Attachment 6

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ANALYSIS OF OPERATION WITH ONE SAFETY/RELIEF VALVE OUT-OF-SERVICE

FOR

FITZPATRICK NUCLEAR POWER PLANT

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ABSTRACT

The purpose of this report is to provide the technical basis for operation of the FitzPatrick Nuclear Power Plant with one of the safety/relief valves (S/RVs) out-of-service.

With one S/RV out-of-service, there will be no impact on the transient parameters that influence critical power ratio (ΔCPR) and no change in the calculated ΔCPR. The peak vessel pressure for the main steam isolation valve (MSIV) flux scram event is well below the ASME code upset limit of 1375 psig. This analysis has demonstrated that these conclusions are valid for current GE fuel types utilized in the FitzPatrick plant.

Attachment 6

1. INTRODUCTION

The purpose of this report is to provide the technical bases for operation of the FitzPatrick Nuclear Power Plant with one of the 11 safety/relief valves (S/RVs) out-of-service.

The potential effect of one S/RV out-of-service is to change the pressure response of the reactor during transients and postulated accidents. This could conceivably impact the margins or safety limits for plant operation.

Accident and transient considerations for operation with one S/RV out-of-service are presented. The overall conclusion is given in Section 4.

2. LOSS-OF-COOLANT ACCIDENT

2.1 GENERAL DISCUSSION

FitzPatrick has 11 S/RVs. All of these valves have a pressure-actuated safety function and seven of them have an additional pneumatically operated relief function automatically actuated by the Automatic Depressurization System (ADS). For loss-of-coolant accident (LOCA) evaluations, credit is taken for the safety function; however, not all of the S/RVs are actuated during a LOCA..

If the out-of-service valve is one of the S/RVs with the ADS function, there can be a potential impact on the calculated peak cladding temperatures (PCT) for small break sizes of less than approximately 0.2 ft^2. This may occur because, with a worst case postulated single failure of the High Pressure Coolant Injection System (HPCI), the small break response is affected by the time required to depressurize the reactor to the operating pressure of the low pressure Emergency Core Cooling System (ECCS).

For larger postulated break sizes, the blowdown itself depressurizes the reactor vessel rapidly before the ADS actuates, and the number of S/RVs or actuation of the ADS has an inconsequential effect on the calculated PCT.

For the FitzPatrick plant, the limiting LOCA is a large break (greater than 1.0 ft^2 in size), and the loss of the S/RV or ADS function has no effect on the calculated maximum average planar linear heat generation rate (MAPLHGR) limit.

2.2 PLANT SPECIFIC ANALYSIS

A plant specific LOCA analysis has already been performed (Reference 1), using the approved Appendix K methods to determine the increase in small break PCT with one S/RV with an ADS function assumed to be out-of-service. The results of this analysis are repeated in this report.

The results of the analysis are given in Table 1. The most limiting small break in terms of PCT is the 0.07 ft^2 recirculation suction line break. With one ADS out-of-service, the PCT is less than 1300°F, which is over 900°F below the 2200°F limit. This analysis was performed assuming the most limiting fuel type and exposure. The water level, pressure, and PCT for the worst case small break are shown in Figures 1 and 2.

2.3 CONCLUSIONS

With one S/RV out-of-service, there is no impact on the calculated MAPLHGR limits for FitzPatrick, even if the out-of-service S/RV has the ADS function. This conclusion is valid for all current GE fuel types utilized in the FitzPatrick plant.

3. TRANSIENTS

3.1 GENERAL DISCUSSION

Operation with one S/RV out of service can affect the system response in the event of an abnormal operating transient. The decreased relief capacity can lead to higher transient pressures, which could affect the change in critical power ratio (ΔCPR) and the ASME code overpressure limits.

The following limiting abnormal operational transients were evaluated with one S/RV out-of-service using the NRC approved ODYN computer code for transient analysis:

a. Load rejection without bypass at 104% power/100% flow, to evaluate the ΔCPR.

b. Main steam isolation valve (MSIV) closure, flux scram, to evaluate the ASME code overpressure protection event.

The S/RV setpoints and groupings for the analysis of these events are presented in Table 2. Note that the S/RV in the lowest setpoint group is conservatively assumed to be out-of-service in this analysis.

To justify that the conclusions of this analysis are cycle independent, transient analyses were performed using Cycle 5, Cycle 6 and an extrapolated end-of-equilibrium cycle (EOEC) nuclear data.

3.2 LIMITING ΔCPR EVENT, LOAD REJECTION WITHOUT BYPASS

The transient analysis results for normal operation are reported in References 2 and 3 for Cycle 5 and Cycle 6 operation. The load rejection without bypass (LRNB) is the limiting ΔCPR event. A base case for the EOEC nuclear data was evaluated to assess the sensitivity of the analysis to cycle-dependent nuclear parameters.

The LRNB transient was performed with one S/RV out-of-service. The peak neutron flux, the peak heat flux, and the minimum critical power ratio (MCPR) remain unchanged for this event. This is because both the peak neutron heat and the peak heat flux occur before the S/RVs are actuated during this event. Therefore, the effect of one S/RV out-of-service has no impact on the ΔCPR of the limiting transient LRNB event. While the time of peak neutron heat flux can be affected by the fuel nuclear parameters, there is sufficient margin between the peak time and actuation of the SRVs to assure that there is no effect on ΔCPR.

The results of the transient analyses are summarized in Table 3 and the time response of this transient is shown in Figure 3 for a typical case. There was no change in the ΔCPR for the cases considered.

The non-pressurization events (e.g., rod withdrawal error) are independent of valve setpoint and valve capacity, therefore, the ΔCPR values are unchanged as a result of one S/RV out-of-service.

3.3 MSIV FLUX SCRAM

The adequacy of the S/RV capacity based on ASME code requirements is demonstrated by the MSIV closure transient with high flux scram. The peak vessel pressure for this event increases a maximum of 15 psi as a result of one S/RV out-of-service, resulting in a peak pressure of 1290 psig for Cycle 5. The Cycle 6 and EOEC cases show an increase in peak vessel pressure of 12 psi, with a maximum pressure of 1268 psi. Therefore, there is a large margin to the ASME code limit of 1375 psig.

The output parameter data for the MSIV flux scram transient considered in this analysis is summarized in Table 4. The time response of key variables for this transient is shown in Figure 4 for a typical case.

3.4 CONCLUSION

With one S/RV out of service, there will be no impact on the transient parameters that influence ΔCPR and no change in the calculated ΔCPR. The peak vessel pressure for the MSIV flux scram event is well below the ASME code upset limit of 1375 psig. This analysis has demonstrated that these conclusions are valid for current GE fuel types utilized in the FitzPatrick plant.

4. OVERALL CONCLUSION

The operation of the FitzPatrick Nuclear Plant at full power with one S/RV out-of-service will have no impact on operating limits. This conclusion is valid for current General Electric fuel types, operating strategies and analysis methods, as applied to the FitzPatrick Nuclear Plant.

5. REFERENCES

1. FitzPatrick Nuclear Power Plant, Cycle 5, Analysis for Operating with One Safety/Relief Valve Out of Service. NEDO-22226, September 1982.

2. Supplemental Reload Licensing Submittal for FitzPatrick Nuclear Power Plant Reload 4, Y1003J01A25, August 1981.

3. Supplemental Reload Licensing Submittal for FitzPatrick Nuclear Power Plant Reload 5, Y1003J01A56, March 1983.

Table 1

FITZPATRICK LOSS-OF-COOLANT ACCIDENT ANALYSIS
ONE ADS VALVE OUT-OF-SERVICE

Recirculation Line Break Upstream of Discharge Valve.
System Failed: HPCI
Systems Remaining: 2 LPCS + 2 LPCI + 6 ADS Valves[a]

Break Size (ft^2)	Uncovery Time (sec)	Reflooding Time (sec)	Peak Cladding Temperature (°F)
0.05	324.6	412.4	1103
0.07	265.6	372.4	1271
0.10	232.1	330.5	1241

[a]Two of the LPCI systems inject into the broken loop and it is conservatively assumed that all of the injected water is lost through the break.

Table 2

S/R VALVE SETPOINT AND GROUPING

Previous S/RV Grouping		New (One S/RV Out-of-Service) Grouping	
Setpoint (psig)	S/RV No. Target Rock	Setpoint (psig)	S/RV No. Target Rock
1090 + 1%	2	1090 + 1%	1
1105 + 1%	2	1105 + 1%	2
1140 + 1%	7	1140 + 1%	7

Table 3

RESULTS OF LOAD REJECTION WITHOUT BYPASS ANALYSIS

Case	Power/Flow (%)	Peak Neutron Flux (%)	Peak Heat Flux (%)	Time to Peak Heat Flux (sec)	Time for First S/RV to Open (sec)
Cycle 5	104/100	653.1	125.17	1.0337	1.3302
Cycle 6	104/100	623.1	127.80	1.0103	1.3126
EOEC	104/100	596.1	127.6	1.0249	1.3116

Table 4

RESULTS OF MSIV FLUX SCRAM TRANSIENT ANALYSIS

Case	Power/Flow (%)	Peak Vessel Pressure (psig)	Change in Pressure Margin (psi) (Relative to Base Case)
Cycle 5 Base Case	104/100	1275	
Cycle 5 1 S/RV Out of Service	104/100	1290	- 15
Cycle 6 Base Case	104/100	1256	
Cycle 6 1 S/RV Out of Service	104/100	1268	- 12
EOEC Base Case	104/100	1256	
EOEC 1 S/RV Out of Service	104/100	1268	- 12

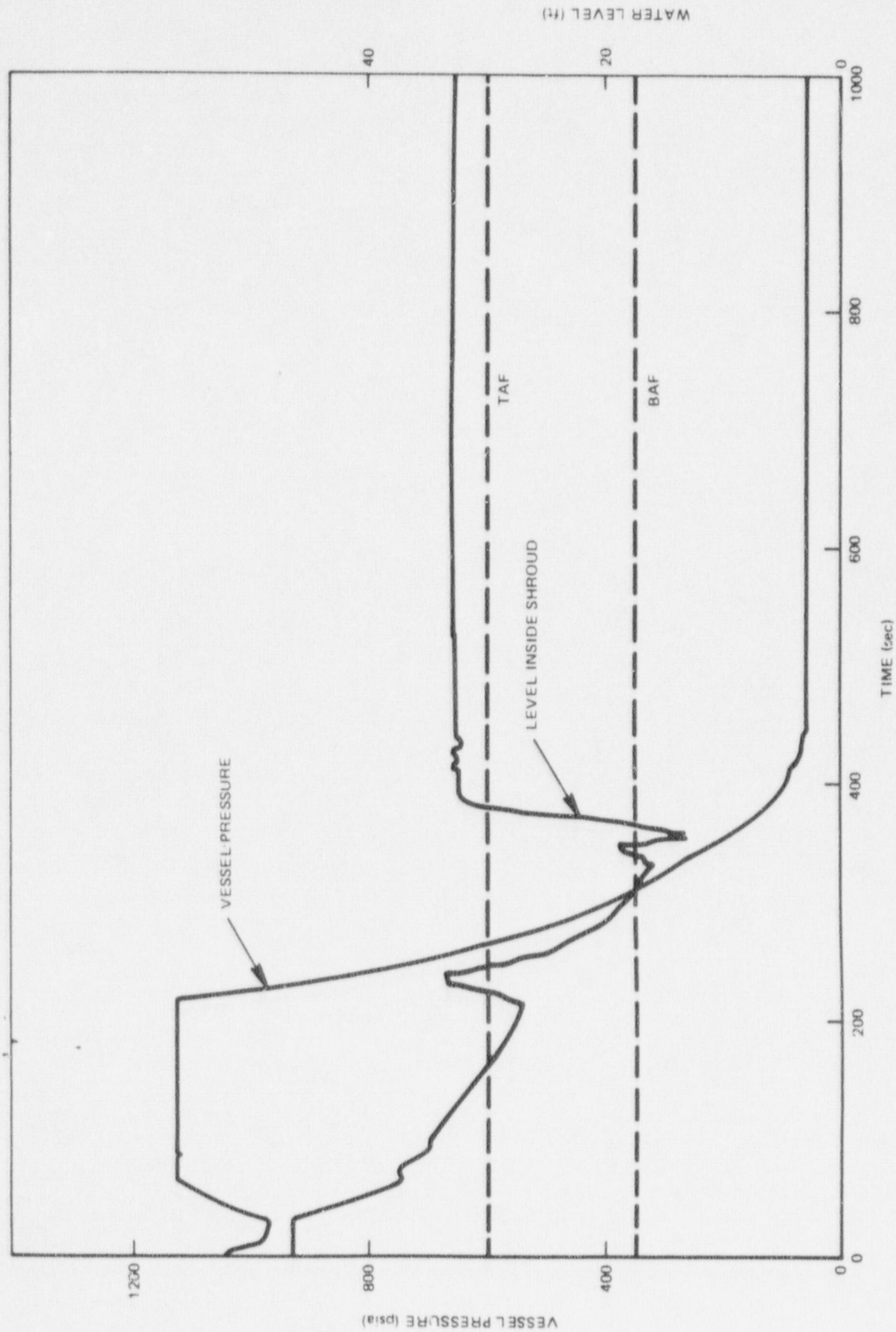

Figure 1. Water Level Inside the Shroud and Reactor Vessel Pressure Following a Small Recirculation Line Break, HPCI Failure, One ADS Valve Out-of-Service, Break Area = 0.07 ft^2

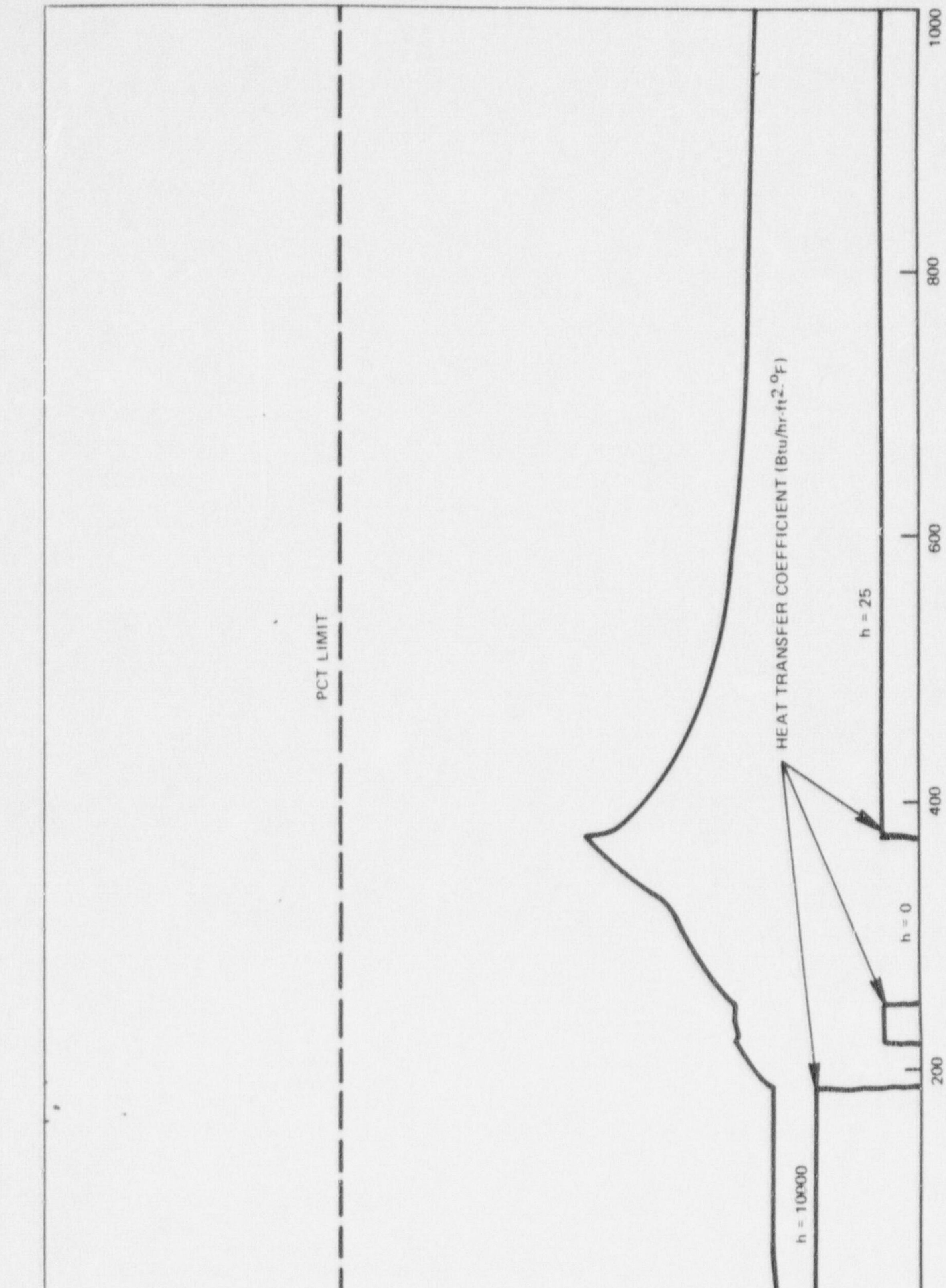

Figure 2. Peak Cladding Temperature Following a Small Recirculation Line Break, HPCI Failure, One ADS Valve Out-of-Service, Break Area = 0.07 ft^2

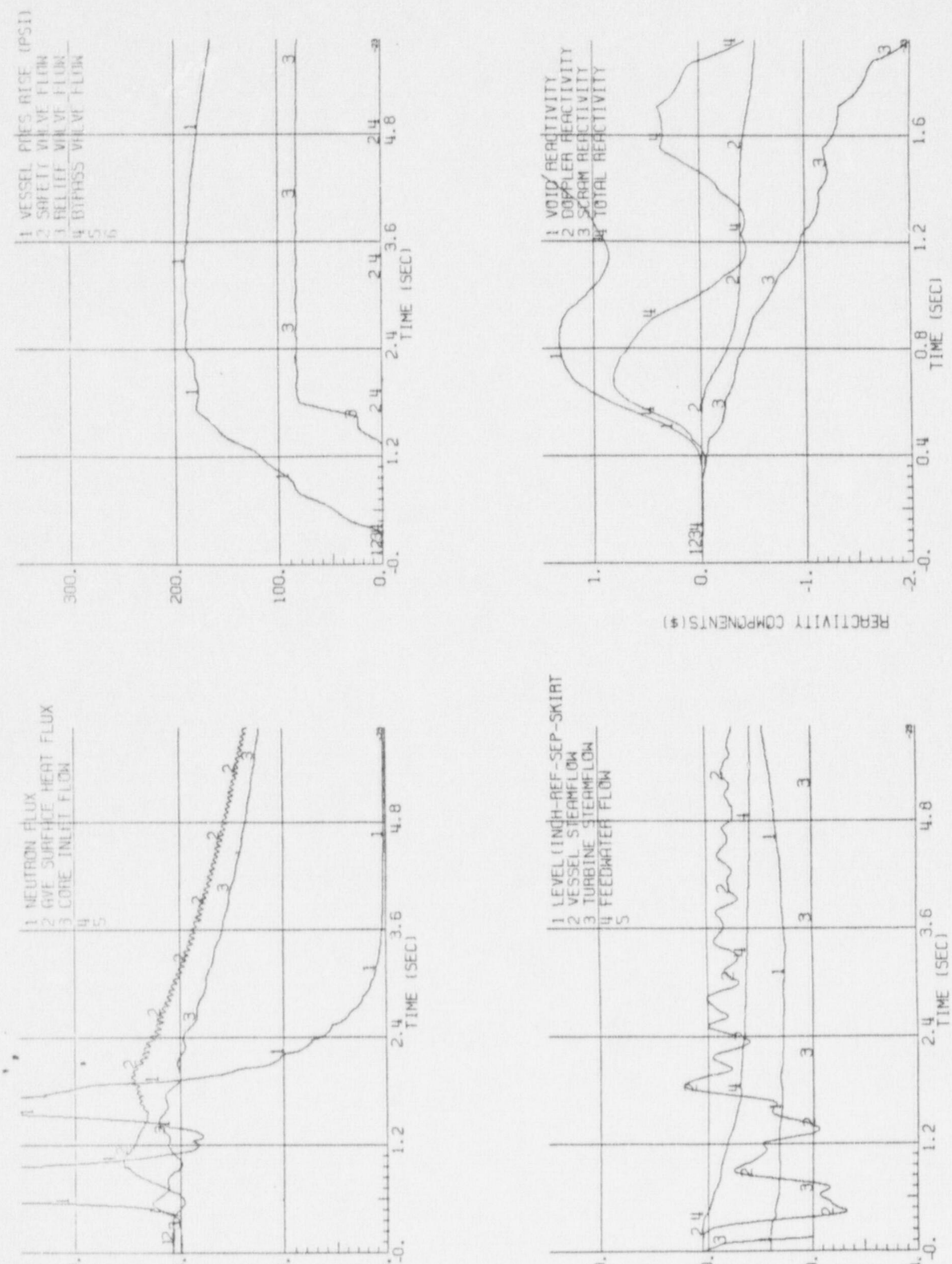

Figure 3. Plant Response to Generator Load Rejection Without Bypass

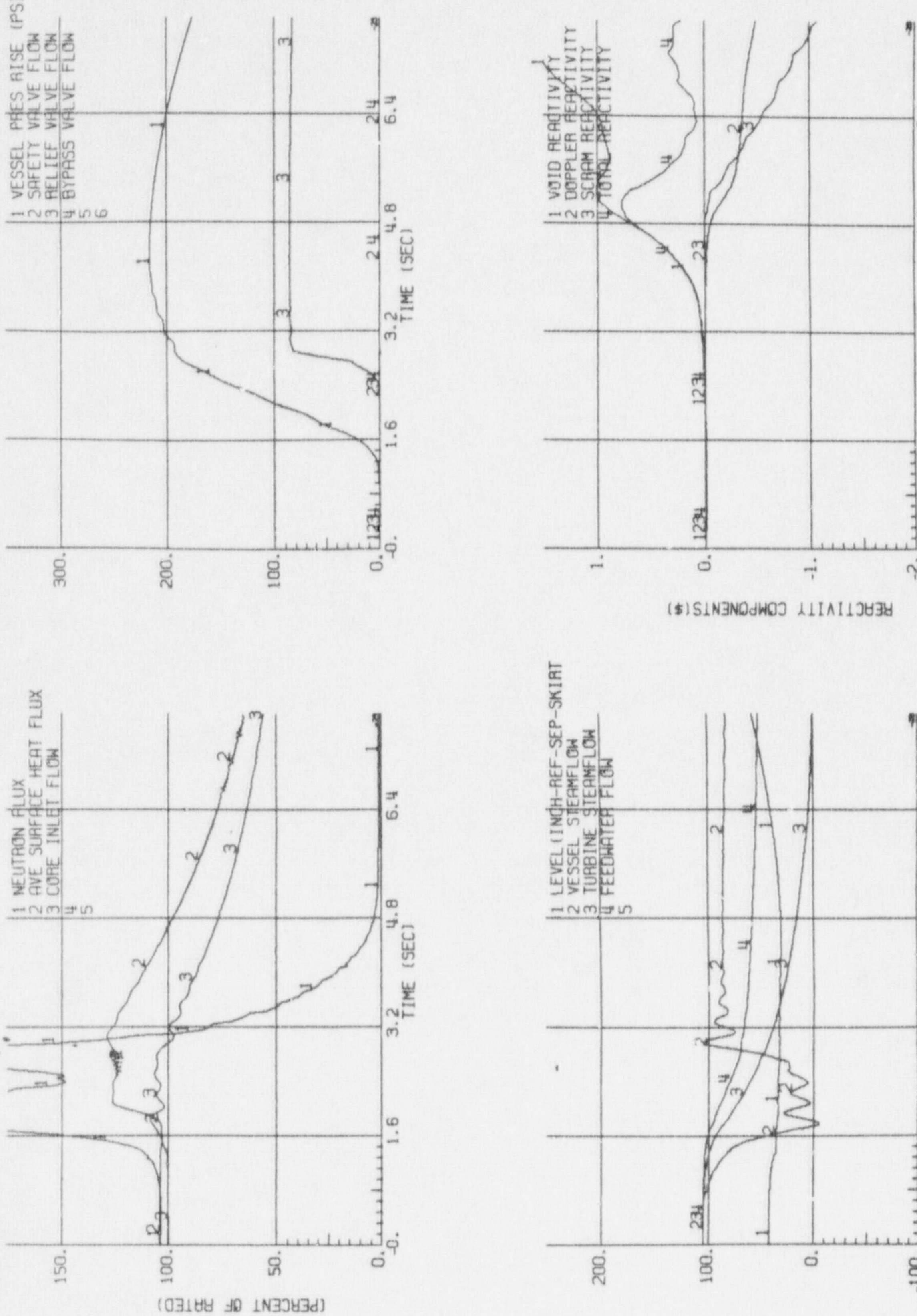

Figure 4. Plant Response to MSIV Closure (Flux Scram)

Attachment 7

EVALUATION OF SRV OPERATION DURING THE JULY 19, 1985 SCRAM

The plant transient commenced at 14:33:17 on July 19, 1985 following a problem associated with the transfer of electrical buses. A turbine trip and reactor scram occurred at 14:33:58. MSIV's closed at 14:34:02. MSIV closure isolated the reactor from bypass valve pressure regulation. Since the reactor had been operating at 100% power, decay heat removal and pressure control is accomplished by the SRVs. There are three instrument systems that monitor SRV operation. The systems rely on secondary effects like noise or temperature. Temperature response is slow and provides no additional information on subsequent SRV operations. The sonic monitor provides fast response and continuous monitoring but may provide erroneous data from crosstalk. The quality of instrumentation made the analysis very difficult. The reactor level/ pressure chart was used to analyze the transient. This is a dual speed recorder with chart speeds of one inch per hour and one inch per minute. The chart speed is changed manually. The shift to fast speed was done about eight minutes after the scram.

Using the level/pressure chart the eight-minute transient was plotted in the (1) one inch per minute timeframe using post-trip data (chart available on site). Time reference marks were established on the chart. Once time reference was made, a real time event could be established.

At 14:34:47, or 50 seconds after the scram, SRVs K, L, J, H, and A opened for 29, 11, 9, 9, and 2 seconds respectively. The post-trip data shows torus level changing at 14:34:48 which supports SRV operation.

The chart shows all the valve operations for the remainder of the transient. It is interesting to note the coincidence of SRV valve operation with each level and pressure upset. At no time during the transient did the pressure exceed 1120 psig. According to sonic indication, the following valves operated: K, L, J, H, A, E, G, and F. According to SRV tailpipe temperature alarms, the following valves operated: A, K, L, H, J, F, and G.

The attached table provides the recent setpoint data for the SRVs.

Based on this table, if one wanted to use the reactor pressure dome indication and known Wyle Laboratory tested SRV data as the determinant factor when the SRVs should have lifted, then the only relief valves that should have lifted would have been C, K, and L (assuming its setpoint is still in the 1095-1098 region).

Conclusion: Attempting to use reactor dome pressure as a measure of SRV setpoints is too prone for error. The event described in Attachment 8 also supports this.

SRV & Serial #		Nominal Setpoint (psig)	Best Known Actual Setpoint and Date of Test Prior to Event	Tested First "Pop" Setpoint and Date Following Event
A	1045	1140	1134-1138 on 04/04/85	Not tested '87
B	1088	1140	1140-1148 on 07/16/83	1144 on 02/03/87
C	1053	1140	1135-1140 on 03/26/85	1120 on 02/04/87
D	1080	1105	1112 on 03/25/85	1145 on 02/05/87
E	1056	1105	1100-1103 on 04/10/85	1243 on 02/03/87
F	1097	1140	1138-1140 on 03/29/85	Not tested '87
G	1012	1140	1133-1135 on 07/16/83	1131 on 02/05/87
H	1087	1140	1130-1135 on 01/28/83	Not tested '87 (put in service '85)
J	1110	1140	1133-1136 on 05/10/83	Not tested '87 (put in service '85)
K	1047	1090	1089-1093 on 07/15/83	1083 on 02/04/87
L	1062	1090	1095-1098 on 05/08/85	Not tested '87

Attachment 8

SUMMARY OF JANUARY 17, 1983 SCRAM AND SRV OPERATION

Background

On January 17, 1983 while at 86.3% rated power, a reactor scram and isolation occurred due to an apparent low water level due to testing. The peak reactor pressure per the post trip log was 1120.9 psig and 1170 psig based on the strip chart recorder. The following relief valves automatically lifted as indicated by the sonic detectors: L, J, G, and H. Relief valves K and J were pulled for testing due to a suspected setpoint problem.

SRV Performance

SRV & Serial #		Nominal Setpoint (psig)	Date of Test at Wyle Following Scram	Lift Pressure of First "Pop" at Test
K	1056	1090	01/24/83	1115 psig
L	1062	1090	06/29/83	1124 psig
D	1080	1105	06/30/83	1145 psig
E	1050	1105	04/12/85	1122 psig
J	1087	1140	01/26/83	1129 psig
G	1012	1140	06/29/83	1110 psig
H	1051	1140	03/21/85	1211 psig
A	1045	1140	03/20/85	1209 psig
B	1088	1140	06/29/83	>1197 psig
C	1052	1140	04/11/85	1207 psig
E	1013	1140	06/83	1174 psig

Observation: If one used reactor dome pressure to determine relief valve setpoints, the following valves should have lifted:

a) Based on 1170 pressure peak - K, L, D, E, J, G
b) Based on 1120 pressure peak - K, G

These actual test results when compared to which valves actually did lift, support the discussion of Attachment 3.