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WNP-2 LOCA ANALYSIS FOR SINGLE LOOP OPERATION

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ACKNOWLEDGMENT

Advanced Nuclear Fuels Corporation appreciates the major contribution to the WNP-2 LOCA analysis for single loop operation made by C. E. Hendrix of Intermountain Technologies, Inc.

1.0 INTRODUCTION

The results of a LOCA-ECCS analysis for the Supply System Nuclear Project No. 2 (WNP-2) to support single loop operation (SLO) are reported in this document. These calculations were performed with the generically approved Advanced Nuclear Fuels Corporation (ANF) EXEM/BWR Evaluation Model[1,2] in accordance with Appendix K of 10 CFR 50[3], and the results comply with the U.S. NRC 10 CFR 50.46 criteria.

The initial SLO condition selected for this analysis was at 75 percent power and 50 percent flow. The objective of the analysis was to demonstrate that the Maximum Planar Linear Heat Generation Rate (MAPLHGR) vs. exposure curve calculated for ANF fuel in two loop operation is applicable for SLO.

2.0 SUMMARY

The results of the ECCS analysis presented herein support the use of the two loop MAPLHGR's for ANF fuel when the WNP-2 reactor is operating in the SLO mode.

Single loop operation of WNP-2 with the two loop ANF fuel MAPLHGR's assures that the emergency core cooling systems for the WNP-2 plant will meet the U.S. NRC acceptance criteria of 10 CFR 50.46 for loss-of-coolant accident breaks up to and including the double-ended severance of a reactor coolant pipe. That is:

1. The calculated peak fuel element clad temperature does not exceed the 2200 F limit.

2. The calculated total oxidation of the cladding nowhere exceeds 17% times the total cladding thickness before oxidation.

3. The calculated maximum hydrogen generation does not exceed 1% of the zircaloy associated with the active fuel cladding in the reactor.

4. The LOCA cladding temperature transient is calculated to be terminated at a time when the core is still amenable to cooling.

5. The system long-term cooling capabilities provided for the initial core and subsequent reloads remain applicable to ANF fuel.

The MAPLHGR vs. exposure curve applicable to ANF fuel in WNP-2 which supports MCPR consistent with the flow dependent MCPR curve (1.35 at 50 percent of

rated flow) for two loop and SLO is shown in Figure 2.1. The APLHGR limit (MAPLHGR) of Figure 2.1 is constant at 13.0 kw/ft for assembly average exposures from 0 to 20 GWd/MTU and the limit decreases approximately linearly from 13.0 to 7.9 kw/ft for assembly average exposures from 20 to 35 GWd/MTU.

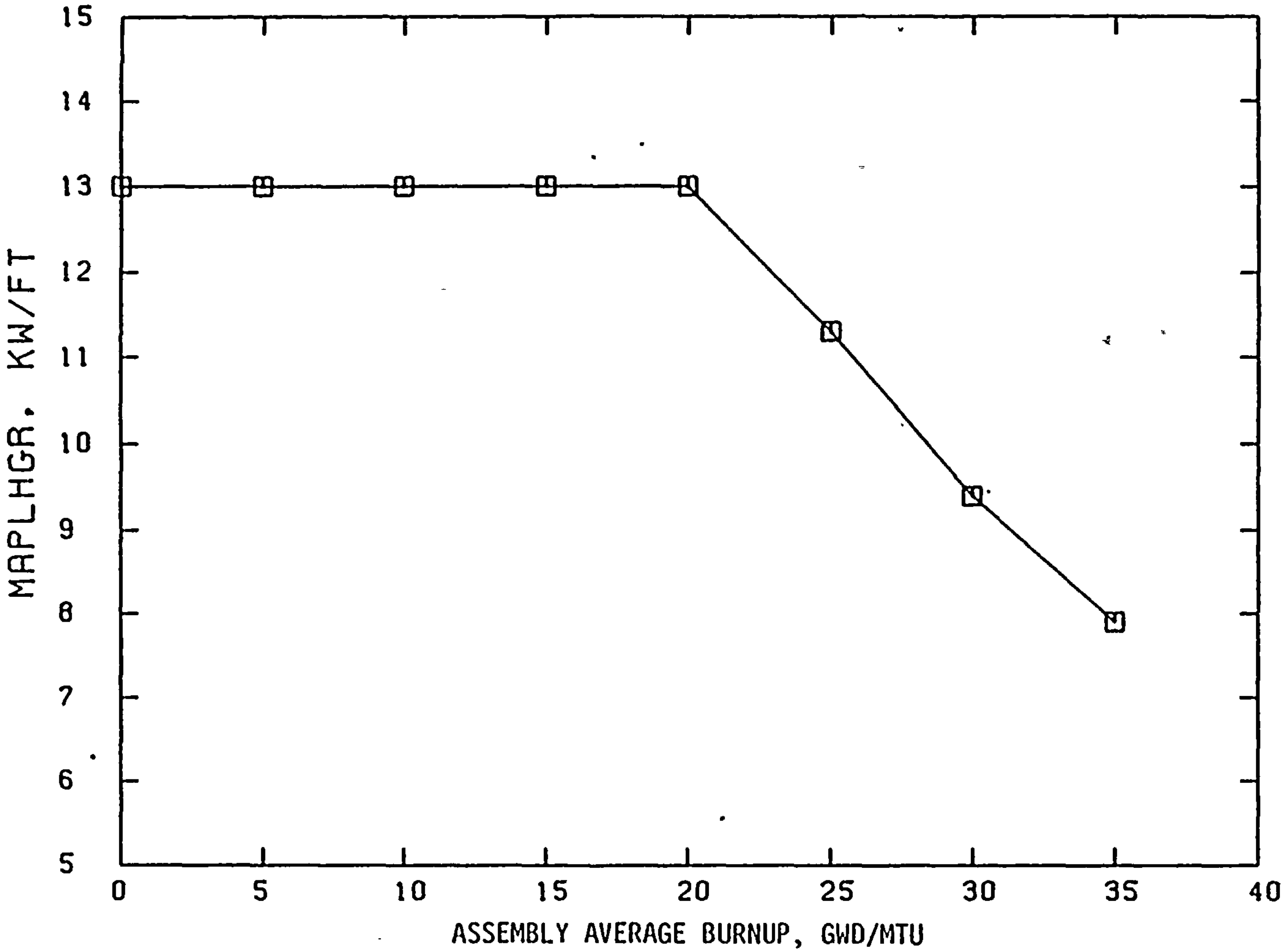

Figure 2.1 MAPLHGR vs. Assembly Average Burnup For ANF Fuel In WNP-2

3.0 JET PUMP BWR ECCS EVALUATION MODEL

3.1 LOCA During Single Loop Operation

The loss-of-coolant accident (LOCA) break spectrum analysis for a BWR/5 for two loop operation conditions is described in Reference 4, and the WNP-2 plant specific MAPLHGR analysis for two loop operation is described in Reference 5. This document describes LOCA-ECCS analysis and MAPLHGR justification for WNP-2 SLO operation. The same single failure assumption was made in both the two loop and the SLO LOCA-ECCS analyses.

During SLO the recirculation pump in the inactive loop is not in operation and the rotor is effectively locked while the recirculation flow control valve is at the minimum position[6]. Both intake and discharge block valves in the idle recirculation loop remain open during SLO. A significant resistance to the loop recirculation flow does exist, but a small amount of flow can pass through the idle loop during LOCA conditions. A break can be hypothesized to occur in either loop. However, a break in the inactive loop would behave essentially like a break during two loop operation except that substantial break flow would come from only one side of the break (because of the fixed pump rotor and recirculation flow control valve at its minimum position). System performance would then be like that resulting from a somewhat smaller break during two loop operation. The scenario of breaks smaller than the limiting break has already been covered by the BWR/5 LOCA break spectrum analysis[4]. Further consideration in this report will be given only to the case where a break occurs in the active loop. This case differs from the two loop case in one important respect: there is no flow coastdown in the intact (idle) recirculation loop.

Previous SLO analysis[7,8] assumed that the consequence of a lack of recirculation loop coastdown flow (which continues to supply liquid from the downcomer to the lower plenum, during two loop operation) would be an almost immediate flow stagnation in the core and a very early CHF (0.1 sec); this

resulted in degraded heat transfer very early in the transient and required that a MAPLHGR reduction factor of 0.84 be imposed for SLO conditions on the MAPLHGR for NSSS vendor fuel for WNP-2.

3.2 EXEM/BWR Application To WNP-2

The ANF EXEM/BWR ECCS Evaluation model codes were used for this SLO LOCA-ECCS calculation. The codes which comprise EXEM consist of RODEX2[9], RELAX[10], FLEX[11] and HUXY/BULGEX[12,13]. The latest versions of these codes were used for this SLO analysis, and they are referenced in the generic two loop analyses[4,5].

In the unlikely event that a LOCA would occur during SLO, a rapid drop in core flow would be expected to occur during the early phase of the event because the idle loop pump is not operating. The core flow transient during the early phase (0 to 5 seconds) of a single loop LOCA is the principal event which could distinguish such an accident from a LOCA occurring during normal two loop operation. Generally, the magnitude of the initial drop in core flow will increase as the break size increases, and the critical heat flux (CHF) may occur earlier in time due to the decreased core flow. Therefore, the effect of break area on PCT was considered in this analysis to confirm this trend for SLO condition.

The system behavior during a LOCA is determined primarily by the LOCA break parameters: break location, break size and break configuration together with the ECCS systems and plant geometry. Variation in core geometric parameters produce only secondary effects on the system behavior. Thus, by using bounding core neutronic parameters, the LOCA-ECCS results established by this analysis will apply for future cycles unless substantial changes are made in the plant operating conditions, plant hardware or core design such that the analysis no longer bounds the plant conditions.

The system blowdown calculations for the WNP-2 SLO differ from those for two loop operation in several ways. The back flow through the idle loop jet pump was modeled consistent with expected SLO steady state conditons. The core, steam, feedwater and control rod drive coolant flows and reactor power for SLO were taken from Reference 6. The calculation was performed for a core composed of all ANF 8x8 fuel. Other than the changes mentioned, the system blowdown model remains the same as for the two loop model. The initial conditions are shown in Table 3.1, and the system blowdown nodalization diagram is shown in Figure 3.1.

The system blowdown calculations are followed by HOT CHANNEL calculations. The core geometry is identical with that used in the two loop analysis. The initial conditions were adjusted to correspond to those for the single loop operating point[6]. The power, axial peaking and flow of the hot assembly were determined by an XCOBRA thermal hydraulics calculation to support a MCPR consistent with the flow dependent MCPR curve (1.35 at 50 percent flow). The nodalization and geometry used in the reflood calculation are identical to those of the two loop analysis. In the FLEX code the intact loop is not modeled in detail because intact loop flows are insignificant at the time of rated low pressure core spray. Thus, no changes were required in the FLEX nodalization or geometry. The initial conditions for the reflood calculation are entirely determined by the system blowdown calculation.

The HUXY/BULGEX heatup calculation of the hot plane (center one foot node) was done identically with previous two loop analyses: fuel stored energy, gap thermal conductivity and dimensions from RODEX2 as a function of power and exposure; time of rated spray, decay power, heat transfer coefficients and coolant conditions from RELAX; and time of hot-node-reflood from FLEX. Bounding fission and actinide product decay heat obtained with end-of-cycle neutronics in the system blowdown calculation assure that the power input to the HUXY/BULGEX heatup calculation is conservative. Appendix K spray heat transfer coefficients are used for the spray cooling period: for heated fuel rods, the values of 1.5, 3.0 and 3.5 Btu/hr-ft2-F are used in the interior,

corner, and peripheral rods respectively; for the unheated fuel channel and water rod surfaces, the values of 5.0 and 1.5 Btu/hr-ft2-F are used. For the reflood period, the Appendix K reflood heat transfer coefficient value of 25 Btu/hr-ft2-F is used. Peak cladding temperature (PCT) and the cladding oxidation percentage are specifically determined for the ANF fuel geometry.

TABLE 3.1 WNP-2 REACTOR SYSTEM DATA FOR SINGLE LOOP OPERATION

Primary Heat Output, MW (102% of rated)	(0.75 x 3389) = 2542
Total Reactor Flow Rate, lb/hr (49.8% of rated)	54.0×10^6
Active Core Flow Rate, lb/hr	48.26×10^6
Steam Dome Pressure, psia	1020.
Reactor Inlet Enthalpy, Btu/lb	511.2
Recirculation Loop Flow Rate, lb/hr	13.62×10^6
Steam Flow Rate, lb/sec	10.62×10^6
dwater Flow Rate, lb/sec	10.59×10^6
Rated Recirculation Pump Head, ft	800.
Rated Recirculation Pump Speed, rpm	1782.
Moment of Inertia, lbm-ft^2/rad	22,700
Recirculation Suction Pipe I.D., in	21.56
Recirculation Discharge Pipe I.D., in	21.56

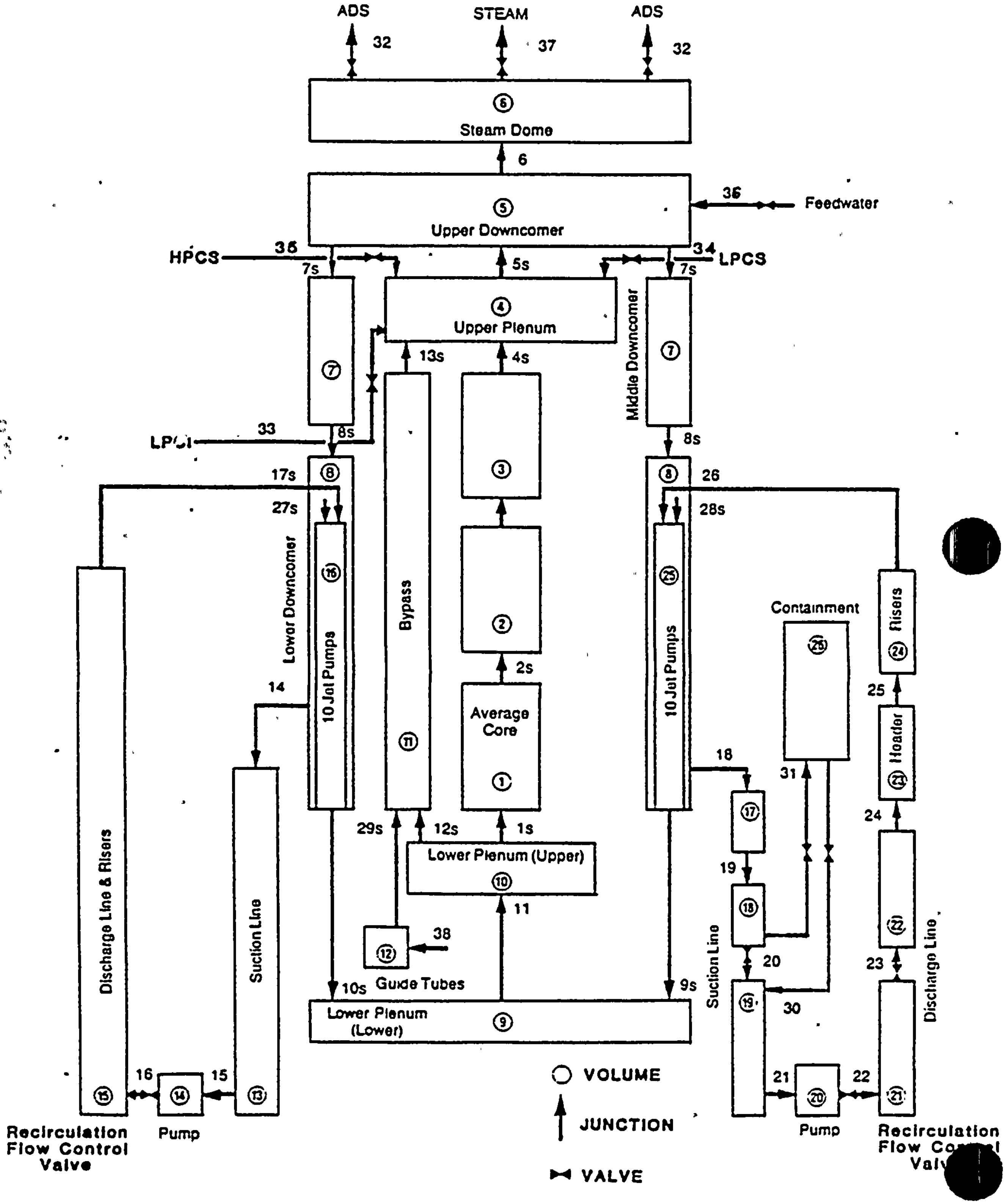

Figure 3.1 System Blowdown Nodalization For WNP-2

4.0 ANALYSIS RESULTS AND CONCLUSIONS

4.1 Break Spectrum

The existing break spectrum for WNP-2, performed for two loop operation, showed the most limiting break to be a split break in the recirculation suction piping with an area of six-tenths of the double-ended pipe area (0.6 DES/RS)[4]. A larger break occurring during SLO might cause an earlier CHF to occur near the core mid-plane resulting in higher PCT's than those calculated for the 0.6 DES/RS break. Therefore, analysis of a LOCA during SLO was made for two suction break sizes, at beginning of life (BOL) fuel exposure, to determine if phenomena unique to SLO initial conditions would cause a change in the previous calculated limiting break. The two breaks analyzed were a double-ended guillotine break on the recirculation pump suction side (1.0 DEG/RS) and the limiting 0.6 DES/RS. A third analysis for a double-ended guillotine break on the recirculation pump discharge side (1.0 DEG/RD) was performed through the time of rated LPCS to confirm the limiting break remained in the suction line.

Table 4.1 shows the calculated PCT's for LOCA's initiated from both normal operation and SLO for the 0.6 DES/RS and the 1.0 DEG/RS breaks. The calculated PCT for the 0.6 DES/RS break is higher than the 1.0 DEG/RS break for both SLO and normal two loop operation by approximately the same amount. These results demonstrate that the break spectrum results presented in Reference 4 also apply for SLO. In addition, the peak fuel rod stored energy and surface temperatures at the time of rated LPCS were compared for the 0.6 DES/RS and the 1.0 DEG/RD breaks. This comparison showed these temperatures for the 0.6 DES/RS break were higher than those for the pump discharge side break by approximately 100°F at the time of rated low pressure core spray (LPCS). Also at the time of LPCS, the blowdown core decay power in the limiting break (0.6 DES/RS) is twelve percent higher than for the discharge break. Therefore, it was not necessary to continue the discharge break calculation through core reflood.

4.2 MAPLHGR Results

The MAPLHGR results were obtained by analysis of the 0.6 DES/RS break at the most limiting fuel exposure(5). At 20 GWd/MTU the fuel stored energy used to initialize the RELAX/HOT CHANNEL calculation is the maximum calculated to occur in the cycle as determined by the RODEX2 code. This means that the fuel temperature in the hot channel is at its maximum calculated value regardless of the time in the cycle that the pipe rupture is assumed to occur. The PCT at 20 GWd/MTU was also observed to the highest value over the exposure range for WNP-2. With no changes in the fuel design and using bounding neutronic input data to the system blowdown calculation, the MAPLHGR result at 20 GWd/MTU is the only calculation required to show that the two loop MAPLHGR results are applicable to SLO.

Table 4.2 lists the major event times for this analysis. System blowdown results are given in Figures 4.1 through 4.16. System refill and reflood results are shown in Figures 4.17 through 4.19. Results from a RELAX/HOT CHANNEL calculation are given in Figures 4.20 through 4.22. The results are applied as boundary conditions for the HUXY/BULGEX heat up calculation at the most limiting exposure. The resulting clad temperatures as calculated by HUXY/BULGEX are given in Figure 4.23.

An examination of these plots reveals the following information:

1. The sudden loss of drive fluid in the jet pumps allowed a sudden drop in lower plenum pressure of sufficient magnitude to allow flow through the inactive jet pumps to "reverse" from their initial negative flow to a positive flow in the earlier part of the blowdown.

2. The lack of a pump coastdown in the intact loop allowed flow through the suction and exit junctions of the operating (broken loop) jet pumps to remain in the positive direction (the drive, of course

reversed to supply fluid to the break) during the earlier part of the blowdown.

3. Because of 1. and 2. above, the initial drop in core flow was not sufficient to cause an early CHF at the core mid-plane. CHF is delayed until approximately 8 seconds after the time of break.

The MAPLHGR results are presented in Tables 4.2 and 4.3. The PCT values for the two full SLO breaks are approximately 100°F higher than the corresponding breaks in the two loop analysis primarily because the recirculation flow control valve is not in the full open position. This added resistance does not affect the system blowdown calculation time, because the flow at the break is choked during the system blowdown. During the latter part of the blowdown, the break is unchoked and the additional resistance from the recirculation flow control valve decreases break flow and delays the reflood calculation time by approximately 10 seconds. Thus, the SLO PCT values are higher than the two loop PCT values primarily due to this system effect, but there is still more than 300°F of margin to the PCT limit of 2200°F.

The results presented here for the limiting break (0.6 DES/RS) and highest fuel exposure with maximum stored energy show that the calculated PCT is 1884°F for a MAPLGHR of 13 kw/ft. Since the PCT and metal-water reaction percent are well within the Appendix K of 10 CFR 50[3] limits, it is concluded that the MAPLHGR's for ANF fuel as calculated for two loop operation apply for SLO. A MAPLHGR reduction factor for ANF fuel in WNP-2 is not required for SLO.

TABLE 4.1 WNP-2 BREAK SPECTRUM RESULTS FOR NORMAL OPERATION AND SLO

Break	PCT(°F) (Two Loop Operation)	PCT(°F) (SLO)
0.6 DES/RS	1698	1786
1.0 DEG/RS	1650	1752

All at beginning of life exposure.

TABLE 4.2 LOCA EVENT TIMES - SINGLE LOOP OPERATION SUCTION BREAKS

Event	Time (sec) 0.6 DES/RS ($3.04\ ft^2$)	1.0 DEG/RS ($5.07\ ft^2$)
Start	0.00	0.00
Initiate Break	0.05	0.05
Feedwater Flow Stops	0.55	0.55
Steam Flow Stops	5.05	5.05
Low Mixture Level	9.2	9.0
Jet Pumps Uncover	11.4	11.3
Recirculation Suction Nozzle Uncovers	16.8	16.4
Lower Plenum Flashes (Quality > 0)	16.3	18.4
HPCS Flow Starts	17.7	17.5
LPCI Flow Injection Starts	62.2	62.1
Rated Spray Calculated	75.3	75.0
Depressurization Ends (vessel pressure reaches 1 atm)	126.3	119.0
Start of Reflood (high density fluid enters core)	156.9	150.2
Peak Clad Temperature Reached	182.3	177.0

TABLE 4.3 MAPLHGR RESULTS FOR ANF FUEL - SINGLE LOOP OPERATION

Result	0.6 DES/RS Break
Assembly Average Burnup, GWd/MTU	20.
MAPLHGR, kw/ft	13.0
Peak Cladding Temperature, °F	1884
Peak Temperature Axial Location, ft	6.25
Local Zr/Water Reaction (Max), %	1.26
Total Hydrogen Generated, % of Total Zr Generated	<1

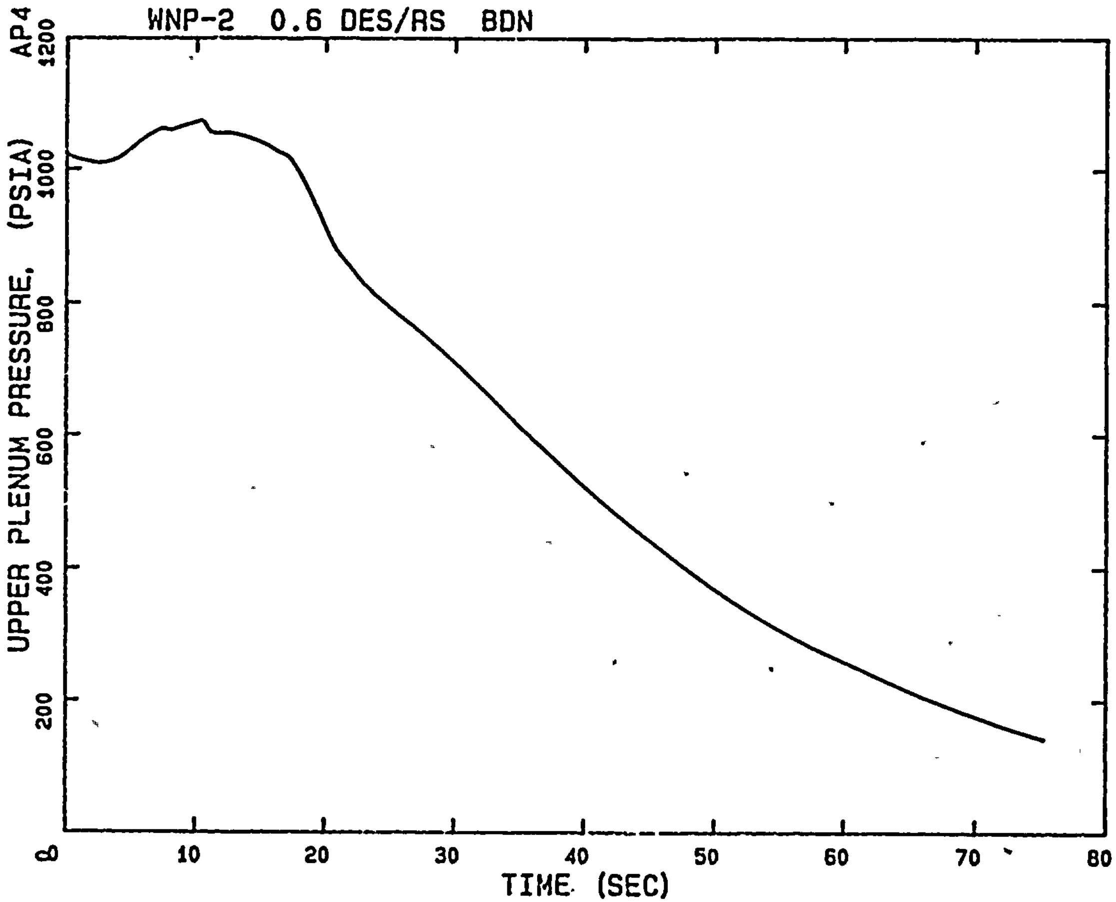

Figure 4.1 Blowdown System Pressure

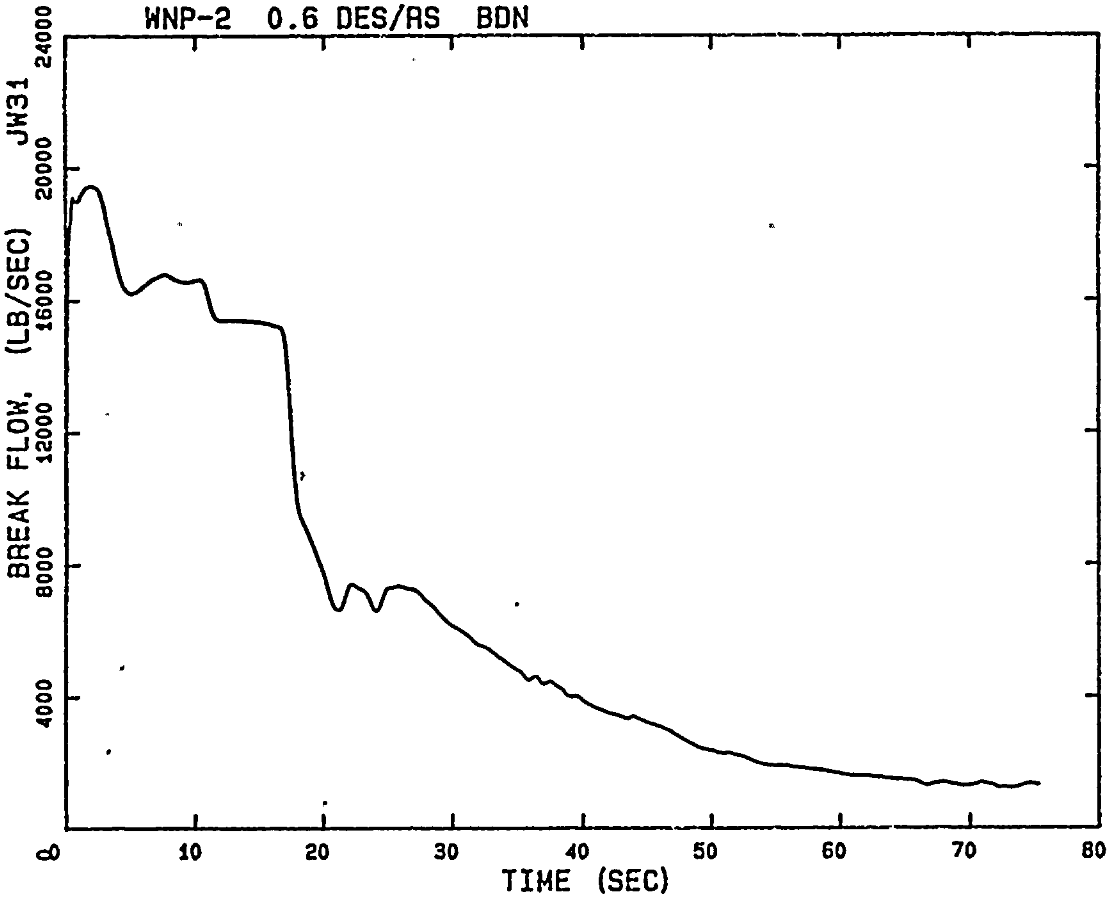

Figure 4.2 Blowdown Total Break Flow

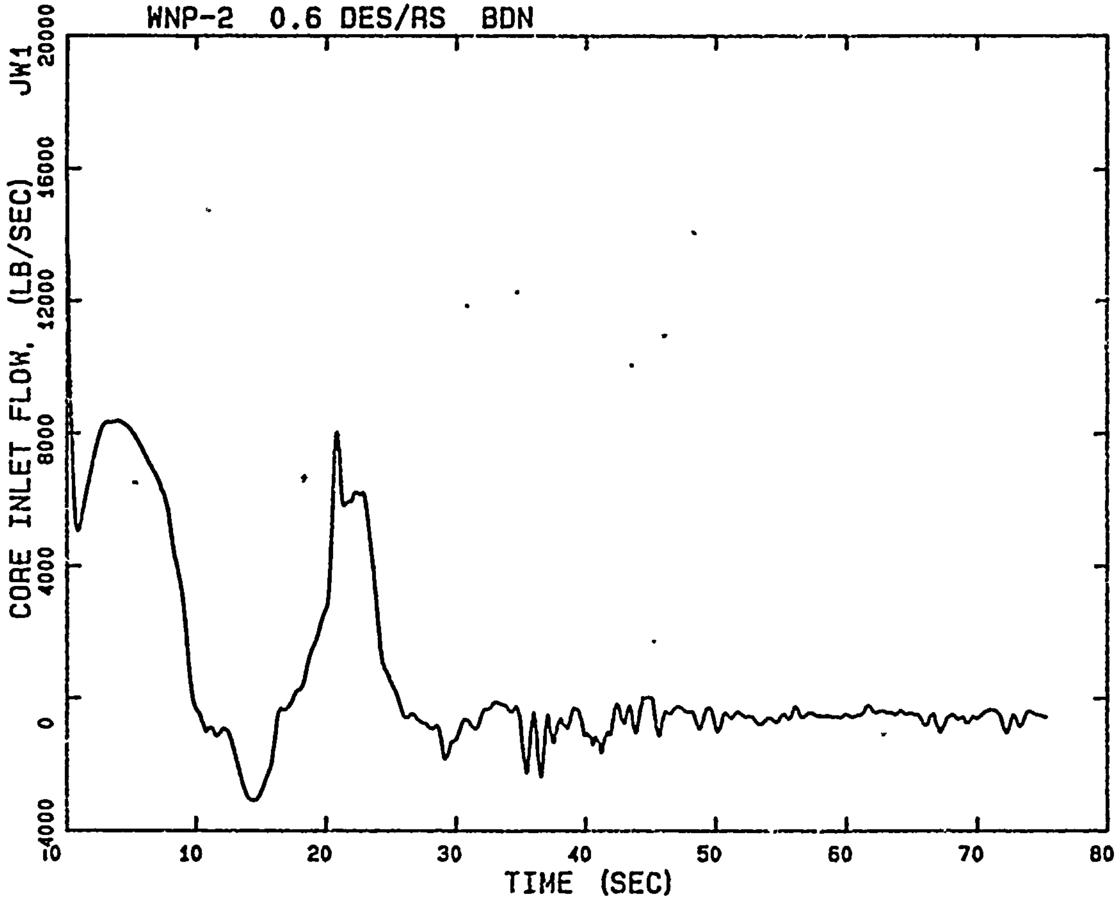

Figure 4.3 Blowdown Average Core Inlet Flow

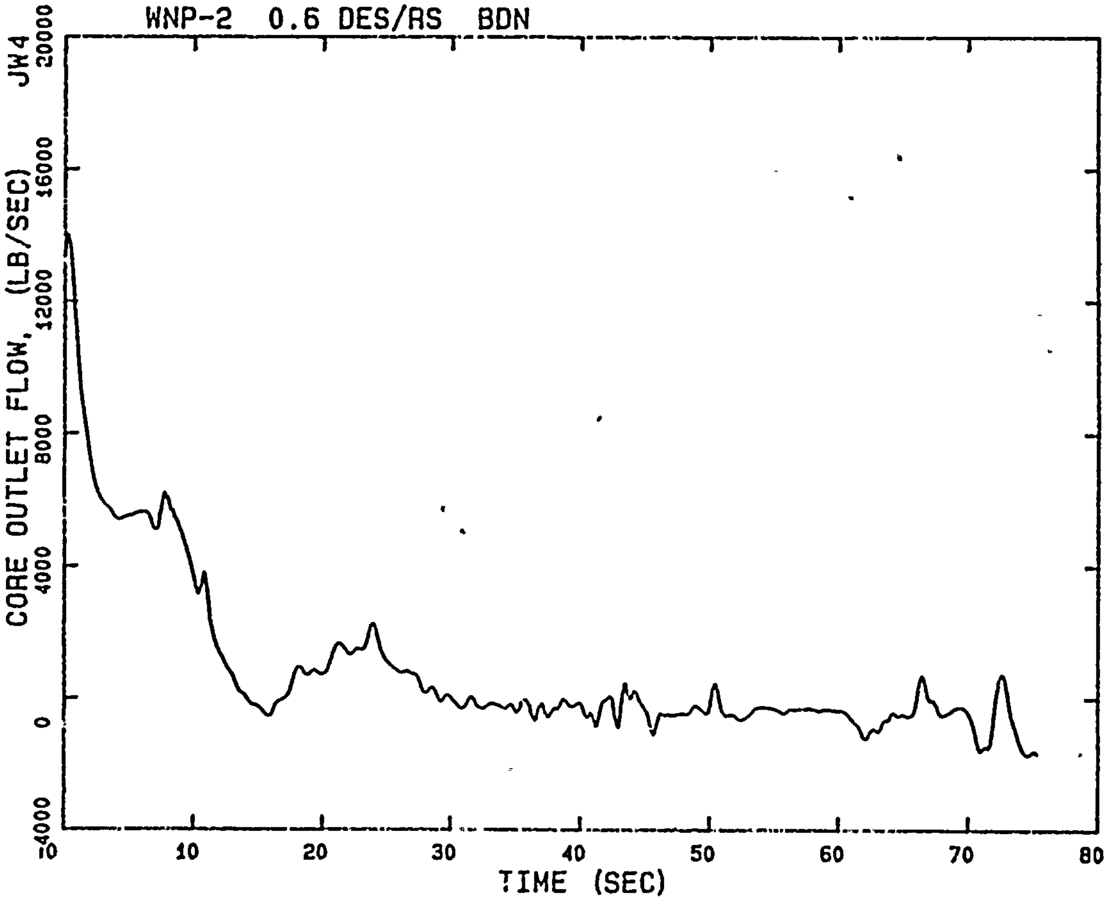

Figure 4.4 Blowdown Average Core Outlet Flow

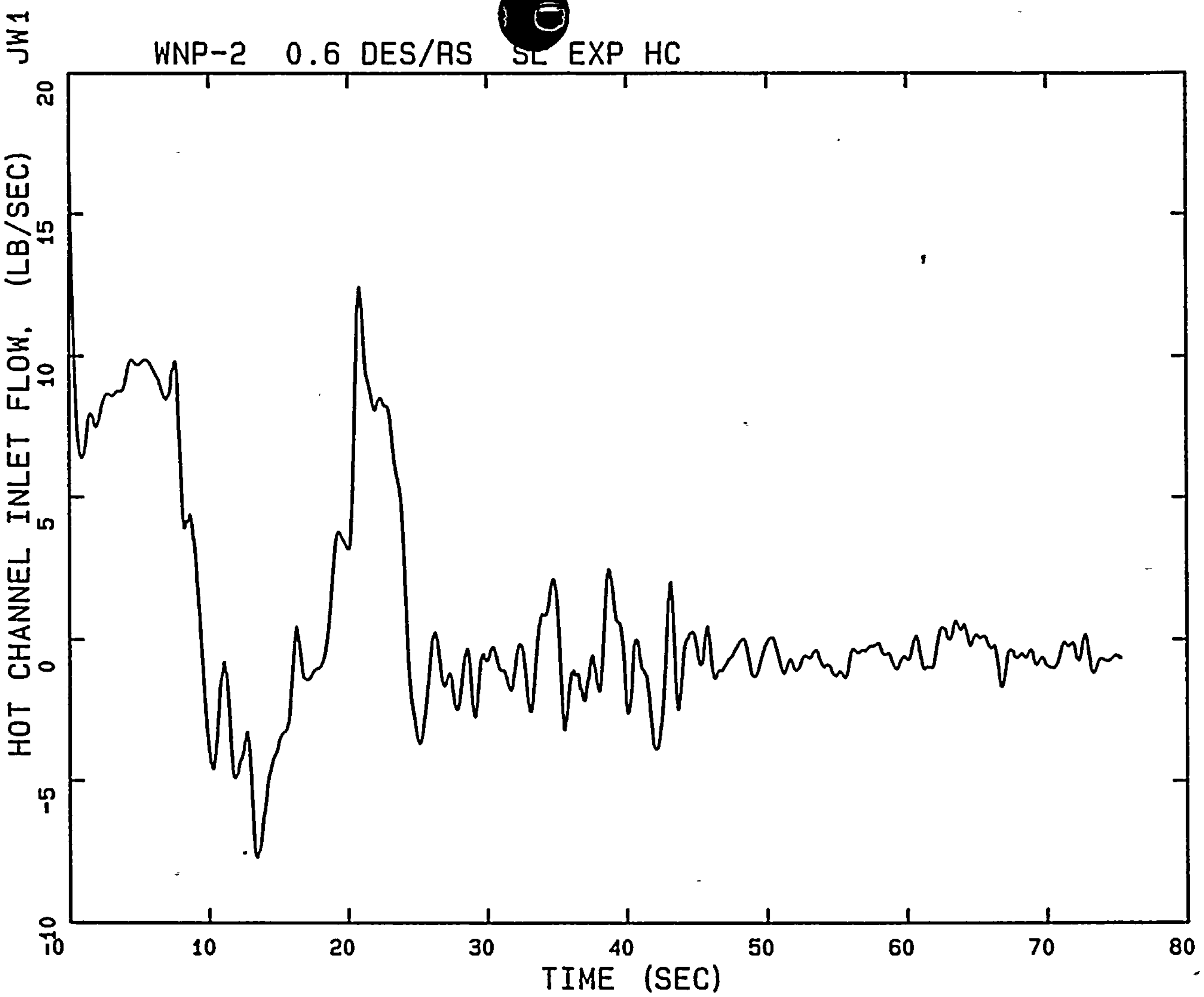

Figure 4.5 Blowdown Hot Channel Inlet Flow

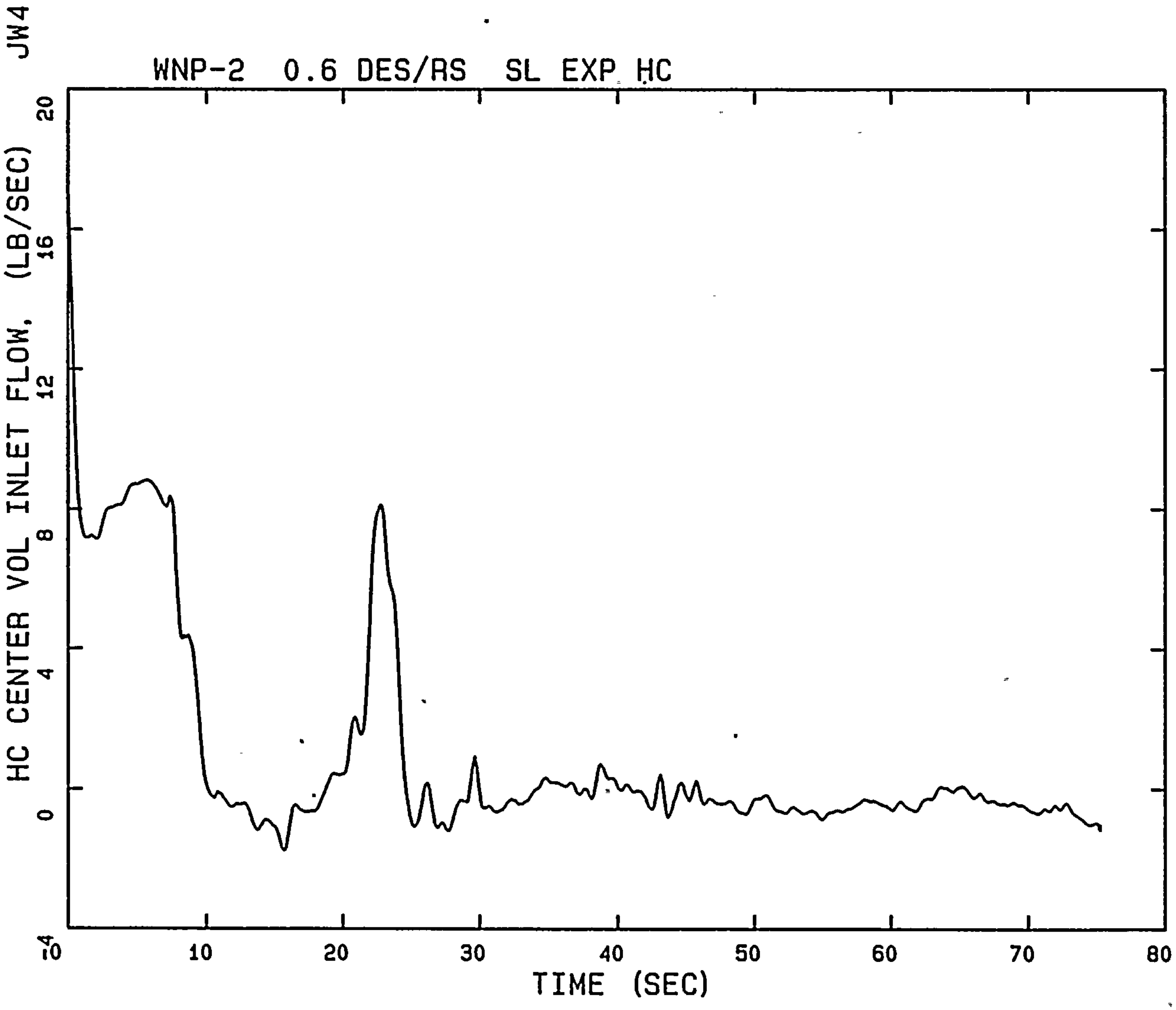

Figure 4.6 Blowdown Hot Channel Midplane Flow

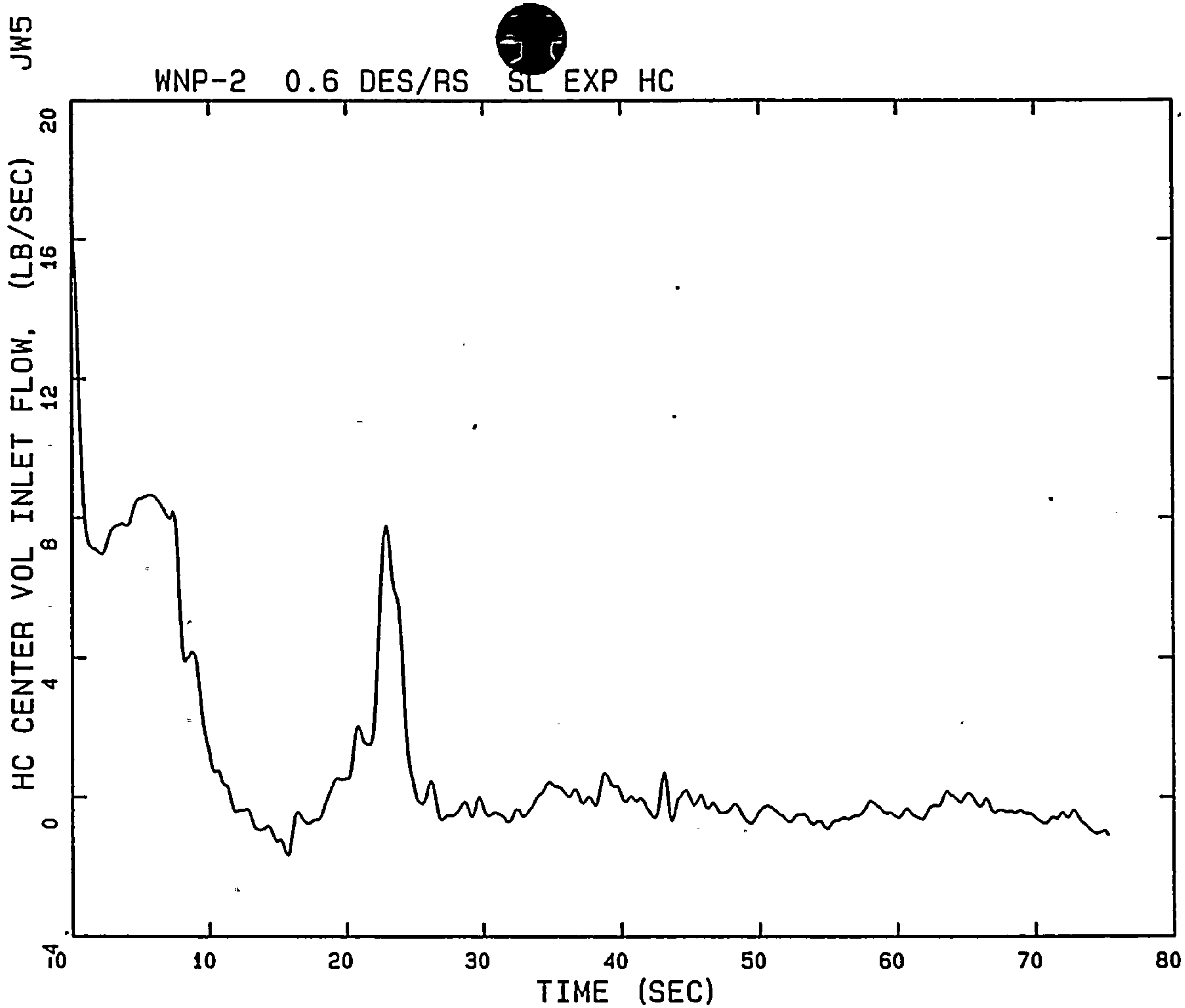

Figure 4.7 Blowdown Hot Channel Outlet Flow

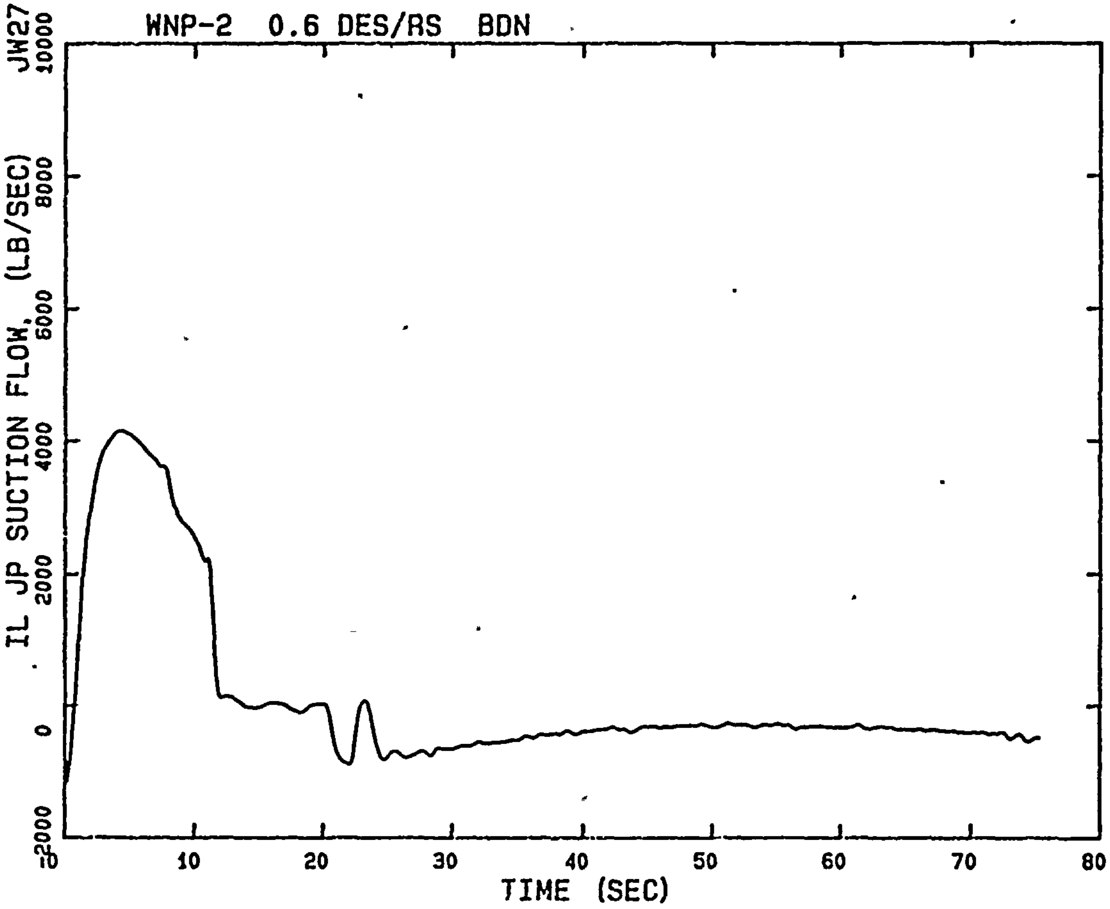

Figure 4.8 Blowdown Intact Loop Jet Pump Suction Flow

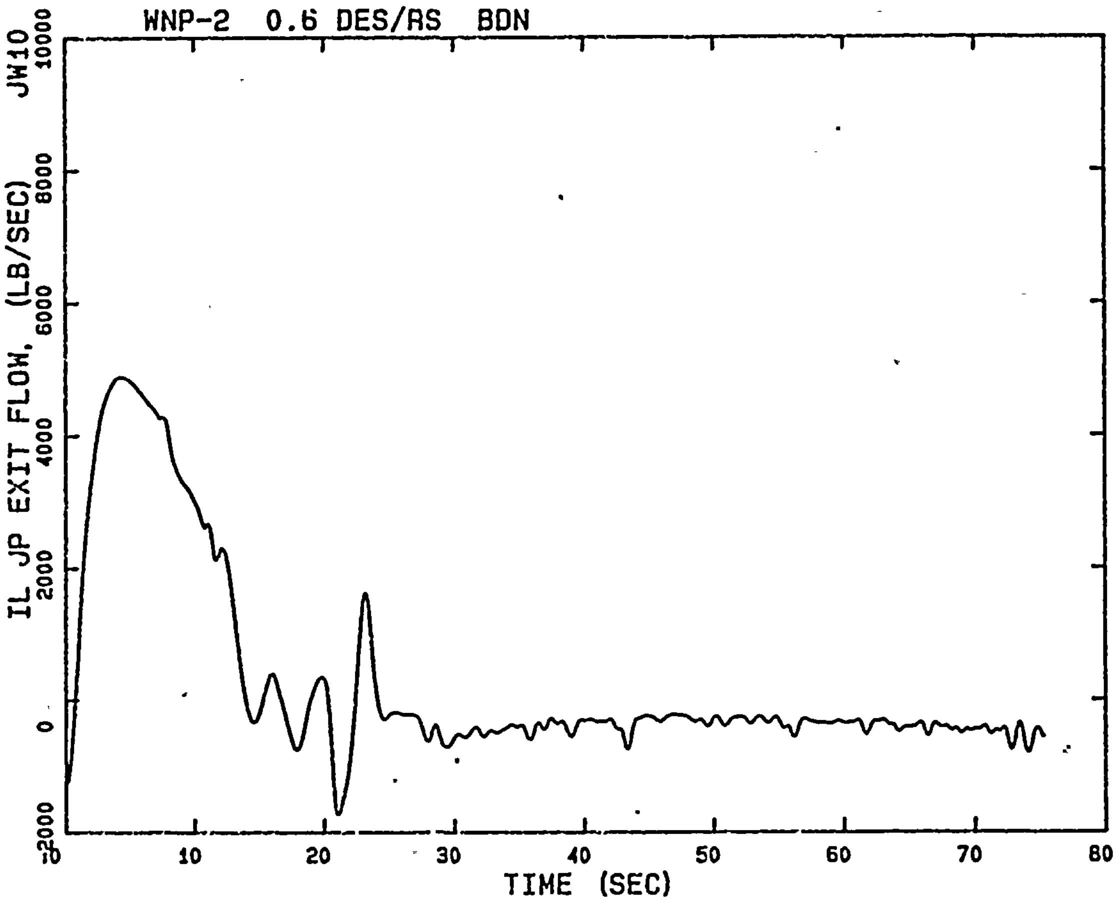

Figure 4.9 Blowdown Intact Loop Jet Pump Exit Flow

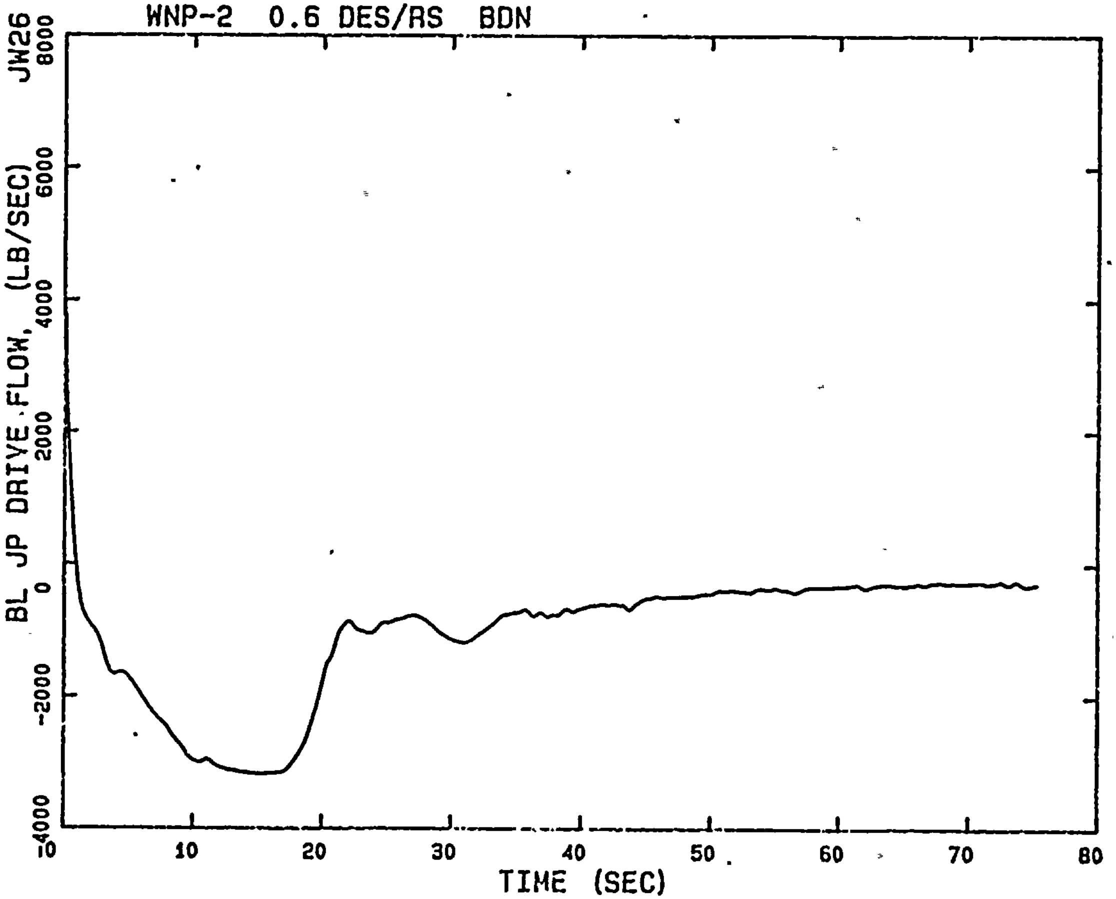

Figure 4.10 Blowdown Broken Loop Jet Pump Drive Flow

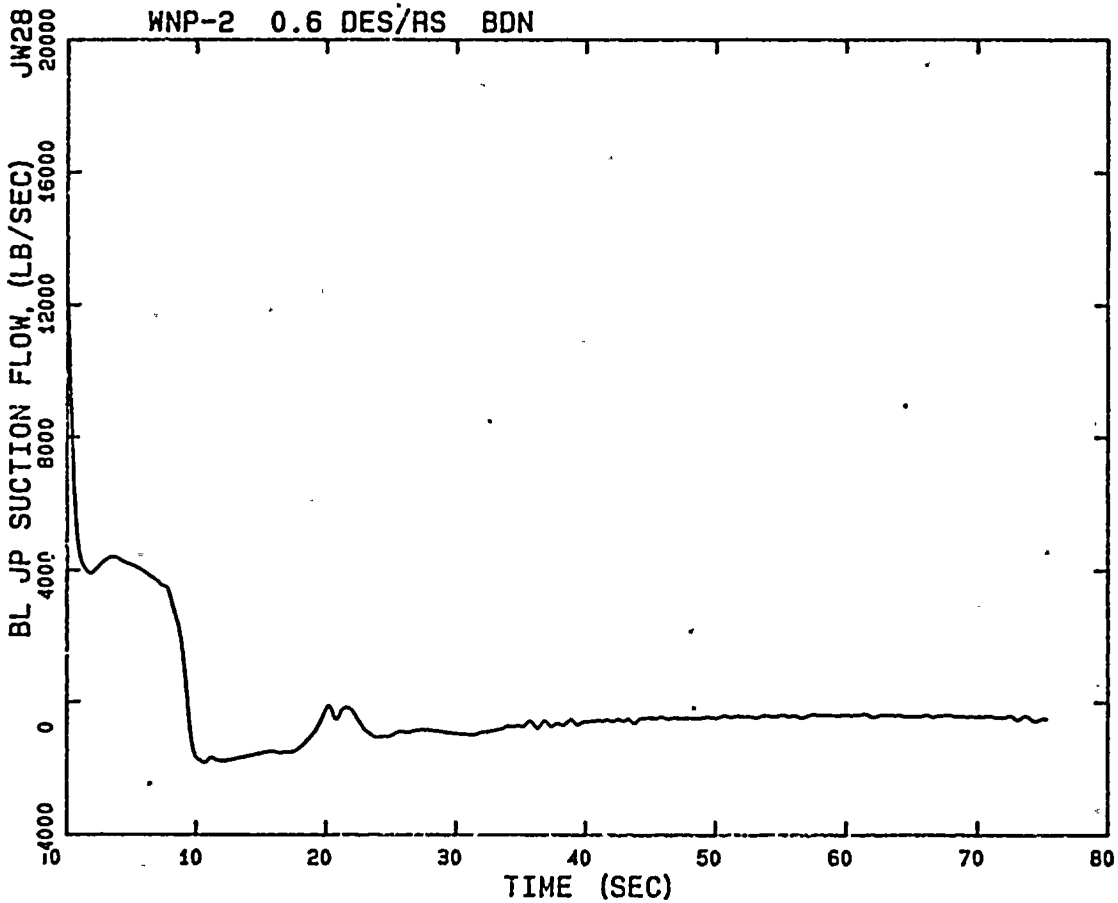

Figure 4.11 Blowdown Broken Loop Jet Pump Suction Flow

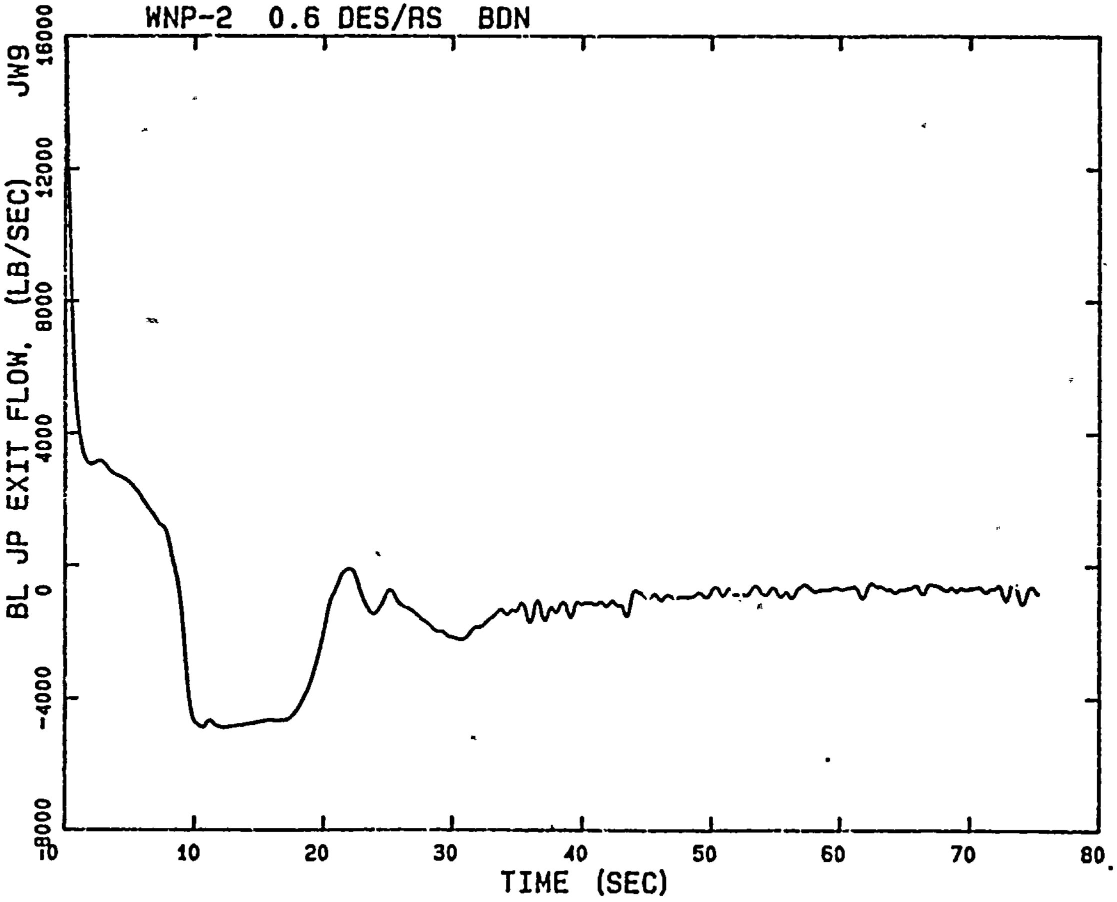

Figure 4.12 Blowdown Broken Loop Jet Pump Exit Flow

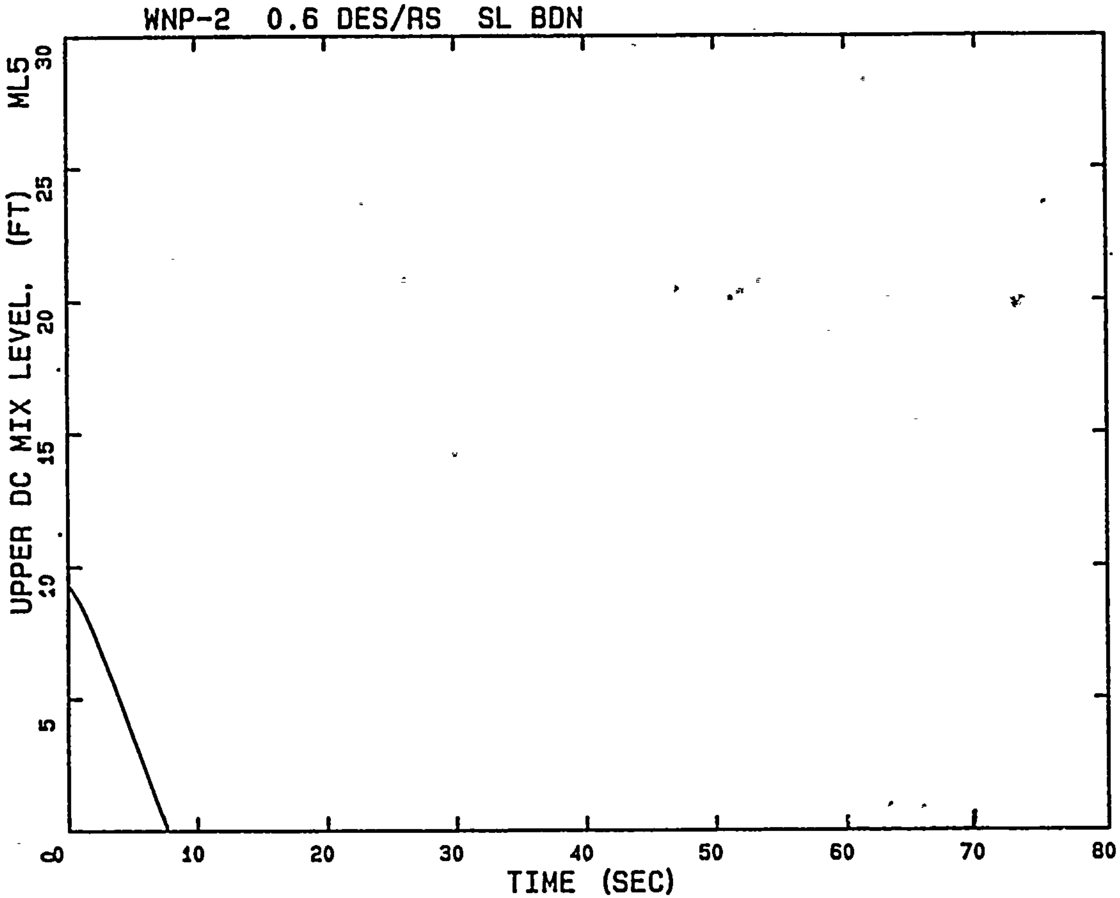

Figure 4.13 Blowdown Upper Downcomer Mixture Level

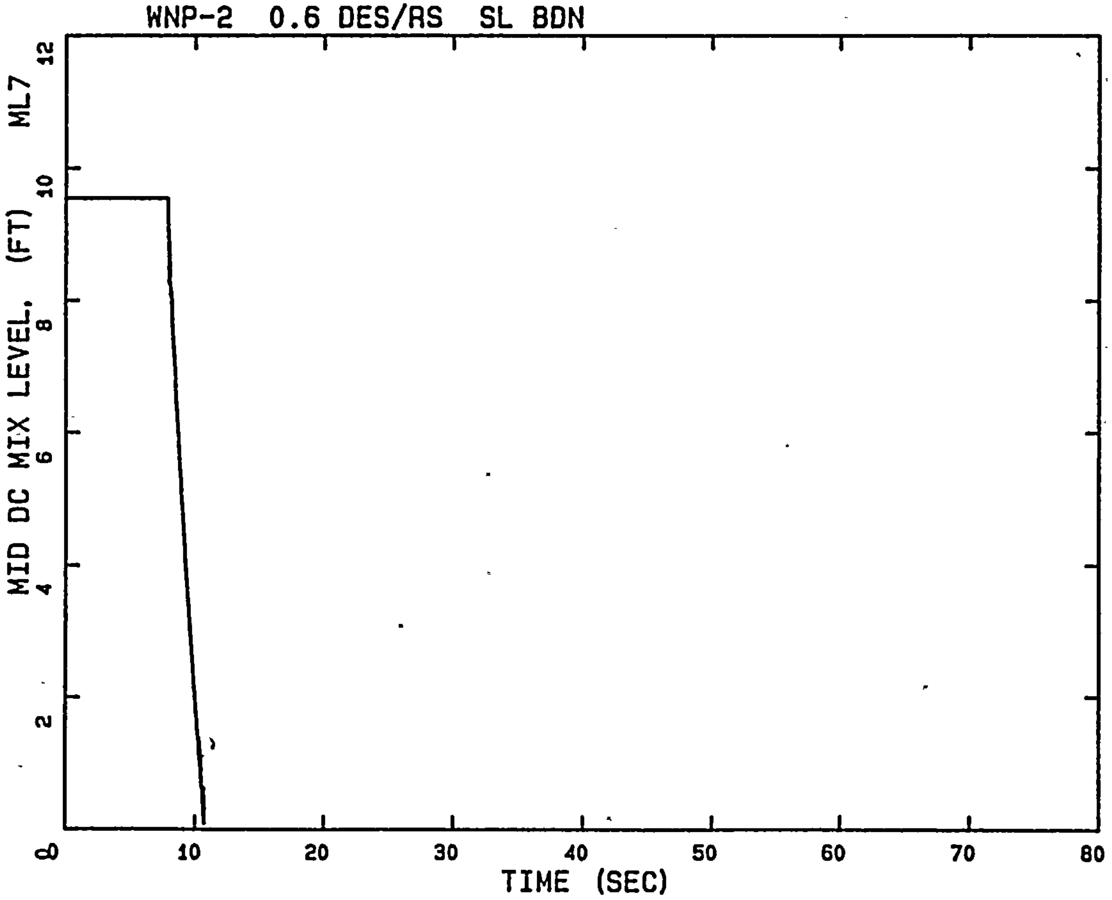

Figure 4.14 Blowdown Middle Downcomer Mixture Level

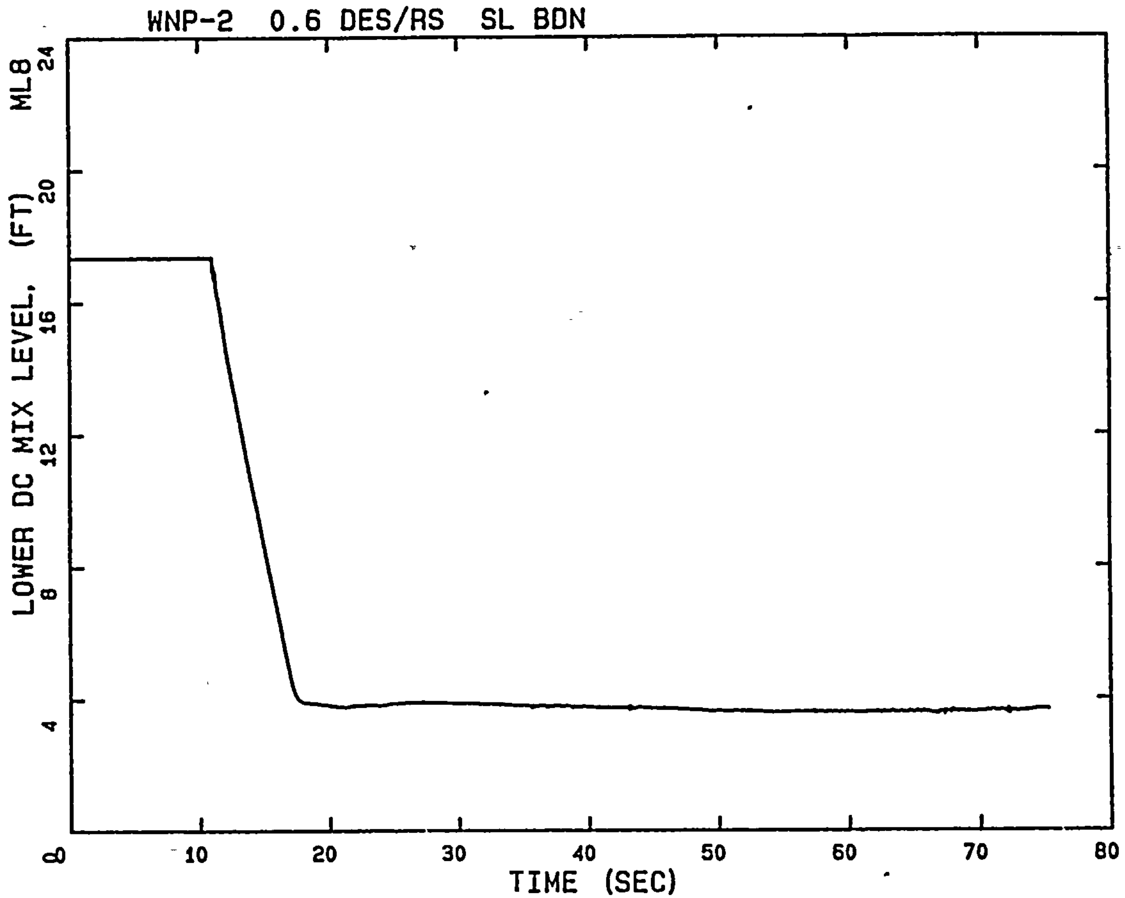

Figure 4.15 Blowdown Lower Downcomer Mixture Level

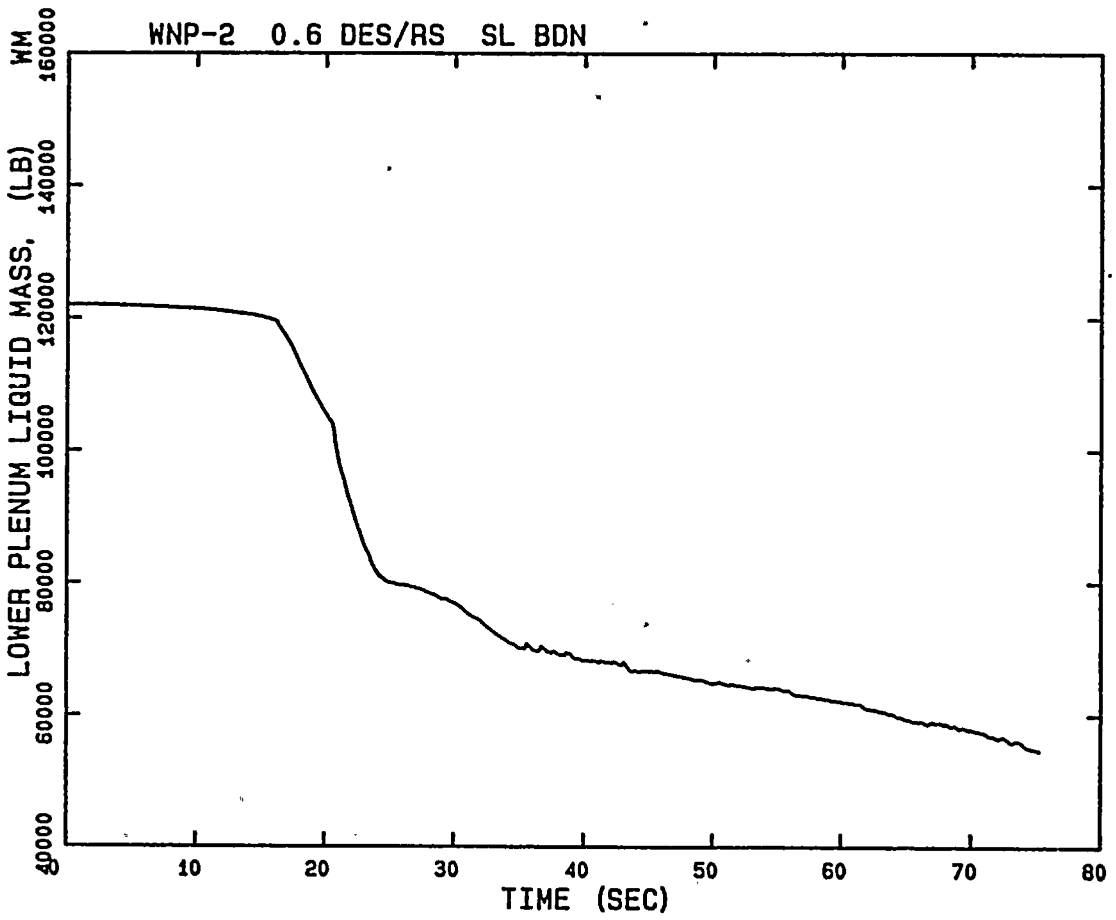

Figure 4.16 Blowdown Lower Plenum Liquid Mass

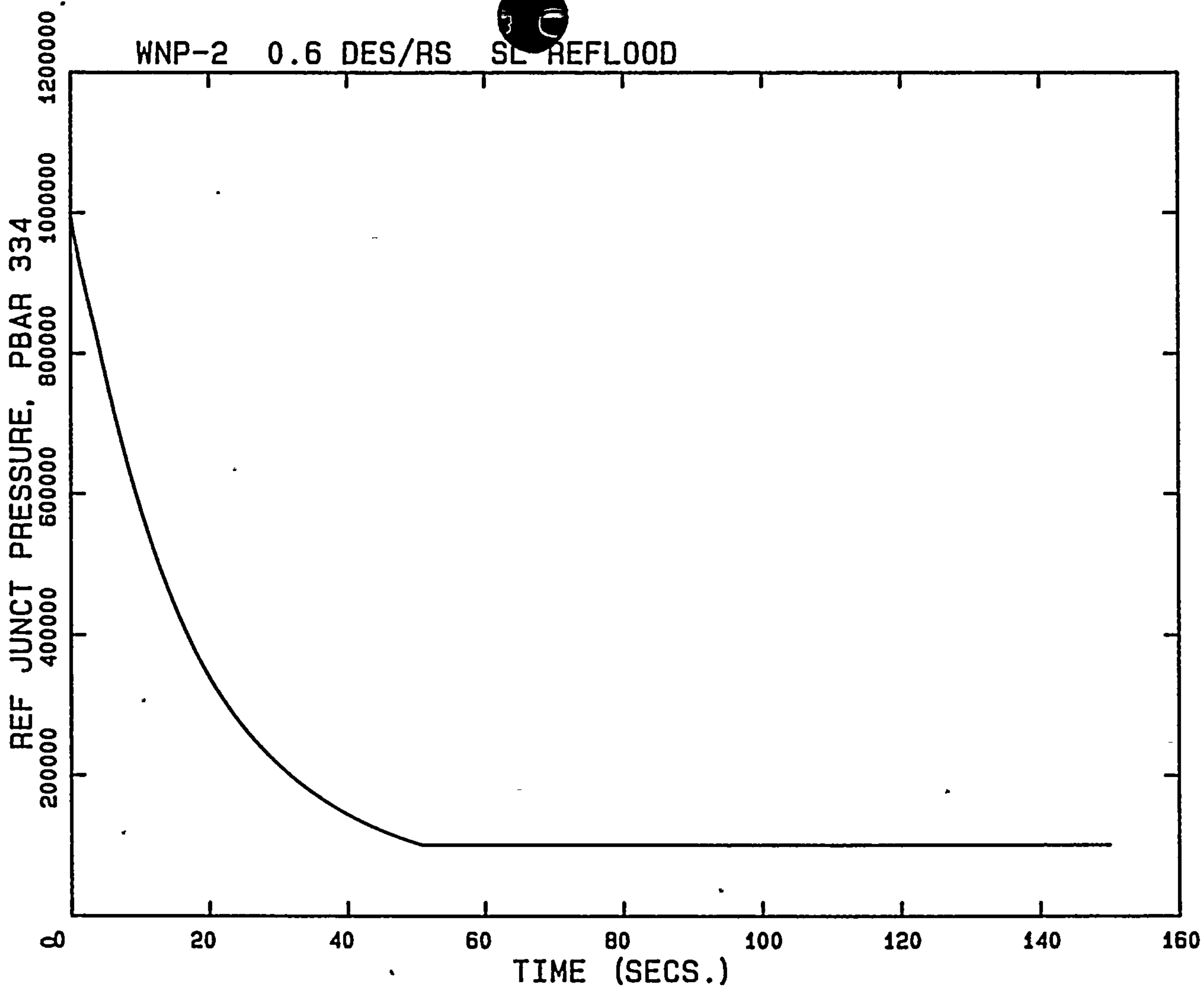

Figure 4.17 Refill/Reflood System Pressure

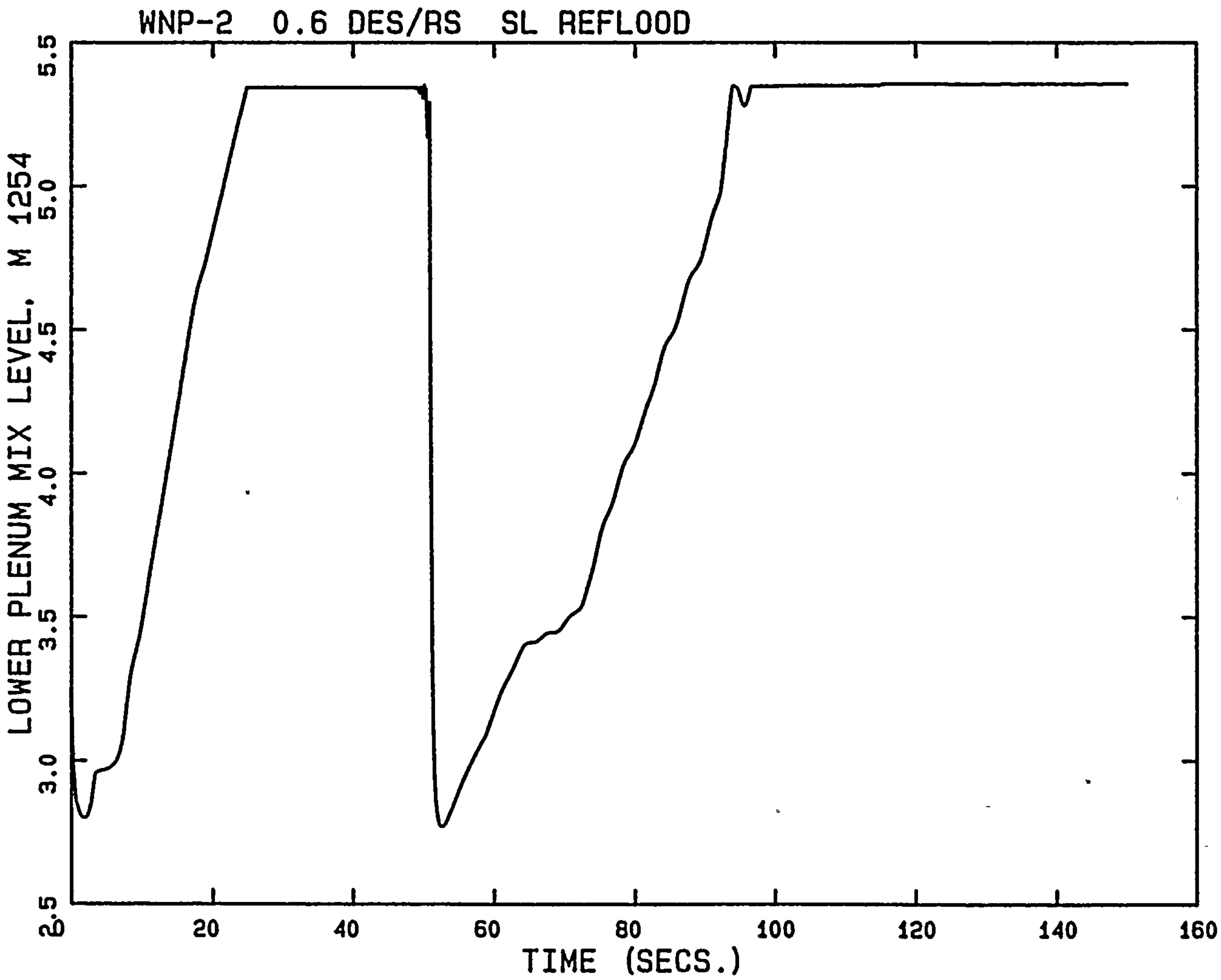

Figure 4.18 Refill/Reflood Lower Plenum Mixture Level

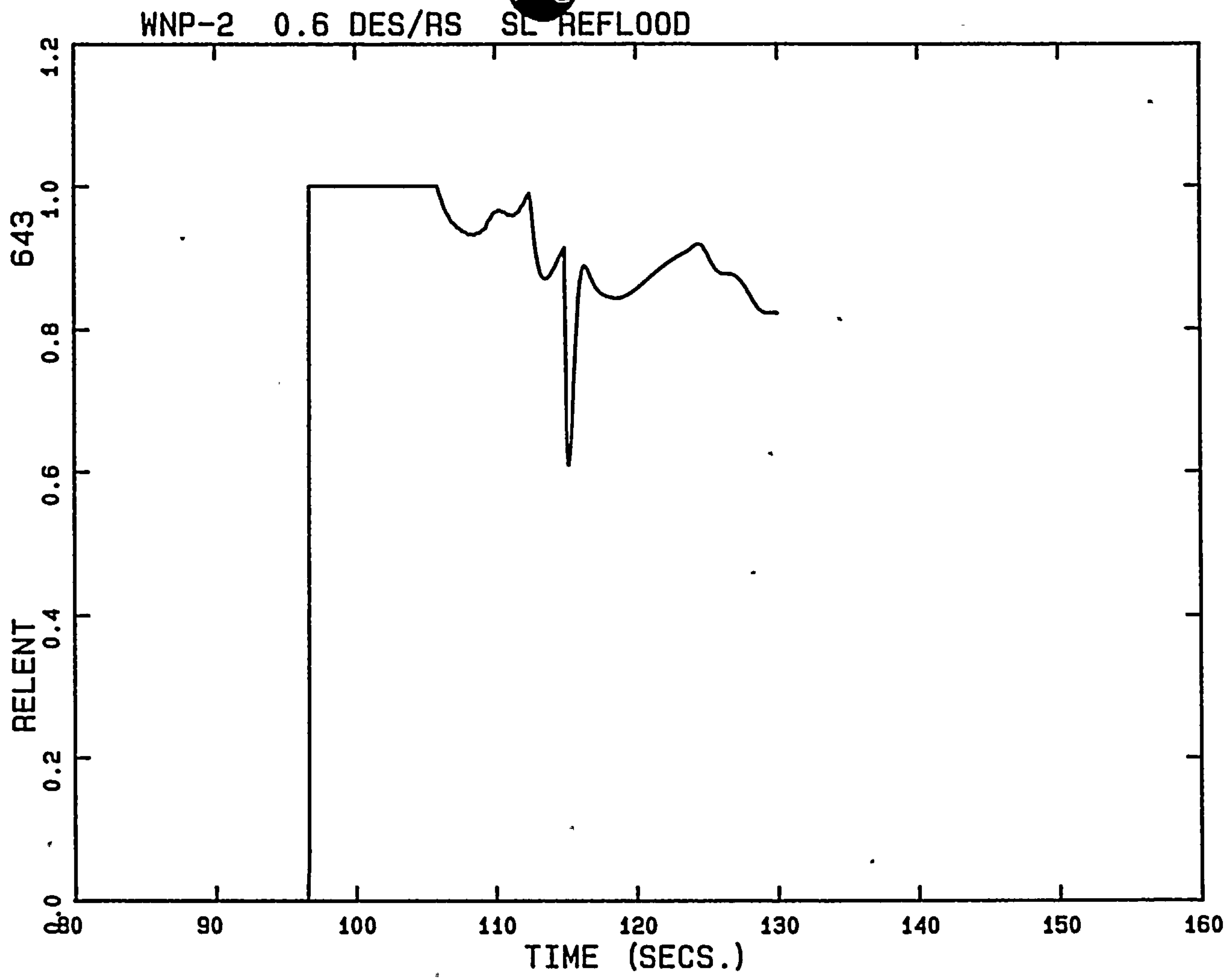

Figure 4.19 Refill/Reflood Core Midplane Relative Entrainment

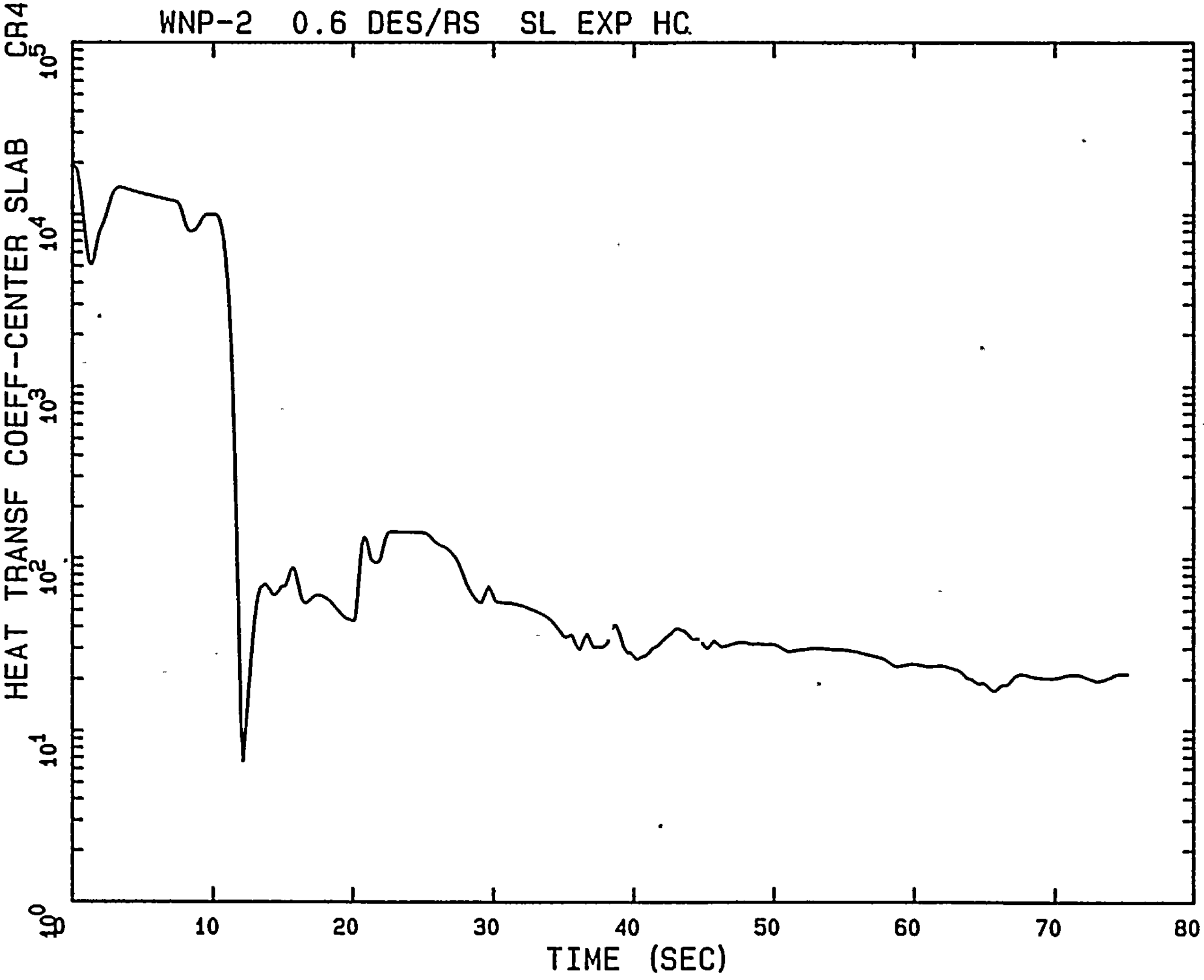

Figure 4.20 Blowdown Hot Channel Heat Transfer Coefficient

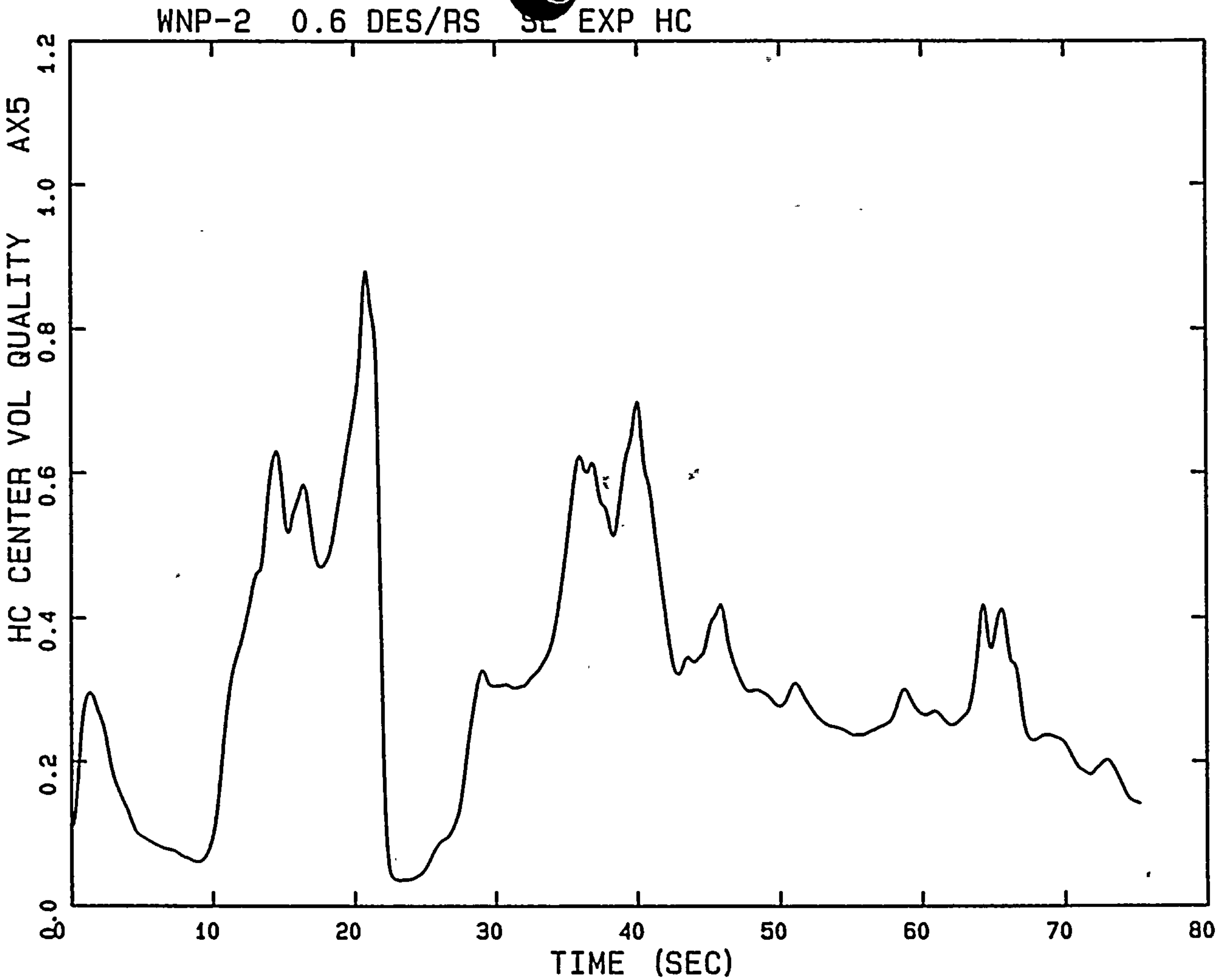

Figure 4.21 Blowdown Hot Channel Center Volume Quality

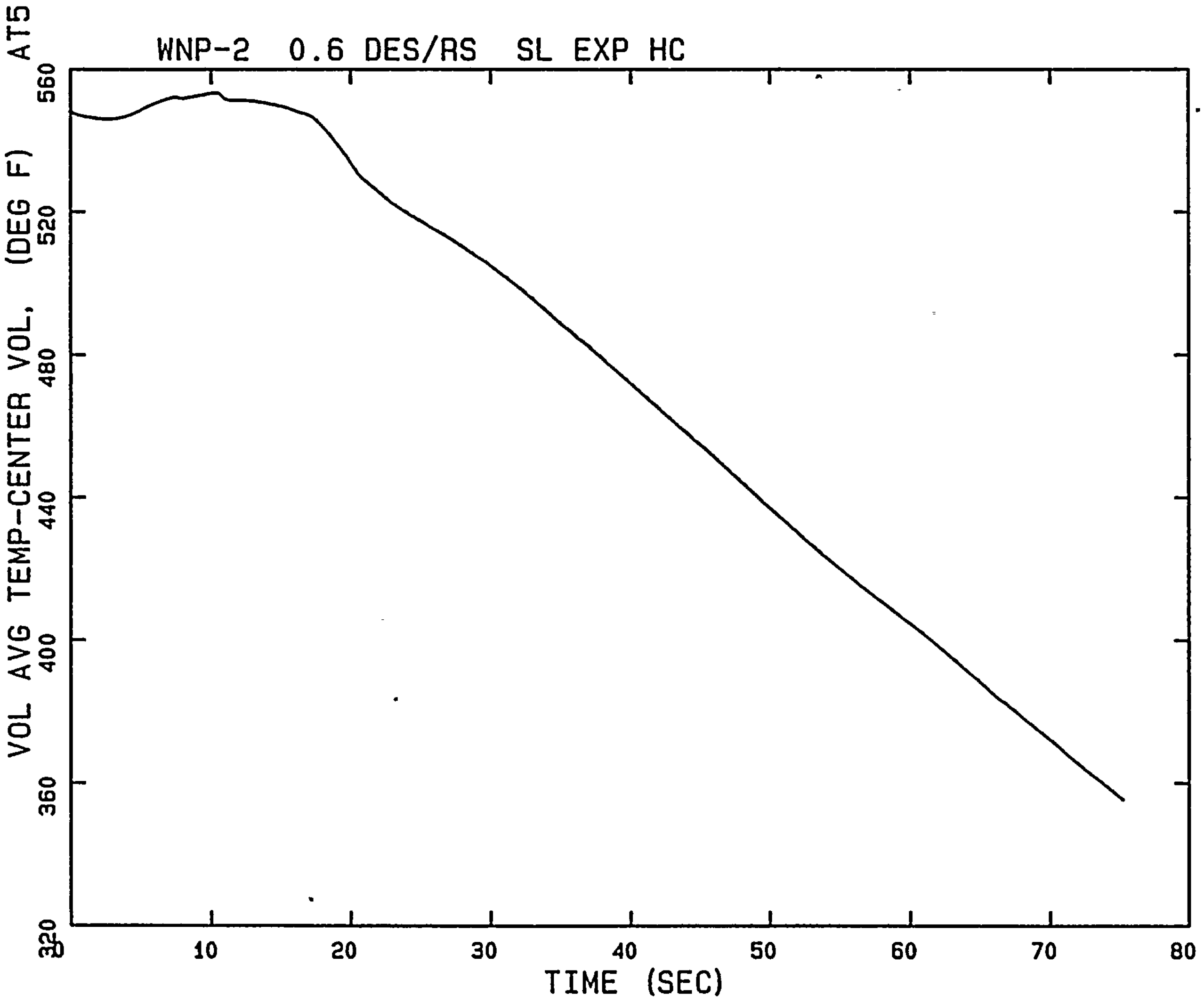

Figure 4.22 Blowdown Hot Channel Center Volume Coolant Temperature

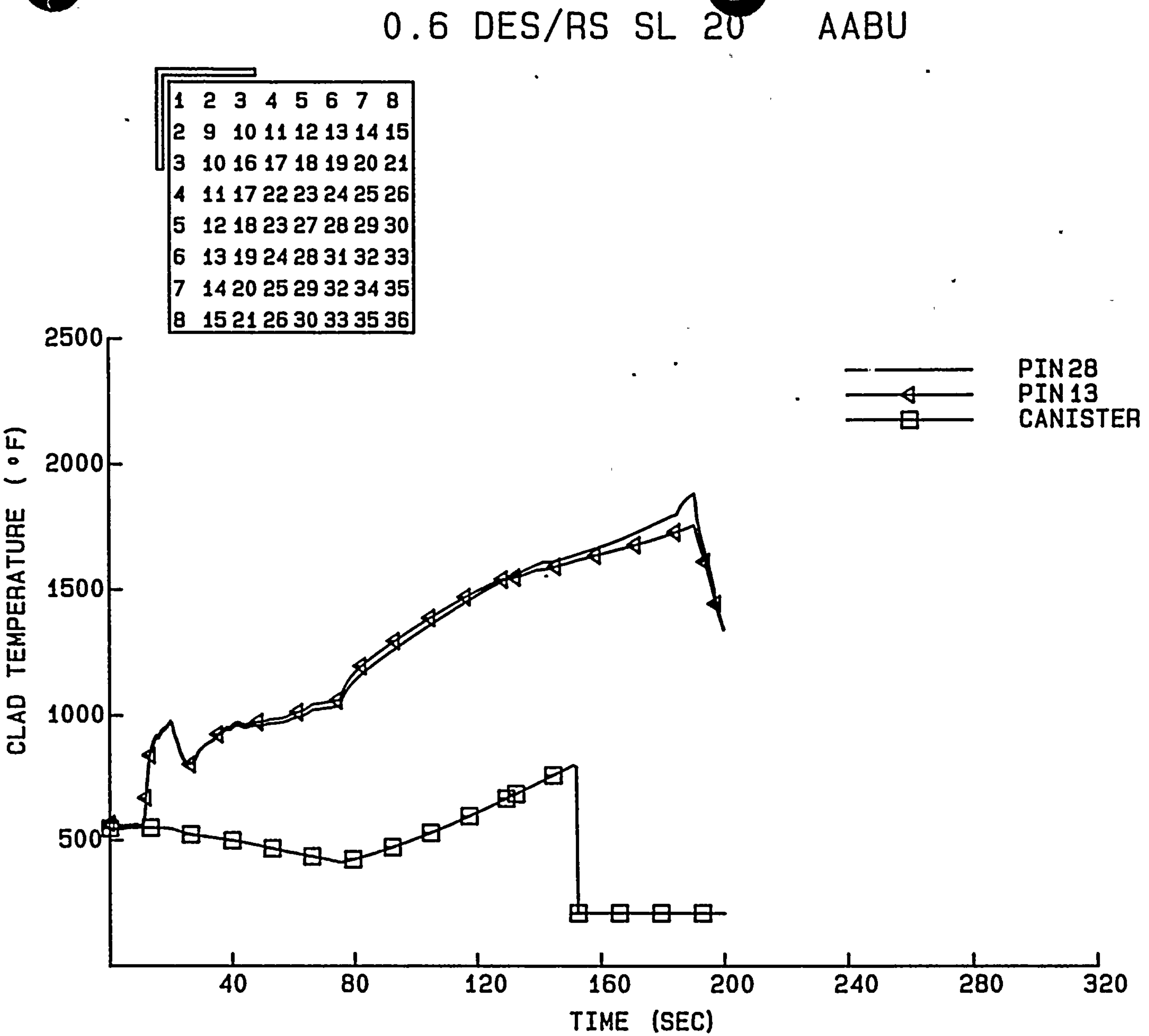

Figure 4.23 Typical Hot Assembly Heatup Results For ANF Fuel (20 GWd/MTU)

5.0 REFERENCES

1. "Generic Jet Pump BWR 3 LOCA Analysis Using the ENC EXEM Evaluation Model," XN-NF-81-71(A), Supplement 1, Exxon Nuclear Company, September 1982.

2. "Generic LOCA Break Spectrum Analysis for BWR 3 and 4 with Modified Low Pressure Coolant Injection Logic," XN-NF-84-117(P), Exxon Nuclear Company, December 1984.

3. "Acceptance Criteria for Emergency Core Cooling Systems for Light Water Cooled Nuclear Power Reactors," 10 CFR 50.46 and Appendix K of 10 CFR 50, Federal Register, Volume 39, Number 3, January 4, 1974.

4. "LOCA Break Spectrum Analysis for a BWR 5," XN-NF-85-138(P), Exxon Nuclear Company, December 1985.

5. "WNP-2 LOCA-ECCS Analysis; MAPLHGR Results," XN-NF-85-139, Exxon Nuclear Company, December 1985.

6. Letter, R. A. Vopalensky to J. B. Edgar, "Supply System Data Package No. 44," WANF-2B-87-0023, March 2, 1987.

7. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance With 10 CFR 50 Appendix K, Amendment 2-One Recirculation Loop Out-of-Service," NEDO-20566-2, Revision 1, Class 1, July 1978.

8. Letter, R. A. Vopalensky to J. B. Edgar, "Single Loop Operation Contract No. 2808-2B," WPEN-2B-85-0075, July 11, 1985.

9. "RODEX2: Fuel Rod Thermal-Mechanical Response Evaluation Model," XN-NF-81-58(A), Revision 2, Exxon Nuclear Company, February 1983.

10. "RELAX: A RELAP4-Based Computer Code for Calculating Blowdown Phenomena," XN-NF-80-19(P), Volume 2B, Revision 1, Exxon Nuclear Company, June 1981.

11. "FLEX: A Computer Code for the Refill and Reflood Period of a LOCA," XN-NF-80-19(P), Volume 2B, Revision 1, Exxon Nuclear Company, June 1981.

12. "HUXY: A Generalized Multirod Heatup Code with 10 CFR 50 Appendix K Heatup Option - User's Manual," XN-CC-33(A), Revision 1, Exxon Nuclear Company, November 1, 1975.

13. "BULGEX: A Computer Code to Determine the Deformation and the Onset of Bulging of Zircaloy Fuel Rod Cladding," XN-74-21, Revision 2, and XN-NF-27, Revision 2, Exxon Nuclear Company, December 31, 1974.

ANF-87-118
Issue Date: 9/11/87

WNP-2 LOCA ANALYSIS FOR SINGLE LOOP OPERATION

Distribution

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 18, 1988

MEMORANDUM TO: John W. Craig, Acting Chief
Plant Systems Branch
Division of Engineering & System Technology

Conrad E. McCracken, Acting Chief
Chemical Engineering Branch
Division of Engineering & System Technology

Faust Rosa, Chief
Electrical Systems Branch
Division of Engineering & System Technology

THROUGH: George W. Knighton, Director
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

FROM: Robert Samworth
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

SUBJECT: WNP-2 FIRE INSPECTION

Charles Ramsey of Region V has initiated a team inspection of fire protection issues at WNP-2 to take place June 6 through June 10. The inspection will have the primary objective of closing open items. The inspection will follow Inspection Module 64100, titled "Postfire Safe Shutdown, Emergency Lighting and Oil Collection Capability at Operating and Near-Term Operating Reactor Facilities" and Module 64704, "Fire Protection/Prevention Program." Where possible, the inspection should close out issues identified (e.g., in previous inspections) and reviewed by the staff.

I have enclosed the draft inspection plan prepared by Mr. Ramsey, along with the source documents listed in the plan. I have also enclosed the schedule for critical events in the inspection. Please review and comment on the inspection plan as appropriate with the objective of focusing the inspection for efficient utilitization of the limited manpower being made available.

We would like to finalize the inspection plan this month to give adequate time for collecting necessary documents for preparation for the inspection. Therefore, please provide your input by April 25th.

Robert B. Samworth

Robert B. Samworth, Senior Project Manager
Project Directorate V
Division of Reactor Projects - III,
IV, V and Special Projects

WNP 2 FIRE INSPECTION SCHEDULE (TENTATIVE)

April 8th	Staff input due back to Ramsey
April 18th	Final Plan
April 22nd	Management Approval (Knighton)
April 25th	Announce Inspection to licensee, Tell them what documents are needed.
May 23rd - 27th	Inspection Preparation (Get info from licensee, Review FSAR, Familiarize self with plant & issues.
June 6th - 10th	Meet at site at 8 a.m. on Monday, 6/6, tour of plant, meet with licensee staff (entrance meeting)
June 10th	1 p.m. exit meeting

7/24/88 ACV

Inspection Plan for Re-inspection of WNP-2

Using Inspection Module No. 64100

Team Leader: Chuck Ramsey, Region V, FTS 463-3767

Background

Region V Inspection Report Nos. 50-397/86-05, 397/86-25, 397/87-02, and 397/87-19 document 14 open/unresolved items and one violation relating to fire protection and safe shutdown capability at WNP-2. In addition, by letter dated November 11, 1987 (G. W. Knighton, NRC, to G. C. Sorensen, SS), the NRC forwarded a Safety Evaluation Report addressing fire protection and safe shutdown capability as described in Amendment No. 37 to the WNP-2 FSAR, in which 28 open items were forwarded to the licensee with a requested response within 60 days. The licensee's resolution to these issues appears to have resulted in a significant change in the safe shutdown methodology described by the licensee in Amendment No. 19 to the WNP-2 FSAR which was reviewed and accepted by the NRC in a Safety Evaluation Report dated March, 1983 (NUREG 0892).

Furthermore, followup on Part 21 Report Nos. 86-14P, 86-14P1; LER Nos. 84-31-L6, 87-29-LO, 87-30-LO; Information Notice Nos. 61-06, 85-09; and Inspection Module No. 64704 regarding safe shutdown and routine fire protection features need to be performed.

Objective

The objective of this inspection is to obtain verification of the adequacy and acceptableness of the licensee's post fire safe shutdown physical configuration, methodology, operating procedures, and routine fire protection program implementation. The scope of the inspection is limited to areas of concern identified in one violation, 14 Region V open/unresolved items, 28 NRR

open items, two Part 21 reports, three LERs, one Information Notice, and routine Inspection Module NO. 64704.

Tasks	Scheduled Completion Date
1. One week to prepare for inspection by reviewing background material designated or provided by the inspection team leader and provide input into the inspection plan.	1700 hours May 15, 1988
2. Attend entrance meeting with licensee and participate in site tour.	0800 hours June 6, 1988
3. Perform inspection of licensee corrective actions to concerns by assignment as follows	
a. Systems Engineer(s) will verify licensee's corrective actions for hot/cold shutdown concerns from inside and outside of the control room, with and without onsite power.	1700 hours June 10, 1988
b. Electrical Engineer(s) will verify licensee's corrective actions for associated circuits relative to common bus, common enclosure, spurious signal and electrical separation concerns.	1700 hours June 10, 1988
c. Fire Protection Engineer(s) will verify licensee's corrective actions	

for physical separation, fire barrier/ detection and suppression provided for safe shutdown capability concerns. In addition, verify the licensee's implementation of routine fire protection program activities as required by Inspection Module 64704.	1700 hours June 11, 1988

4. Confer with team leader as necessary and confirm basis for determinations made. — 1700 hours Daily

5. Prepare Summary of inspection findings:

a. Draft summary of results	1600 hours. June 11, 1988
b. Discuss findings as needed at exit meeting with licensee	1300 hours June 10, 1988
c. Provide handwritten work copy of findings to team leader	5 days after exit meeting
d. Provide final inspection report input to team leader	10 days after exit meeting

6. Followup on Inspection Findings

Assist in the close-out of followup items generated as a result of the inspection.	As needed.

Level of Effort and Period of Performance

The level of effort is estimated at 600 professional staff hours. The inspection is scheduled for the period June 6-10, 1988. Inspection team

members are required to arrive at the site for the entrance meeting at 0730 hours on June 6, 1988. The entrance meeting will be held at 0800 hours.

Onsite inspection effort is scheduled to terminate at 1300 hours on Friday June 10, 1988.

Meetings and Travel

All inspection team members are required to attend the entrance/exit meetings with the licensee and daily meetings (approximately one hour starting at 1700 hours).

Travel arrangements should be made to arrive at motel accommodations near the site in Richland, Washington, by Sunday, June 5, 1988. A minimum of two vehicles will be used for transportation of inspection team members at the site.

Staff Time Requirement

Inspection preparation	5 days	
Travel to and from site	2 days	
Inspection	6 days	
Weekend days on travel	2 days	
Overtime worked	1 day	(one day of overtime expected to accumulate during the week and one day on Saturday)
Documentation	10 days	

Period of Performance

Preparation/in-office review	(1 week)	May 23-May 27
Onsite inspection	(5 days)	June 6-June 10
Documentation	(2 weeks)	June 14-June 28

Team Assignments

A. System Engineer(s)

1. LER No. 84-31 Lo-L6 - Appendix R Deficiencies
2. Unresolved Item 86-25-05 - Safe Shutdown Methodology
3. Unresolved Item 86-25-06 - Safe Shutdown Procedures
4. Unresolved Item 87-19-07 - Loss of All AC Power Due to Fire
5. Unresolved Item 87-19-16 - Use of ADS Inhibit Switch
6. Unresolved Item 86-25-14 - Emergency Lighting Support for S.S.
7. Unresolved Item 86-05-06 - Worst Case Fire Analysis for CR and CSR - need
8. NRR Question No. 9 - Emergency Lighting
9. NRR Question No. 10 - Number of ADS Values Needed for S.S.
10. NRR Question No. 12 - Use of Both Divisions for S.S.
11. NRR Question No. 13 - Equipment Needed for S.S.
12. NRR Question No. 16 - Testing of S.S. Components
13. NRR Question No. 17 - Use of Installed Transfer Switches
14. NRR Question No. 21 - SM-8 Cabinet Transfer Switches
15. NRR Question No. 18 - Operability of Six SRVs
16. NRR Question No. 19 - Process Instrumentation
17. NRR Question No. 20 - Required Repairs for S. S.
18. NRR Question No. 23 - RHR V-8 and V-9 LOCA Prevention
19. NRR Question NO. 26 - Minimum Staffing Needed for S.S.
20. T.I. 2515-61 - Emergency Lighting
21. T.I. 2515-62 - Appendix R, S.S.

B. Electrical Engineer(s)

1. LER No. 84-03 Lo-L6 - Appendix R Deficiencies
2. Information Notice 86-106 - Interaction Between Fire Protection and Security Systems
3. Information Notice 85-09, Isolation Transfer Switches
4. Unresolved Item 86-05-04 - Common Enclosure Analysis
5. Unresolved Item 86-25-07 - Associated Circuits Analysis
6. Unresolved Item 86-25-08 - Hi-Low Pressure Interface Analysis
7. Unresolved Item 87-19-07 - Loss of All Sources of AC Power
8. Unresolved Item 86-05-06 - Worst Case Fire Analysis for CR and CSR
9. NRR Question No. 8 - Conformance with Regulatory Guide 1.75
10. NRR Question No. 11 - Conformance with Appendix R and BTP 9.5-1
11. NRR Question No. 12 - Protection of Both Divisions for S.S.
12. NRR Question No. 16 - Testing of S.S. Components
13. NRR Question No. 17 - Testing of Installed Transfer Switches
14. NRR Question No. 21 - SM-8 Cabinet Transfer Switches
15. NRR Question No. 22 - Coordinated Circuit Protection
16. NRR Question No. 23 - Power to RHR Valves V-8 and V-9
17. NRR Question No. 24 - Divisional Power to RHR Valves V-8 and V-9
18. NRR Question No. 25 - 3 Phase Faults at Hi-Lo Pressure Interfaces
19. T.I. 2515-62 - Appendix R., S.S.

C. Fire Protection Engineer(s)

1. LER 84-031 Lo-L6 - Appendix R Deficiencies
2. LER 87-29 Lo - Fire Rated Floor Penetrations
3. LER 87-30 Lo - Unqualified Fire Wall
4. Part 21 Report 86-14 P - Automatic Sprinkler Model C Valves
5. Part 21 Report 86-14 P1 - Automatic Sprinkler Model C Valves
6. Information Notice 61-06 - Interaction of Fire Protection and Security Systems
7. Unresolved Item 87-02-01 - Unqualified Barriers for Protection of S.S. Components
8. Unresolved Item 86-05-06 - Worst Case Fire Analysis for CR and CSR
9. Unresolved Item 86-25-03 - Fire Main Beneath Safety-Related Structures
10. Unresolved Item 86-25-04 - Deficiencies Not Documented in LER 84-031
11. Unresolved Item 86-25-11 - CSR Design
12. Unresolved Item 86-25-13 - Fire Detection System Design
13. Unresolved Item 86-25-14 - Emergency Lighting
14. Unresolved Item 87-19-07 - Loss of All AC Power Due to Fire
15. Unresolved Item 87-19-08 - Fire Threat to Shutdown Division and Oil Transfer Pump Rooms
16. NRR Question No. 1 - Combustible Inventory Fire Area TG-1
17. NRR Question No. 2 - Bus Duct Penetration Fire Barriers
18. NRR Question No. 3 - Closure of Fire Dampers Under Air Flow
19. NRR Question No. 4 - Protection of Instrument Sensing Lines
20. NRR Question No. 5 - Separation of Redundant Trains in Seismic Gap
21. NRR Question No. 6 - Inspection of Water Control Valves
22. NRR Question No. 7 - Holan Cylinder Storage Procedures
23. NRR Question No. 9 - Emergency Lighting
24. NRR Question No. 11 - Appendix R and BTP 9.5-1 Conformance
25. NRR Question No. 12 - Protection of Both Divisions for S.S.
26. NRR Question No. 15 - Protection for Division II SRVs
27. NRR Question No. 27 - Regulatory Guide 1.39 Procedure Conformance
28. NRR Question No. 28 - NFPA Administrative Procedure Conformance

29. Module 64704 - Routine Program Implemention
30. T.I. 3515-61 - Emergency Lighting
31. T.I. 2515-62 - Appendix R Safe Shutdown

D. Review Materials Required

1. Inspection Reports

- 397/86-05 ✓
- 397/86-25 ✓
- 397/87-02 ✓
- 397/87-19 ✓

2. Licensee Event Reports (LERs)

- 84-31 Lo-L6 ✓
- 87-29 Lo ✓
- 87-30 Lo ✓

3. Information Notices

- 61-06
- 85-09

4. Part 21 Reports

- 86-14 P
- 86-14 P1

5. Safety Evaluation Reports (SERs)

- NUREG 0892
- SER attached to letter dated November 11, 1987 ✓

6. Licensee's Documents

- Associated Circuits Analysis
- Safe Shutdown Equipment List
- January 11, 1988 Response to NRR Open Items
- Responses to Inspection Report Open Items