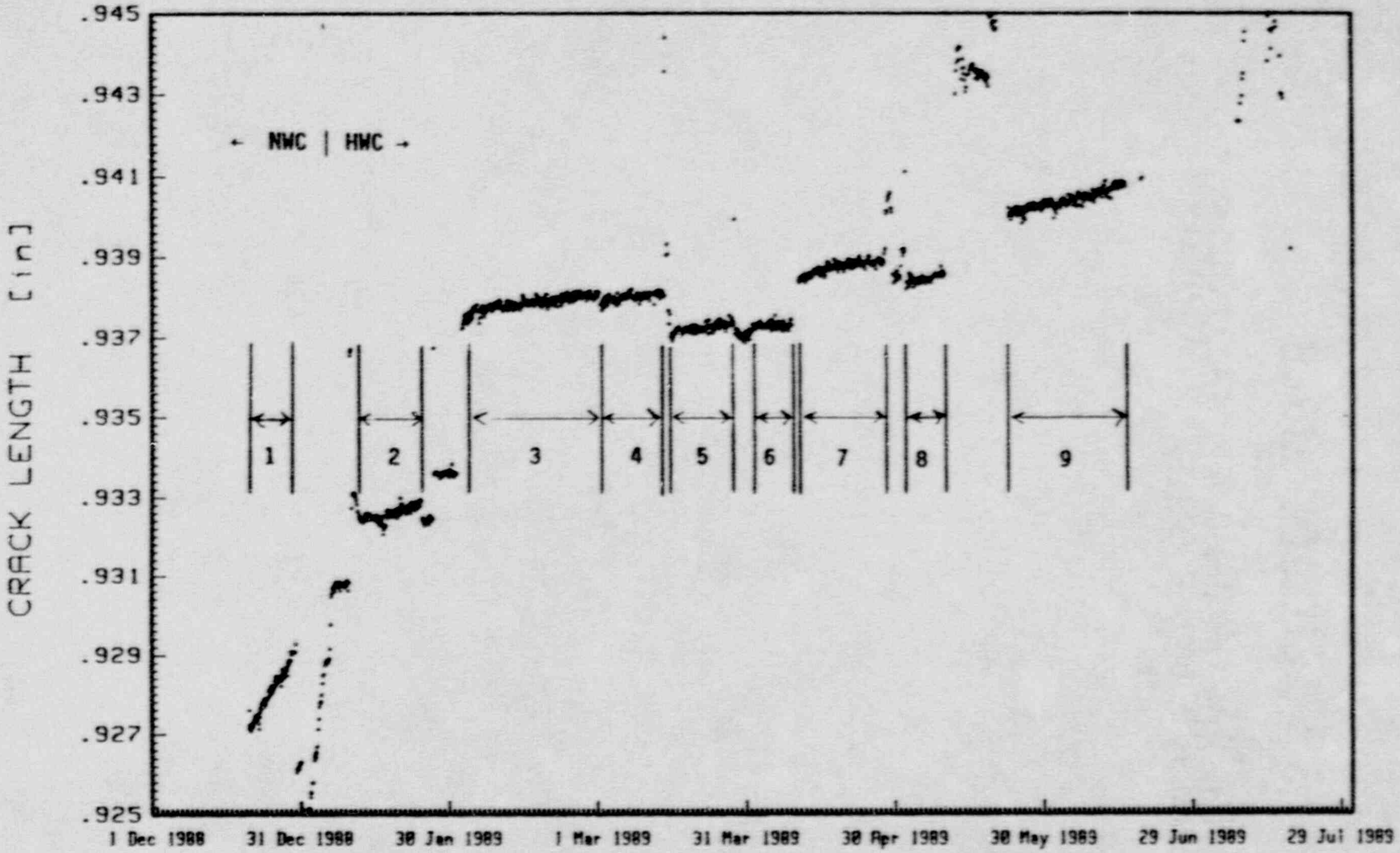
JAFNPP
FIGURE 2-9
CAV SYSTEM INLET FLOW



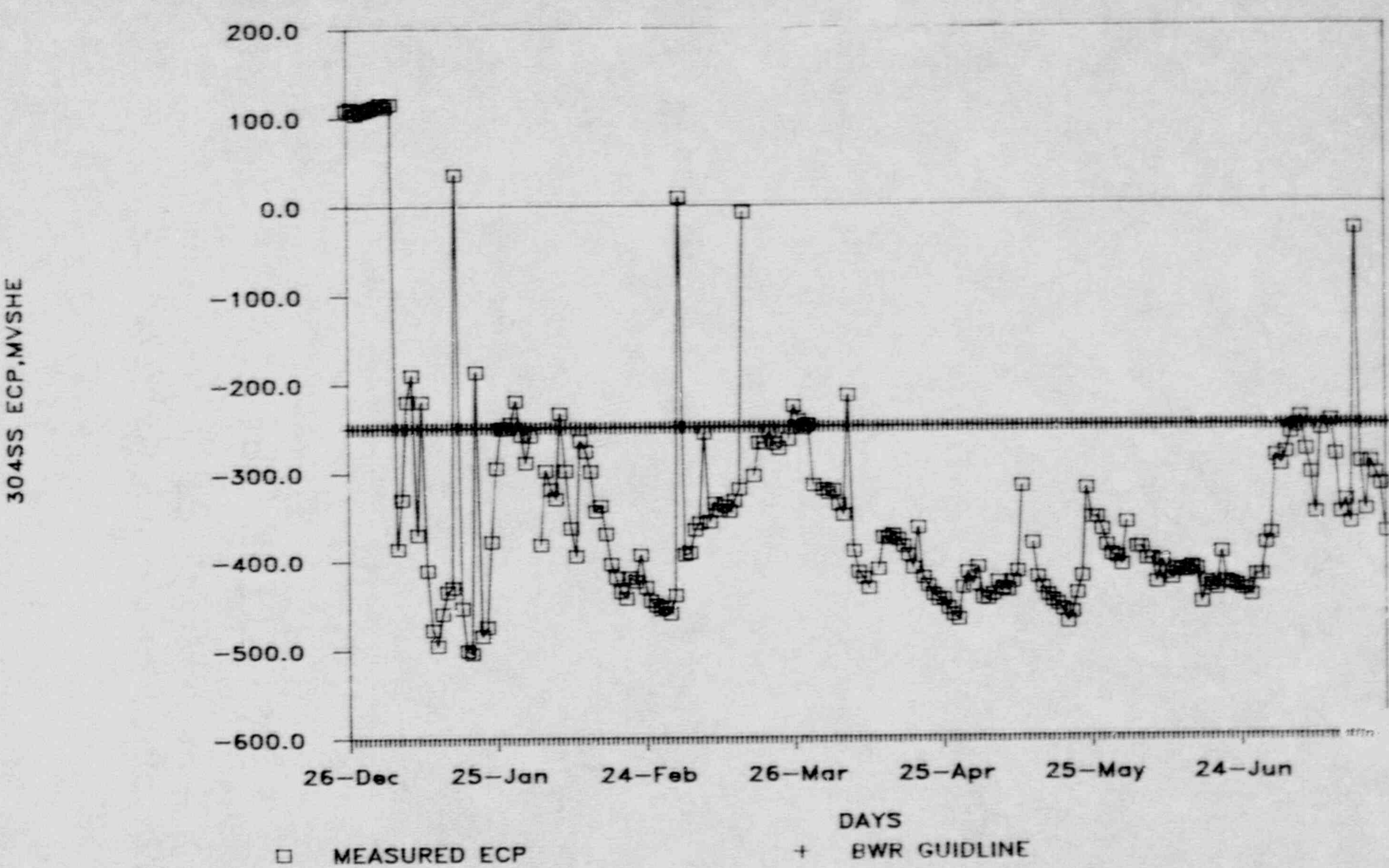
JAFNPP
FIGURE 2-10
CAV SYSTEM TEMPERATURE



JAFNPP
FIGURE 2-11
CRACK GROWTH IN ALLOY 182



JAFNPP
FIGURE 2-12
304 SS ECP



ATTACHMENT 3

SHRINKAGE STRESS ANALYSIS

BACKGROUND

Structural Integrity Associates has performed a shrinkage stress evaluation on the Reactor Recirculation System at the FitzPatrick plant. NUREG-0313 Revision 2, paragraph 3.1.2 discusses the inclusion of shrinkage stresses caused by weld overlays in the fracture mechanics analyses of crack growth in welds in the piping system.

Previously, as detailed in References 6 and 7, the Authority has conservatively used a shrinkage stress of 1000 psi. This value was used, as it was the bounding value calculated in most cases in previous work performed by Structural Integrity Associates at other nuclear power plants.

The analysis performed also determined stresses at locations in the system which contained neither weld overlay repairs nor identified flaws. Such information may be useful in prioritizing weld inspections in the future.

METHODOLOGY

A finite element analysis of each recirculation loop was performed. The recirculation system was assumed to be rigidly constrained at inlet and outlet nozzles, and unrestrained throughout the rest of the system. The finite element model is shown in Figure 3-1.

AS-BUILT DATA

The as-built results for the 1988 refueling outage are included in Table 3-1. The Core Spray weld overlay as-built data has been previously provided in Reference 12. The as-built data for the six weld overlays installed in 1985 and surface finished in 1987 have been previously reported in Reference 1. The as-built shrinkage data is noted in Table 3-2 for each weld.

Weld overlay shrinkage measurements were performed following weld overlay application. The measured shrinkages were imposed at the weld overlay repaired locations in the model of each loop, and were treated as the system loading for the purposes of the analysis. A steady-state static stress analysis was performed by the finite element method using the computer program ALGOR SUPERSAP. Weld overlay shrinkage was simulated by imposing a pseudo-temperature change boundary condition on the model representing the weld overlays, so that the regions containing weld overlays would appear to shrink due to thermal contraction. This thermal contraction at a weld overlay location resulted in a stress, i.e., shrinkage stress, at weld locations in the system.

SUMMARY

The following is a summary of the weld overlay shrinkage induced stress at unrepaired locations.

WELD NUMBER	TOTAL STRESS (psi)	STRESS USED IN EVALUATION (psi)
12-4	484.3	1000
28-33	230.2	1000
28-53	53.0	1000
28-112	46.8	1000

Based on the above evaluation the use of a shrinkage stress on 1000 psi is acceptable for the fracture mechanics evaluations as detailed in References 6 and 7.

Table 3-1

As-Built Data (1988 Refueling Outage Results)

Weld No.	Thickness (in.) Actual	Design	Length (in.) Actual	Design	Shrinkage
28-92 (Note 1)	0.55	0.51	6.4	7	0.092
28-48	0.64	0.52	6.5	6.4	0.094
28-52	0.572	0.47	6.9	6.6	0.027
28-116	0.542	0.48	7.2	6.8	0.048
28-113 (Note 2)	0.582	0.46	5.4	6.6	0.043
22-63 (Note 3)	0.503	0.38	5.75	5.3	0.095
12-15 (Note 4)	0.307	0.24	2.95	4	0.246
N8A-SE-2	0.467	0.285	4.54	4	0.232 (Note 6)
4-118	0.313	0.155	3.45	2 (Note 5)	0.431 (Note 6)

NOTES TO TABLE 3-1

Note 1: Length on valve side is below 3.5" as overlay is blended into the valve taper.

Note 2: Length on valve side is below 3.3" as overlay is blended into the valve taper.

Note 3: Shrinkage does not affect stress analysis. Weld is on end cap. Shrinkage data is provided for information only.

Note 4: Length on sweep-o-let is tapered into the fitting.

Note 5: The weld configuration is a jet pump instrument (JPI) nozzle assembly weld (8"x4" reducer) to the JPI safe end.

Note 6: Minimal shrinkage stress effect as the JPI nozzle assembly is connected to instrument tubing and is anchored on the reactor vessel side only.

Table 3-2

Shrinkage Measurements of Remaining Weld Overlays on the Reactor Recirculation System

Weld	Shrinkage (in.)
12-12	0.103
12-23	0.143
12-64	0.137
12-69	0.126
12-70	0.157
22-22	0.116

Note: Weld 22-22 is an end cap, and thus there are no shrinkage stress effects. In addition, the jet pump instrument nozzle assembly welds are not anchored on the instrument tubing, and thus the shrinkage stress effects are minimal.

REFERENCES

1. NYPA letter, J. C. Brons to D. R. Muller, dated April 9, 1987 (JPN-87-018), "Intergranular Stress Corrosion Cracking Inspection Results for the Reload 7/Cycle 8 Refuel Outage."

2. NRC letter, R. A. Capra to J. C. Brons, dated April 17, 1987, "Review of IGSCC Inspection Results."

3. NRC Generic Letter 88-01, dated January 25, 1988, which transmitted NUREG-0313 Revision 2, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping."

4. NYPA letter, J. C. Brons to NRC, dated August 16, 1988 (JPN-88-041), provided plans relating to pipe replacement, inspection, repair, and leakage detection.

5. NYPA letter, J. C. Brons to NRC, dated August 19, 1988 (JPN-88-043), "Resistance Heating Stress Improvements."

6. NYPA letter, J. C. Brons to NRC, dated November 10, 1988 (JPN-88-061), summarized results of Reload 8/Cycle 9 Refuel Outage inspections.

7. NYPA letter, J. C. Brons to NRC, dated November 10, 1988 (JPN-88-062), provided additional information on results of IGSCC inspections.

8. NRC letter, R. A. Capra to J. C. Brons, dated November 18, 1988, concluded that the FitzPatrick plant could be safely returned to service.

9. NYPA letter, J. C. Brons to NRC, dated March 24, 1989 (JPN-89-012), corrected errors in Reference 7.

10. NYPA letter, J. C. Brons to NRC, dated July 26, 1989 (JPN-89-053), "Response to Request for Additional Information."

11. NYPA letter, J. C. Brons to NRC, dated July 29, 1989 (JPN-89-054), provided inspection summary for Fall 1988 Outage.

12. NYPA letter, J. C. Brons to NRC, dated September 29, 1989 (JPN-89-063), providing inspection summary for the Fall 1989 Maintenance Outage.