PY-CEI/NRR-0174L
ATTACHMENT

PNPP PARTIAL FEEDWATER HEATING OPERATION ANALYSIS

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Cleveland Electric Illuminating Company

Perry 1 & 2 Nuclear Power Stations

APPENDIX 15.D

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15.D PARTIAL FEEDWATER HEATING (PFH) OPERATION

15.D.1 Introduction & Summary

This section presents the results of a safety and impact evaluation for the operation of the Perry Nuclear Power Plants (PNPP) with partial feedwater heating at a steady state conditions during the operating cycle and beyond the end of cycle conditions. This evaluation is performed on an equilibrium cycle basis and is applicable to its initial core and its subsequent reload cycles. The results of this evaluation justify PNPP operation at 100% thermal power steady state conditions with rated feedwater temperature ranging from 420°F to 320°F, and also beyond the end of cycle with rated feedwater temperature ranging from 420°F to 250°F.

Operation with partial feedwater heating (PFH) occurs in the event that (i) certain stage(s) or string(s) or individual heater becomes inoperable, or (ii) intentionally valving out the extraction steam to the feedwater heaters at the end of an operating cycle. Chapter 15 has already evaluated the consequence of the transient with a sudden feedwater temperature loss of 100°F when initiated from 420°F rated feedwater temperature. This appendix will justify the continued operation of PNPP at the steady state condition ranging from rated feedwater temperature of 420°F to 320°F during the operation cycle and, as low as 250°F beyond the normal operating cycle.

Evaluations required to justify PFH operation include the abnormal operating transients, thermal hydraulic stability, the critical feedwater nozzle and sparger fatigue usage conditions and the worst loss of coolant and containment response conditions. The results are summarized below:

a) The abnormal operating transients in Chapter 15 were re-evaluated to determine the required operating MCPR limits for PFH operation. According to the worst limiting transient, the operating limit MCPR needs to be increased by 0.01, that is, 1.19 for the initial core and 1.20 for the reload core during operation when the rated feedwater temperature is between 370°F and 320°F. For operation beyond the cycle ranging from 320°F to 250°F rated feedwater temperature, the operating limit MCPR needs to be increased by 0.03, that is, to 1.21 for the initial core and 1.22 for the reload core.

b) The loss of coolant accident (LOCA) and containment response as described in Chapter 6 were re-evaluated for PFH operating condition. It is found that the conditions with normal feedwater temperature at 420°F bound those at PFH conditions.

c) Fuel integrity was evaluated with respect to general design Criterion 12 (10CFR50, App. A). It is shown that PFH operation satisfies the stability criteria and fuel integrity is not compromised.

d) The effect of acoustic and flow induced loads on the reactor shroud, shroud support and jet pumps were re-investigated to assure that design limits are not exceeded. The effect of PFH on feedwater nozzle and sparger fatigue usage factor was examined. It was found that the increased fatigue usage in 40 years still meets the acceptance criteria.

There are also other impact evaluations such as the feedwater piping, the effect of annulus pressurization and the consequences of Anticipated Transient Without Scram (ATWS). These evaluations concluded that the Perry design is adequate for PFH operation. Operation with feedwater heater(s) out of service during the operating cycle and operation at end of cycle with final feedwater temperature reduction are acceptable for PNPP.

15.D.2 Fuel Integrity - MCPR Operating Limit

15.D.2.1 Abnormal Operating Transients

All abnormal operating transients in Chapter 15 were investigated for PFH operation. Three limiting abnormal operating transients are discussed here in detail. They are:

a) Generator Load Rejections with Bypass Failure (LRNBP)
b) Feedwater Flow Controller Failure (FWCF)
c) Loss of 100°F Feedwater Heating (LFWH)

The evaluations were performed at 104.2% power, 100% core flow with rated feedwater temperature of 370°F, 320°F and 250°F at end of equilibrium cycle. Plant heat balance, core coolant hydraulic and nuclear transient data consistent with FSAR Chapter 15 input were developed and used in the analyses. Full arc (FA) turbine control valve closure characteristics were assumed in the analyses.

The end of equilibrium cycle exposure point with all the control rods fully withdrawn is the most limiting point in the cycle with the worst scram reactivity worth characteristics. A middle of the cycle point (2000 MWD/T before end of equilibrium cycle) was also analyzed for 370°F and 320°F rated feedwater temperatures to demonstrate operation during the operating cycle at these feedwater temperatures. This point is chosen because it is close enough to end of cycle such that the scram characteristics have not been significantly improved relative to earlier points in the cycle but the void reactivity characteristics are different than end of cycle. Scram characteristics are significantly improved at exposure lower than this point and the transient responses will be bounded by the two point analyzed. It is shown that the end of equilibrium cycle condition bounds the middle of cycle conditions.

The computer model described in Reference 15.D.11-1 was used to simulate the transient a) and b) events. The results for the bounding cases are summarized in Tables 15.D-1 and 15.D-2. As shown in Table 15.D-2, the operating MCPR limit shall be 1.19 (1.20 for reload core) for operation between rated feedwater temperature of 370°F and 320°F. Operation between 320°F and 250°F rated feedwater temperature requires a rated operating limit of 1.21 (1.22 for reload core).

Lower initial operating pressure and steam flow rate (due to lower feedwater temperature) provide better overpressure protection for the limiting MSIV closure flux scram event. Hence, it is concluded that the pressure barrier integrity is maintained under partial feedwater heating (PFH) conditions.

The transient responses for transients a) and b) are presented in Fig. 15.D-1 through 15.D-6.

The 100°F loss of feedwater heating transient was evaluated at 104.2% power, 100% core flow with rated feedwater temperatures of 250°F and 420°F at the end of equilibrium cycle using the computer model described in Ref. 15.D.11-2 and methodology described in Ref. 15.D.11-3. Results show that the 100°F loss of feedwater heating has less effect on colder feedwater than on the normal feedwater temperature of 420°F. Thus, the ΔCPR results for the case with 250°F initial rated feedwater temperature are bounded by the 420°F rated normal case. Moreover, it is less likely to have a sudden 100°F loss at an initial feedwater temperature of 250°F.

15.D.2.2 <u>Rod Withdrawal Error</u>

A rod withdrawal error analysis case consistent with those documented in Appendix 15B (BWR 6 generic rod withdrawal error analysis) was performed at initial feedwater temperature of 250°F to bound all rated feedwater temperature conditions. The analysis indicated that the initial steady state feedwater temperature has negligible effect with regard to ΔCPR in a random rod withdrawal error condition. Thus, the ΔCPR values initiating from 250°F feedwater temperature condition fall within the statistical data base used to establish the Rod Withdrawal Limiter System setpoints. Therefore, the generic Rod Withdrawal Error Analysis adequately bounds PFH operation conditions.

Table 15.D-1

Summary of Transient Peak Value Results

104.2% Power, 100% Core Flow

Transient	Exposure Point (MWD/T)	Rated Fdwtr Temp. (°F)	Maximum Neutron Flux % (NBR)	Maximum Dome Pressure (psig)	Maximum Vessel Pressure (psig)	Maximum Steam-line Pressure (psig)
Load Rejection With Bypass Failure	EOEC*	250	235	1193	1221	1189
	EOEC	320	246	1198	1224	1201
	EOEC	370	245	1202	1230	1209
Feedwater Controller Failure	EOEC	250	174	1128	1150	1127
	EOEC	320	139	1145	1167	1145
	EOEC	370	144	1160	1187	1158

*End of equilibrium cycle

Table 15.D-2

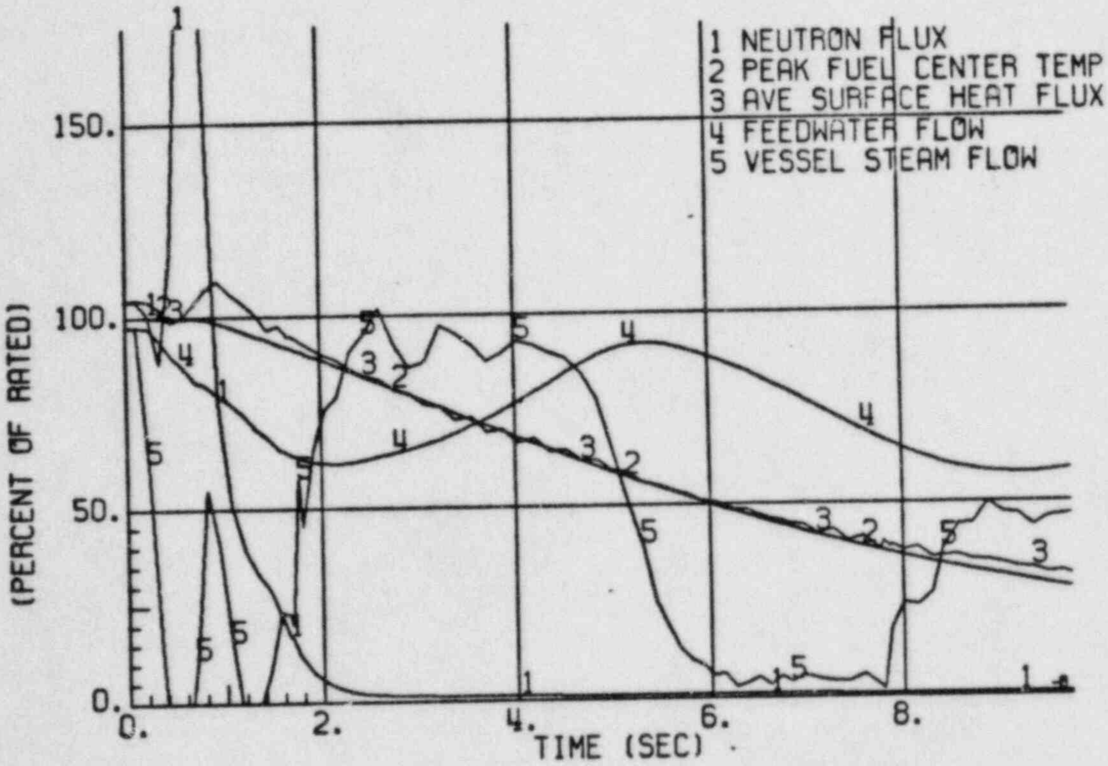
Summary of Critical Power Ratio Results*

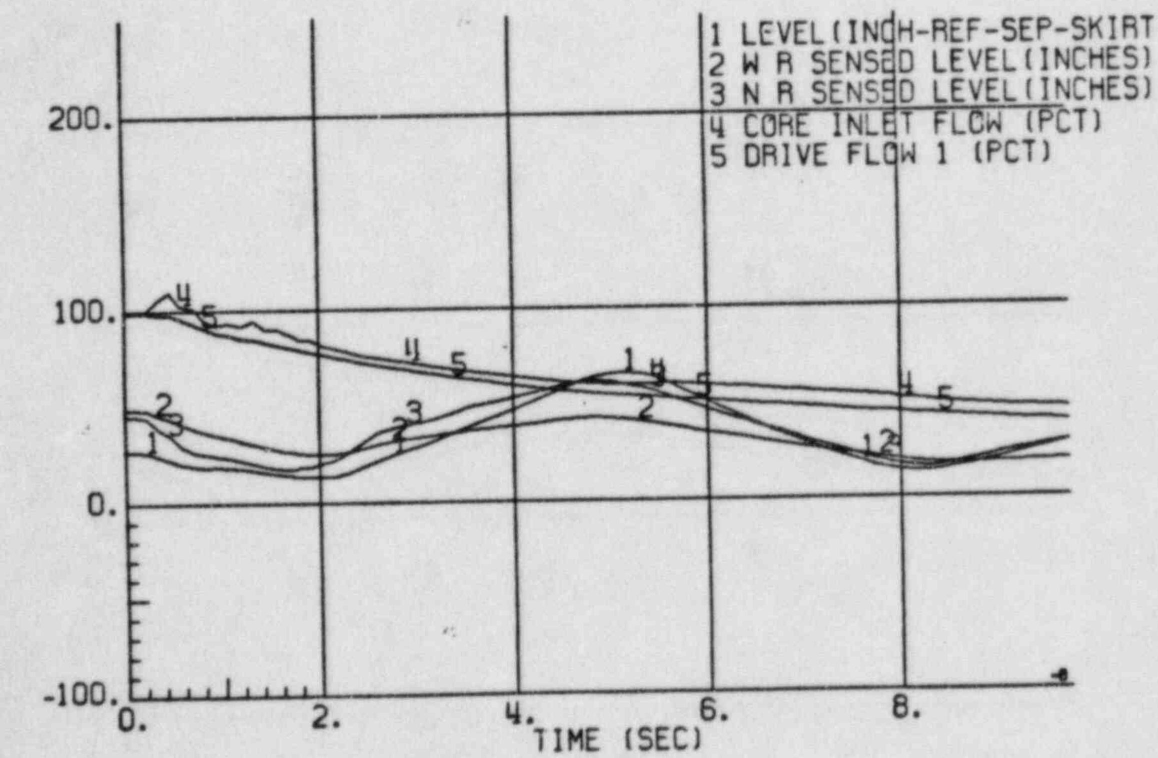
104.2% Power, 100% Core Flow

Transient	Exposure Point	Feedwater Temp. (°F)	Req'd Initial MCPR	ΔCPR	End of Transient MCPR
Load Rejection With Bypass Failure	EOEC+	250	1.18	0.11	1.07
	EOEC	320	1.18	0.11	1.07
	EOEC	370	1.18	0.10	1.08
Feedwater Controller Failure	EOEC	250	1.21	0.15	1.06
	EOEC	320	1.19	0.13	1.06
	EOEC	370	1.18	0.11	1.07

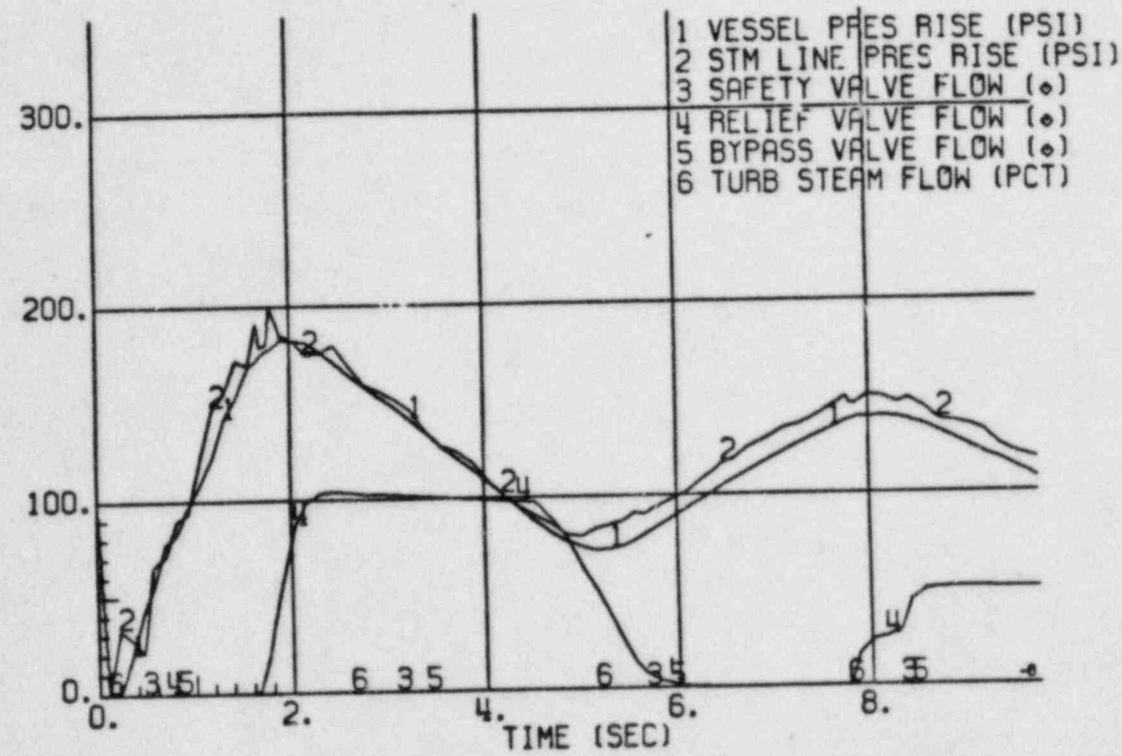
*This table is applicable to initial core with a safety limit MCPR of 1.06. For application to reload core, a 0.01 needs to be added.

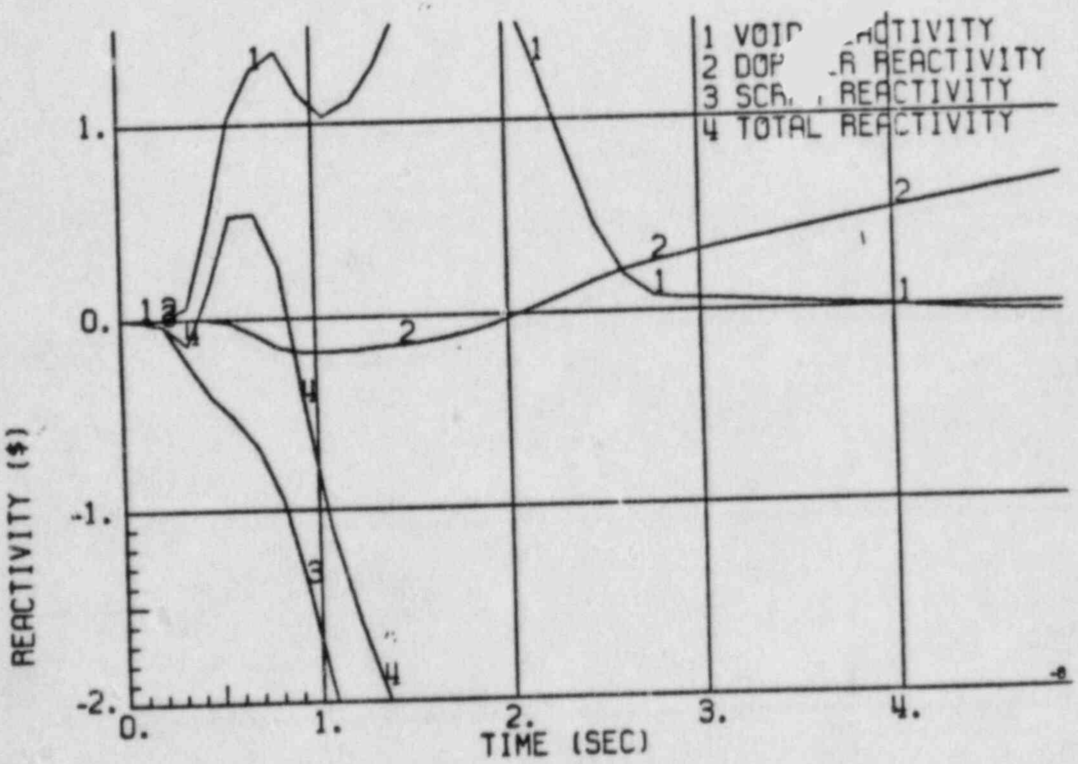
+End of equilibrium cycle.





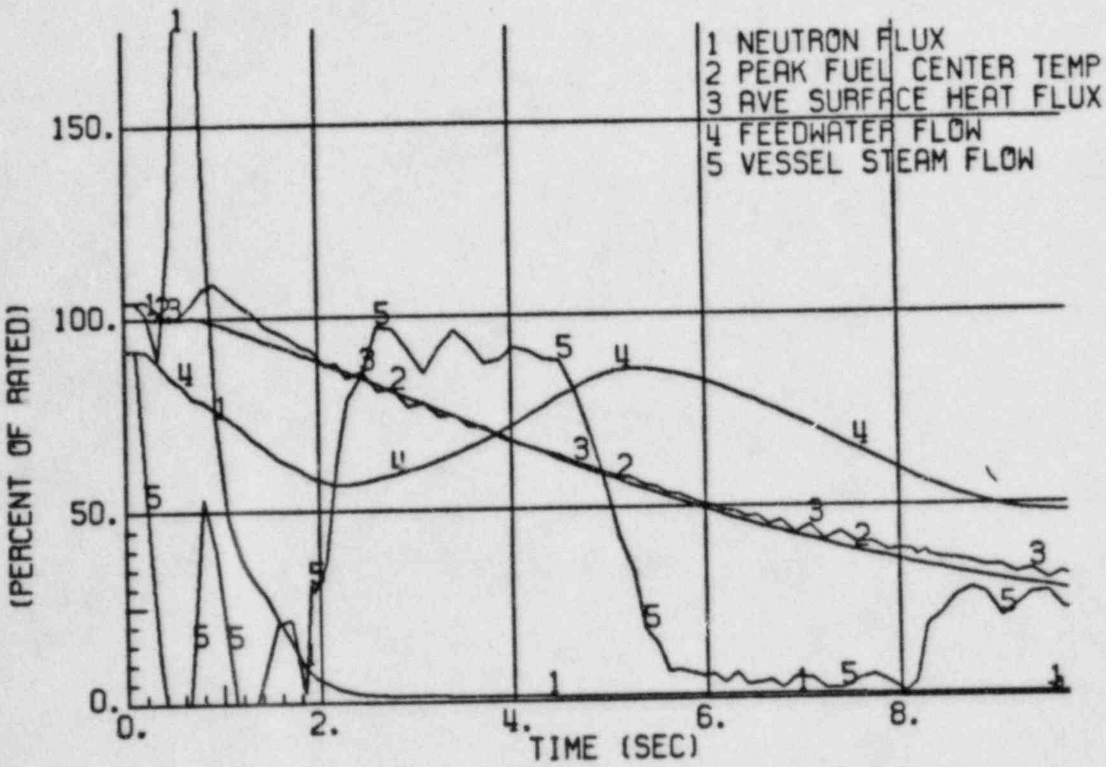
CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 100% FLOW 370°F RATED FWT	FIG. 15D-1

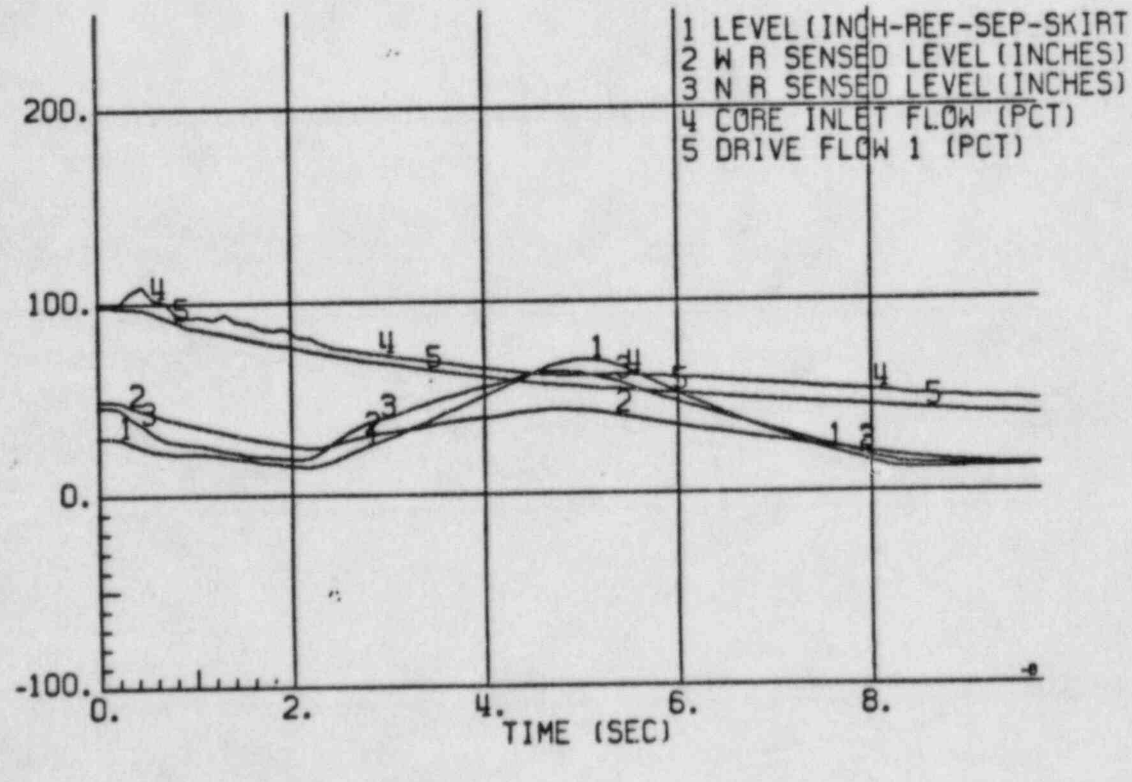




CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 100% FLOW 370°F RATED FWT	FIG. 15D-1

CONT.





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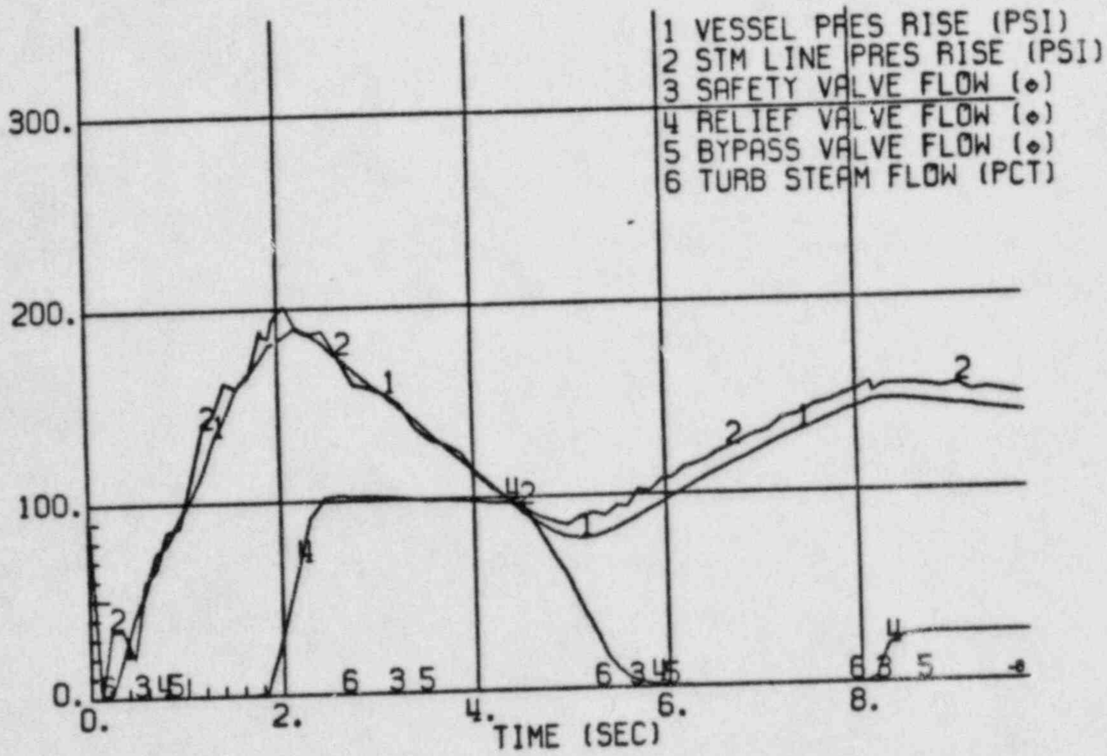
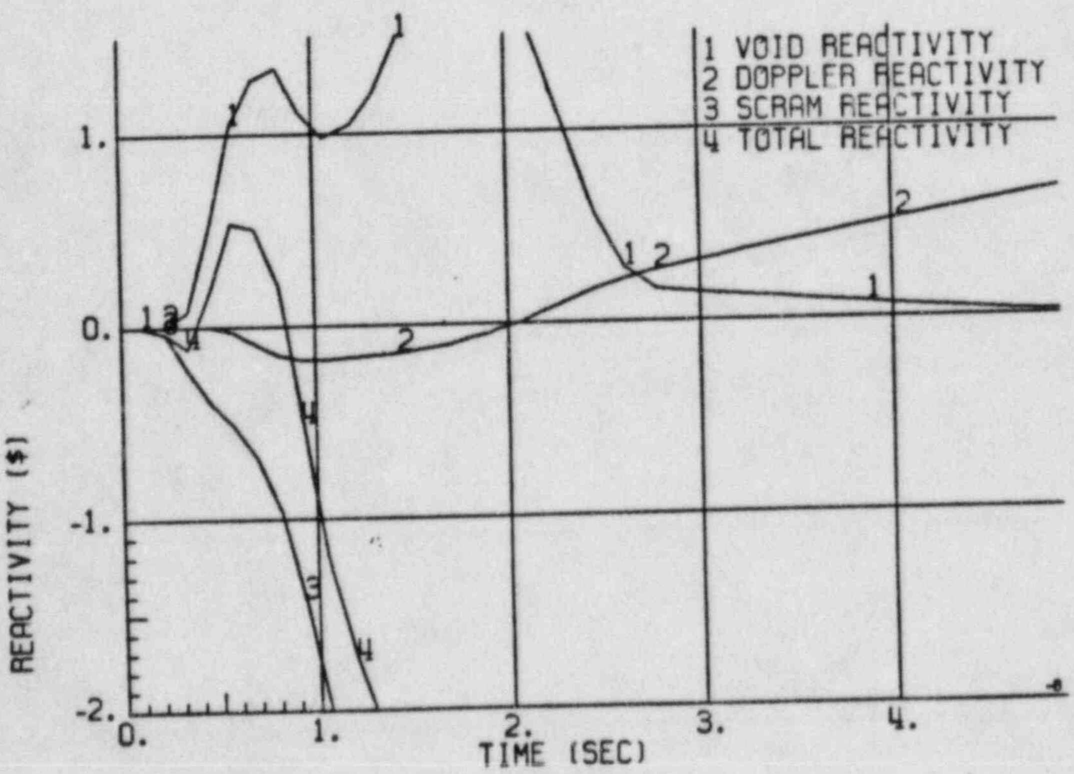
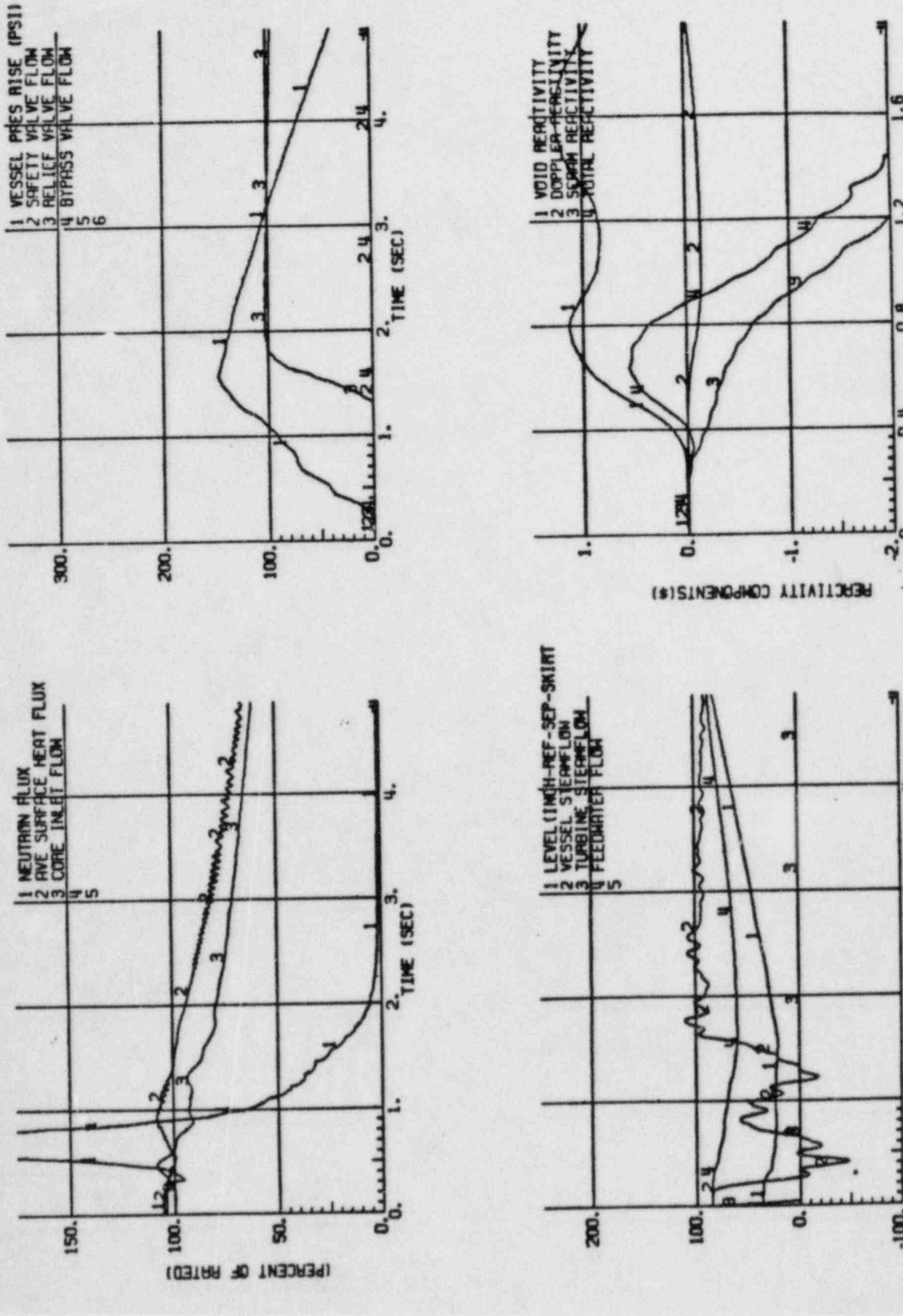


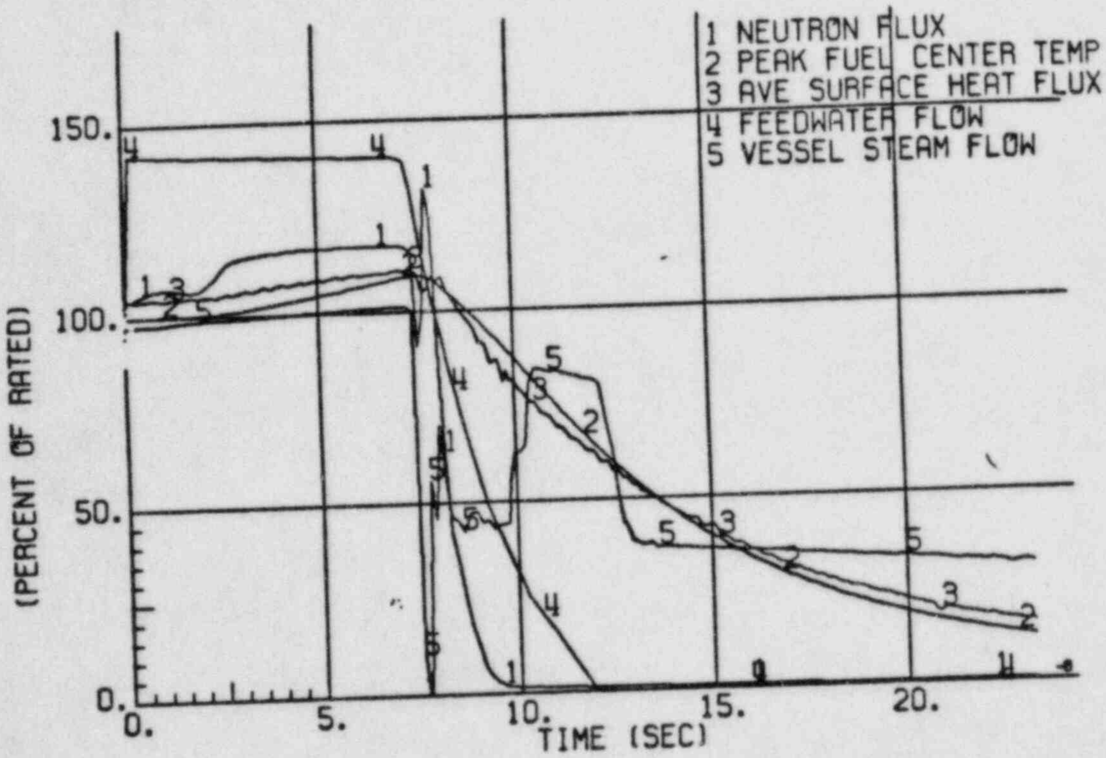


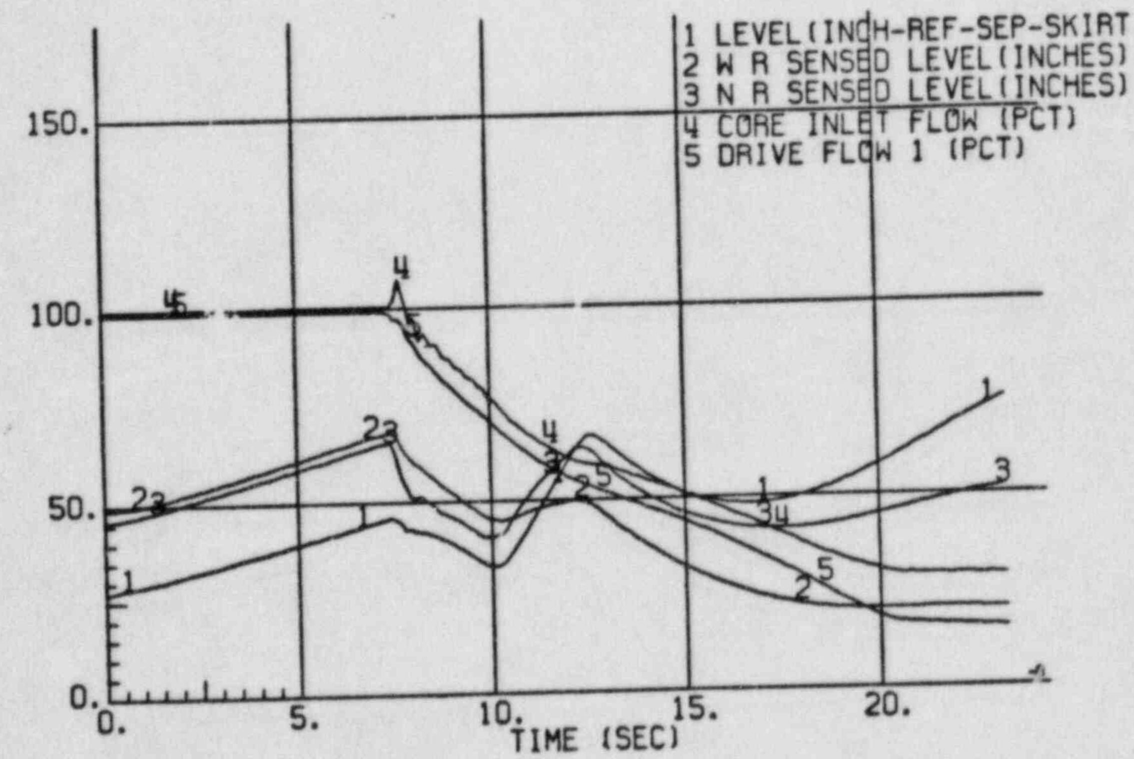
CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 100% FLOW 320°F RATED FWT	FIG. 15D-2

CONT.

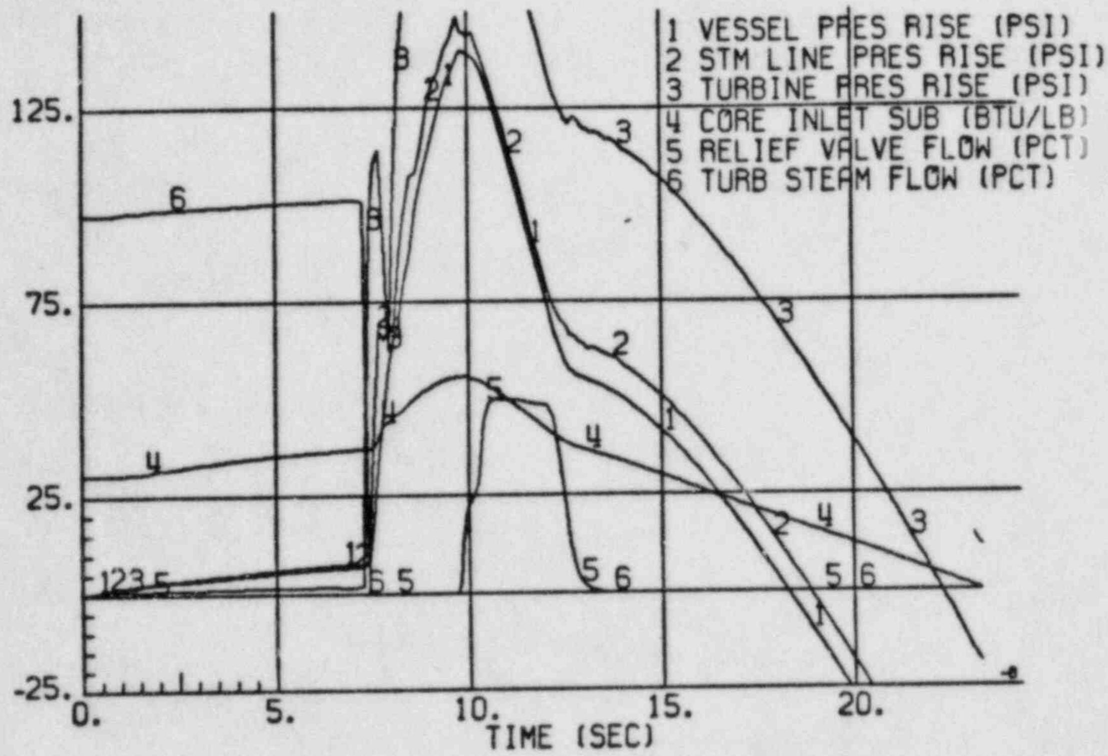


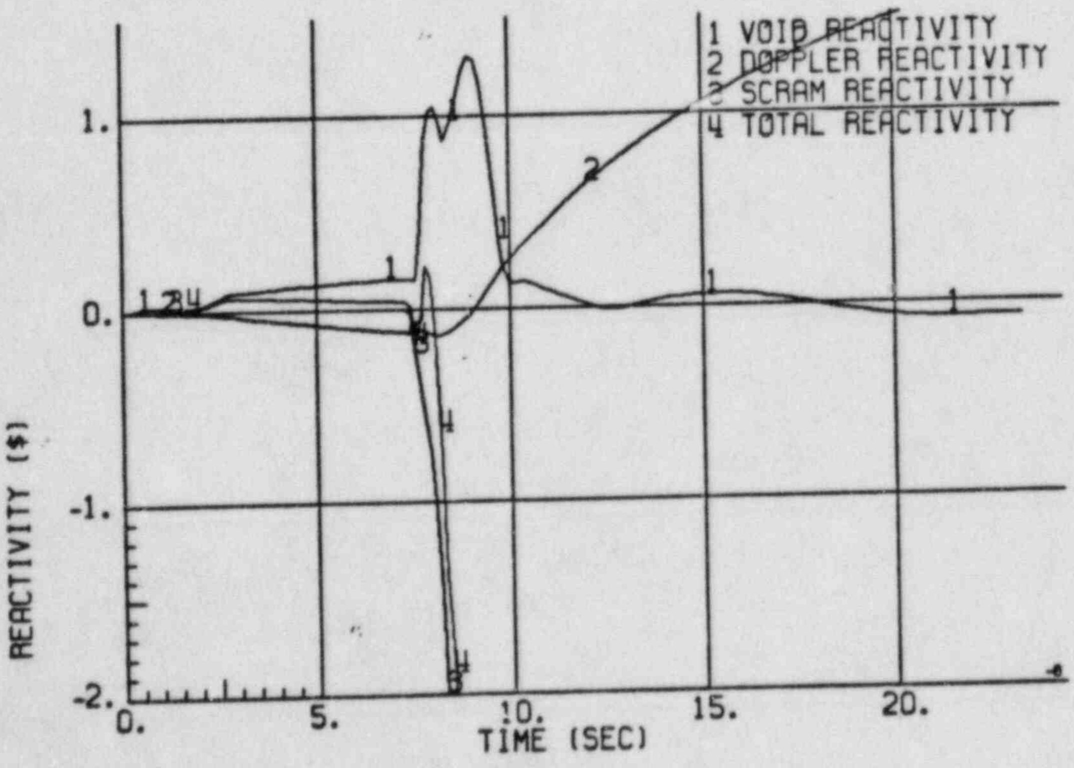
CLEVELAND ELECTRIC	GENERATOR LOAD REJECTION WITH BYPASS FAILURE 104.2% POWER 100% FLOW 250°F RATED FWT	FIG. 15D-3





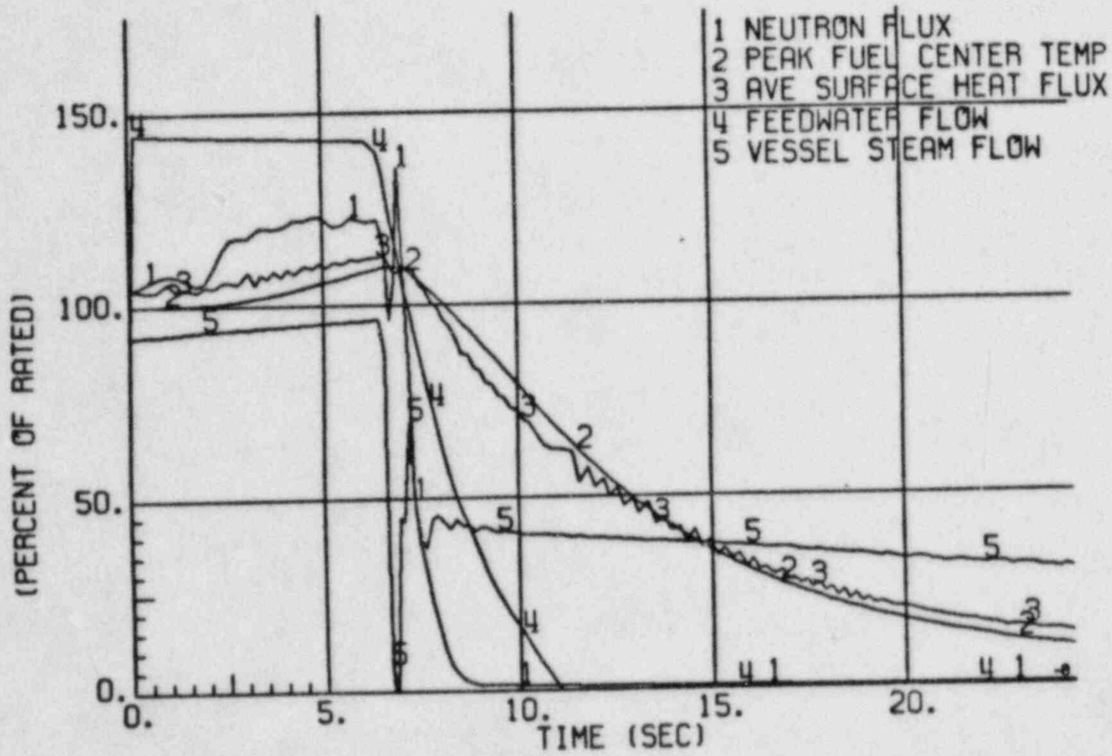
CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% Power 100% Flow 370°F RATED FWT	FIG 15D-4

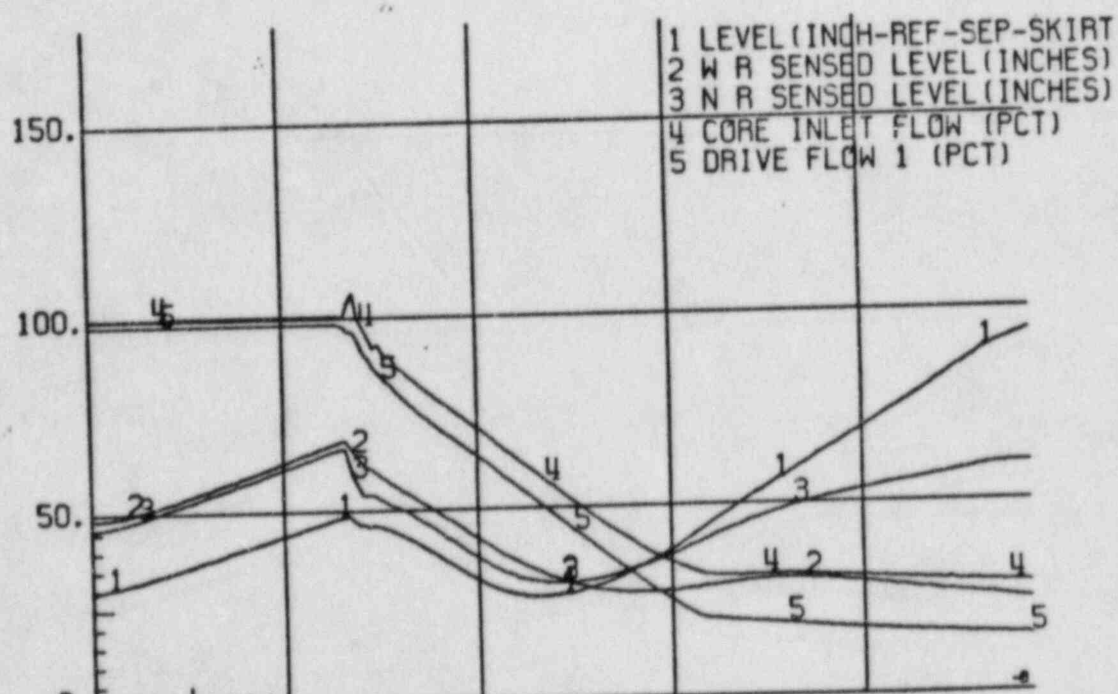




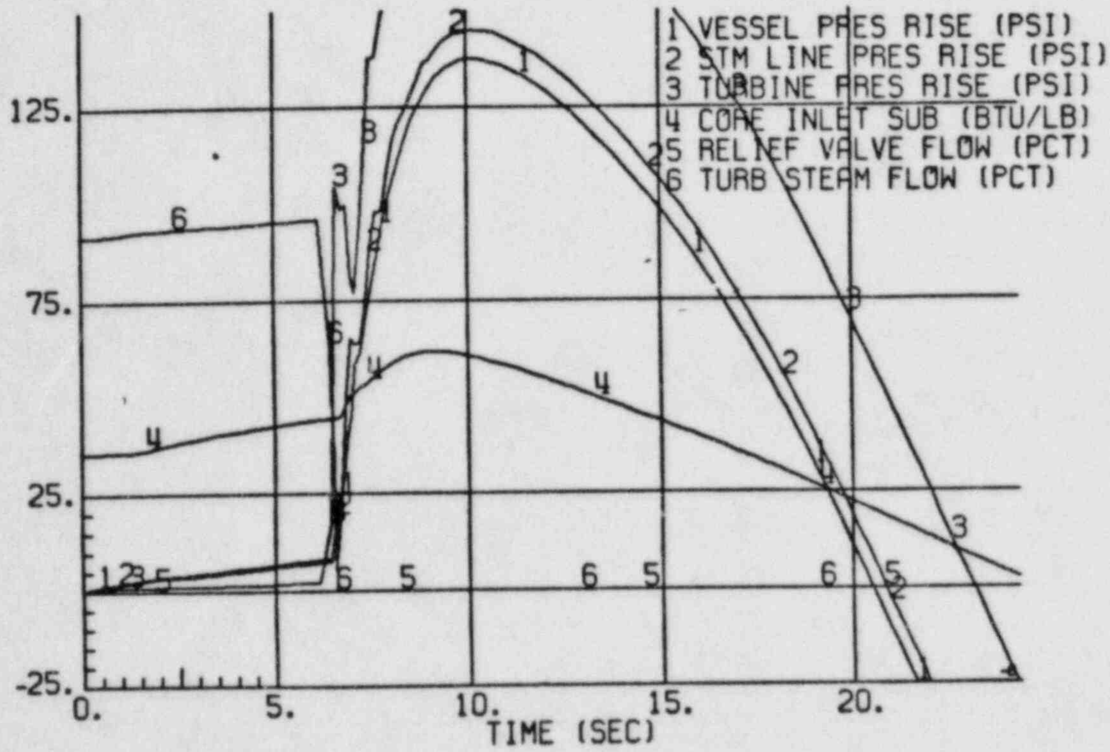
CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% Power 100% Flow 370°F RATED FWT	FIG 15D-4

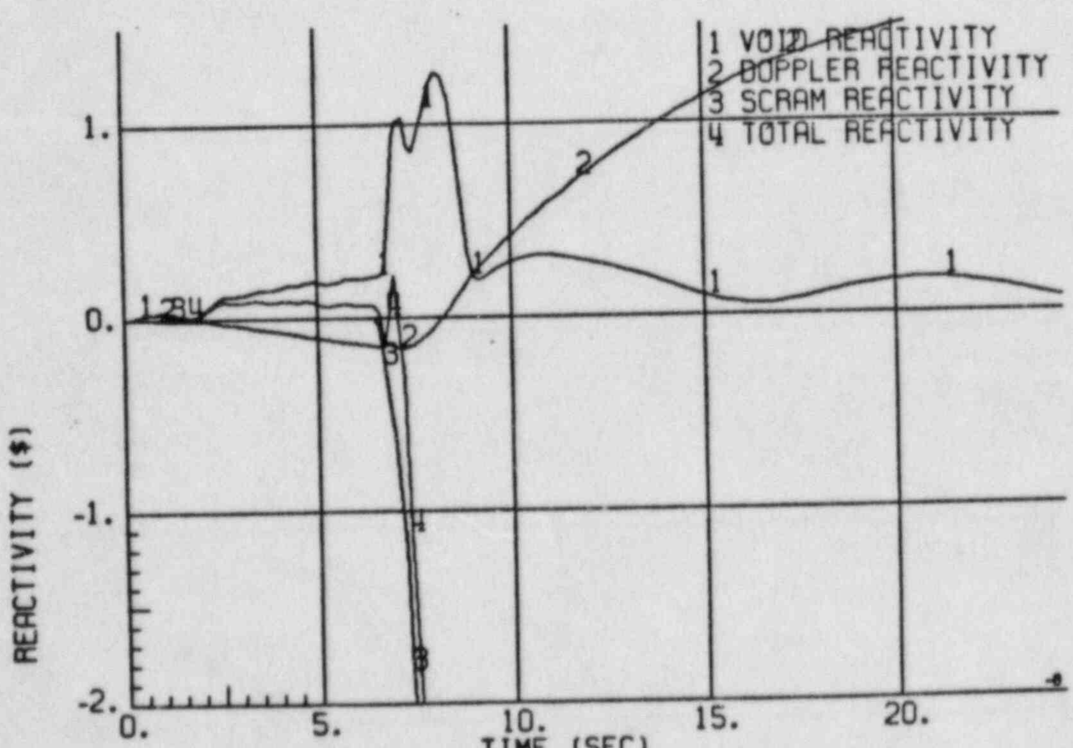
CONT.





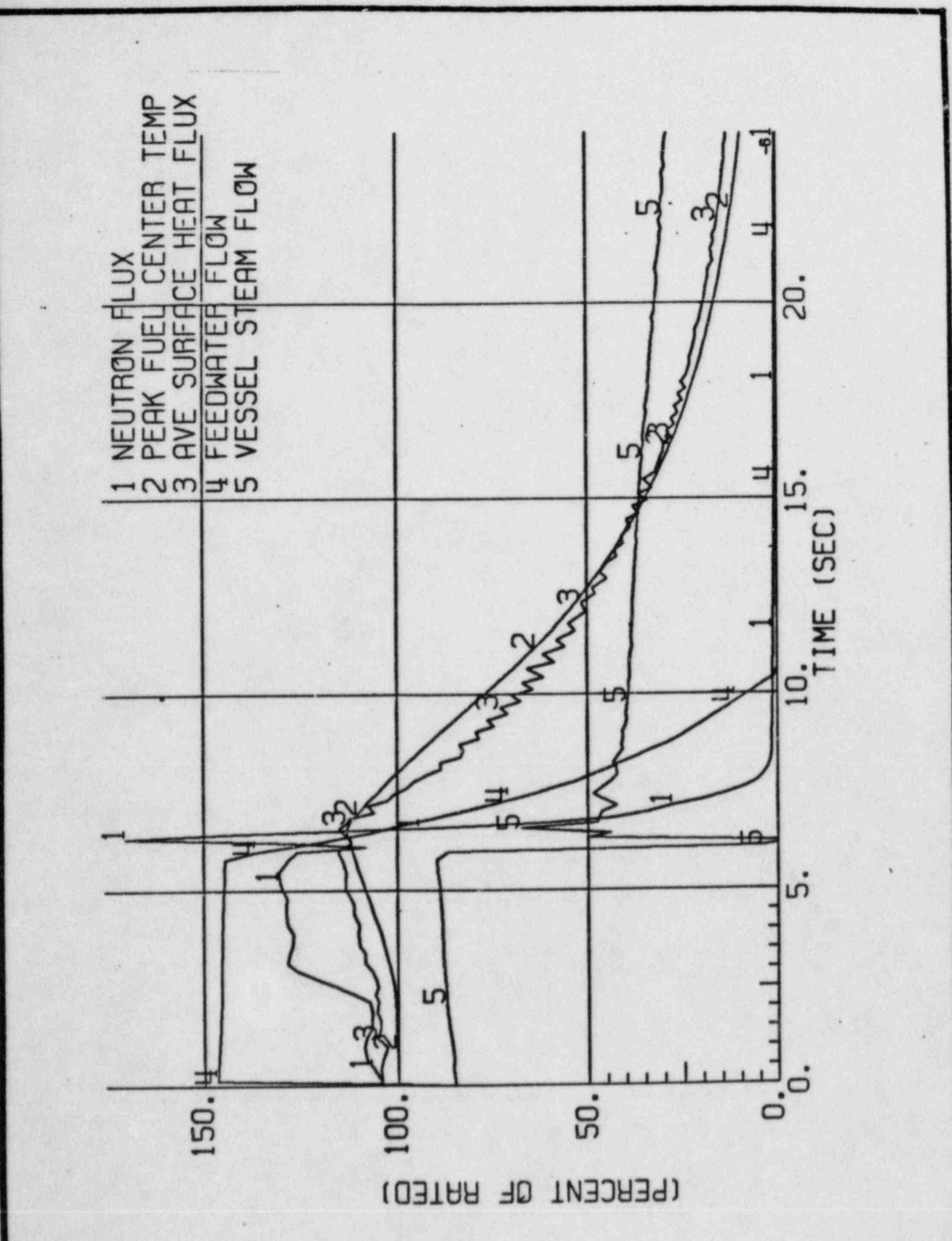
CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% Power 100% Flow 320° RATED FWT	FIG 15D-5





CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% Power 100% Flow 320°F RATED FWT	FIG 15D-5

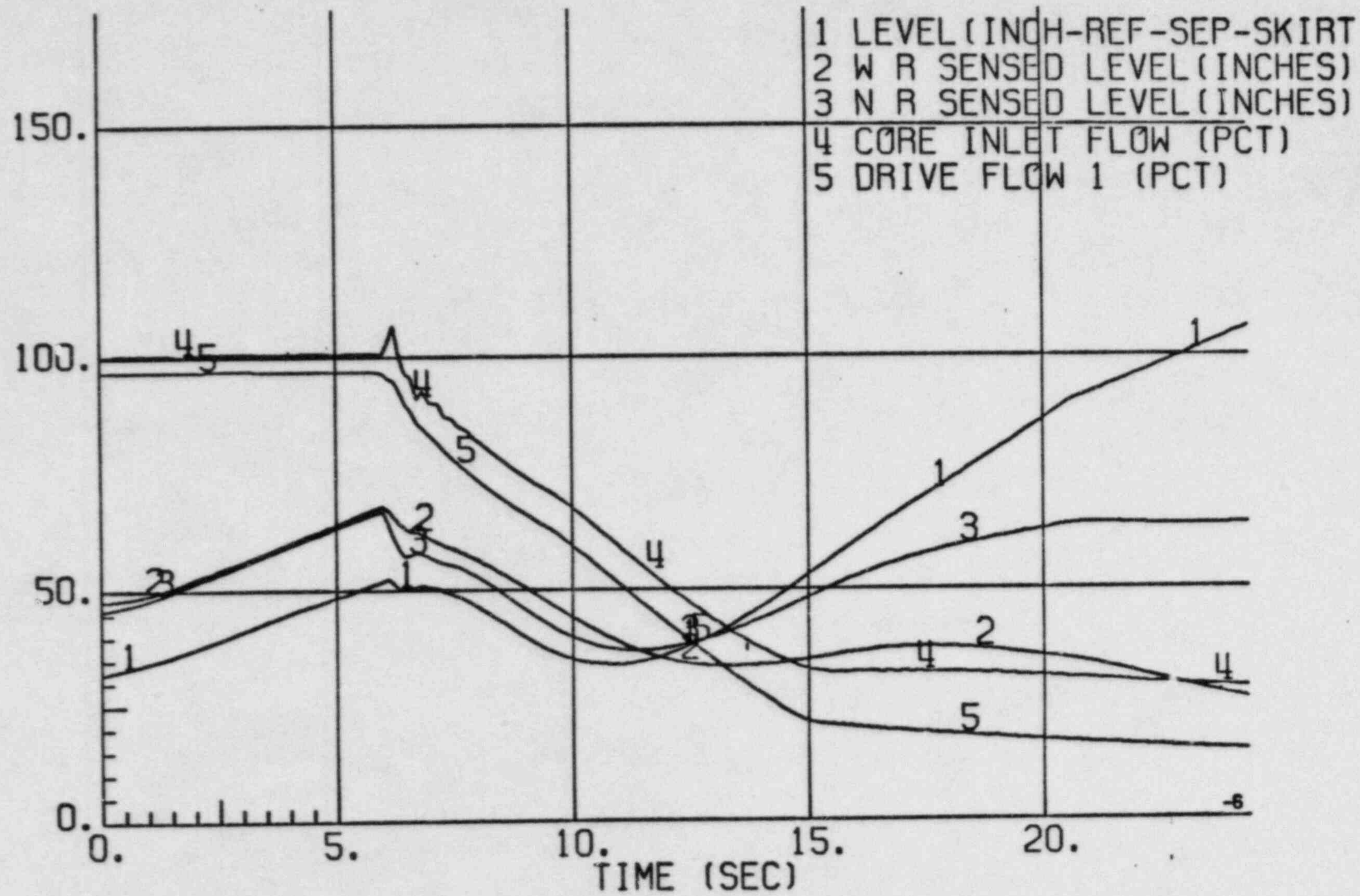
CONT.



CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% POWER 100% FLOW 250°F RATED FWT	FIG. 15D-6

CLEVELAND ELECTRIC

FEEDWATER CONTROLLER FAILURE
104.2% POWER 100% FLOW 250°F RATED FWT



15.D.2-15

FIG. 15D-6

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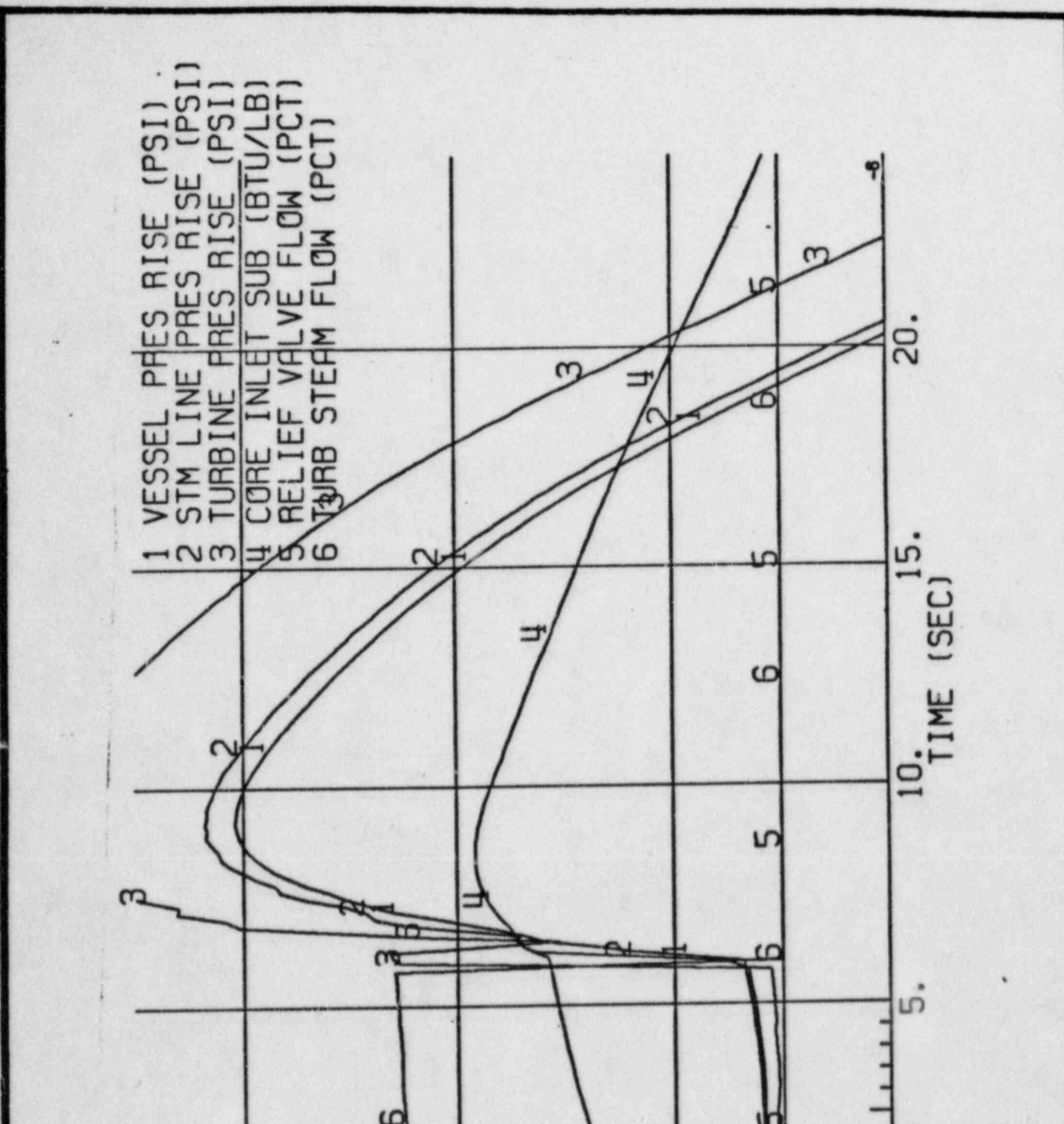
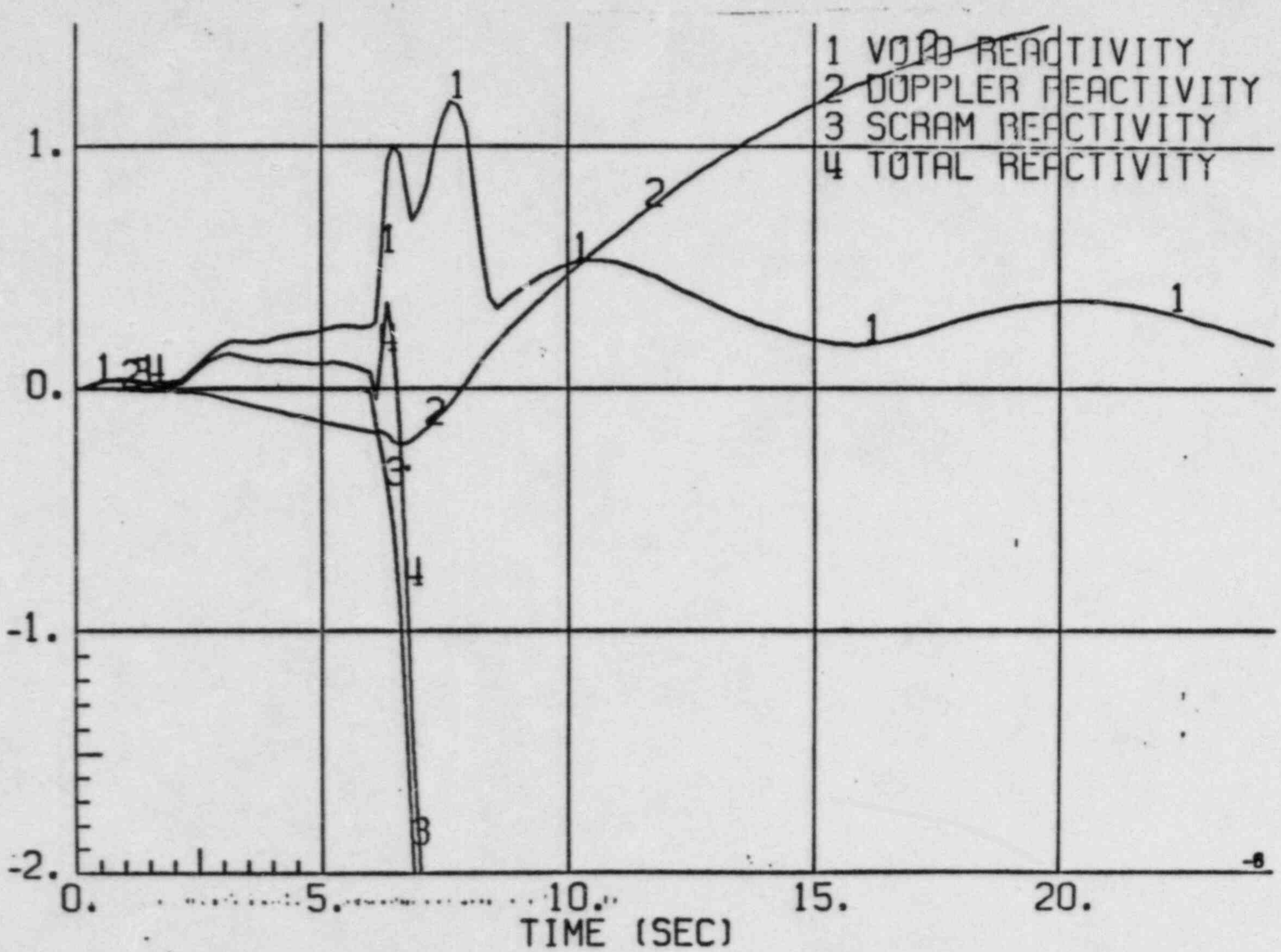


CLEVELAND ELECTRIC	FEEDWATER CONTROLLER FAILURE 104.2% POWER 100% FLOW 250°F RATED FWT	FIG. 15D-6

CONT.



CLEVELAND ELECTRIC

FEEDWATER CONTROLLER FAILURE
104.2% POWER 100% FLOW 250°F RATED FWT

FIG. 15D-6
CONT.

15.D.3 Fuel Integrity - Stability

General Design Criterion 12 (10CFR50, Appendix A) states that power oscillations which result in exceeding specified acceptable fuel design limits be either not possible or be readily and reliably detected and suppressed. Historically, compliance to GDC-12 was demonstrated by assuring that neutron flux oscillations would not occur. This eliminated the need to perform fuel integrity calculations under limit cycle conditions. As a result of stability tests at operating BWRs and extensive development and qualification of GE analytical models, stability criteria have been developed which also demonstrate compliance to GDC-12. Reference 15.D.11-4 provides these stability compliance criteria for GE fueled BWRs operating in the vicinity of limit cycles.

Operation in the partial feedwater heating (PFH) mode is bounded by the fuel integrity analyses in Reference 15.D.11-4. In general, the effect of reduced feedwater temperature results in a higher initial CPR which yields even larger margins than those reported in reference 15.D.11-4. The analyses are independent of the stability margin since the reactor is already assumed in limit cycle oscillations. Reference 15.D.11.4 also demonstrates that for neutron flux limit cycle oscillations just below the 120% neutron flux scram setpoint, fuel design limits are not exceeded for those GE BWR fuel designs contained in General Electric Standard Application for Reactor Fuel (GESTAR, Reference 15.D.11-5). These evaluations demonstrate that substantial thermal/mechanical margin is available for the GE BWR fuel designs even in the unlikely event of very large oscillations.

To provide assurance that acceptable plant performance is achieved during operation in the least stable region of the power/flow map, as well as during all plant maneuvering and operating states, a generic set of operator recommendations has been developed and communicated to all GE BWRs. These recommendations instruct the operator on how to reliably detect and suppress limit cycle neutron flux oscillations should they occur. The recommendations were developed to conservatively bound the expected performance of all current product lines.

When operating in the partial feedwater heating mode during a cycle, the colder feedwater flow increases the core inlet subcooling and will also result in power distribution changes. These changes result in reduced stability margin when operating in the high power/low flow region of the operating domain. Tests performed at an overseas BWR/6 in October, 1984 evaluated the effects of reduced feedwater temperature on stability margins. The result shows that the reduction in stability margin is within the conservative basis of the operator recommendations and therefore the recommendations are applicable for partial feedwater heating during the cycle.

For operation at the end of the cycle with partial feedwater heating to extend the operating cycle, the power distribution approximates the target power shape (typically a Haling power distribution) with all control rods fully withdrawn. Reducing the feedwater temperature at this point will result in an increased peak but at a higher elevation in the core. The change in power shape partially offsets the reduced inlet enthalpy effect on stability and the result is a small change in stability margin. The change in stability margin is well within the conservative basis of the operator recommendations and therefore the recommendations are applicable to operation with PFH down to rated feedwater temperature of 250°F at the end of cycle conditions.

15.D.4 Loss of Coolant Accident Analysis

A Loss of Coolant Analysis (LOCA) was performed for PNPP with PFH operation condition at 250°F rated feedwater temperature. Reduction of feedwater temperature results in increased subcooling in the vessel thus increasing the mass flow rate out of a LOCA break. However, an increase in initial total system mass and a delay in lower plenum flashing also occur. They act together to decrease the impact of increased flow out of the recirculation line break. As a result of this offsetting effect, the peak cladding temperature (PCT) was shown to be lower than the 2115°F value reported for PNPP and below the 2200°F 10CFR50.46 cladding temperature limit.

15.D.5 Containment Response Analysis

The impact of partial feedwater heating (PFH) on the containment LOCA response was evaluated. Both Main Steam Line (MSL) break and recirculation line break were analyzed over the entire power/flow region. Reduced feedwater temperature increases the subcooling of the coolant, and the mass flow rate from the postulated recirculation pipe break also increases, but is limited to the critical flow of the break. The final outcome is that the peak drywell and containment pressures under the partial feedwater heating conditions are bounded by the design values in FSAR Chapter 6.

15.D.6 Acoustic Load and Flow Induced Loads Impact on Internals

Acoustic loads are loads on vessel internals created by a sudden LOCA. Acoustic loading is proportional to total pressure wave amplitude to the vessel due to LOCA.

Loads are created on the shroud, shroud support and jet pumps due to high velocity flow in the downcomer in a postulated recirculation line break. These flow induced loads are affected by the critical mass flux rate out of the break. Partial feedwater heating operation increases subcooling in the downcomer thus increases critical flow. However, PFH also increases density. The reactor internals most impacted by acoustic and flow induced loads are the shroud, shroud support and jet pump. The impacts on these components were evaluated over the operating power flow region. The analyses concluded that these components have been designed to handle the loading during reduced feedwater temperature conditions.

15.D.7 Feedwater Nozzle Fatigue Usage

An evaluation was performed on the PNPP feedwater nozzle with partial feedwater heating at rated feedwater temperature of 250°F for conservatism. An 18 month operating cycle with partial feedwater heating based on an 80% capacity factor is equivalent to 438 full power days per cycle. This results in an additional 0.0214 fatigue usage factor over 40 years of continuous operation at 250°F. Furthermore, if we assume additional end of cycle operation with feedwater temperature between 420°F and 250°F for 41 full power days per cycle for 40 years, the resultant fatigue usage factor would increase by 0.001. The total fatigue usage factor will still be less than 0.8, which is below the limit of 1.0.

The above assumption of 40 years of continuous partial feedwater heating operation is extremely conservative. The nozzle fatigue is expected to be much less than the results presented above. Hence, PFH operation is an acceptable mode even for the most "fatigue-critical" vessel nozzle.

15.D.8 Feedwater Sparger Impact Evaluation

An evaluation was performed to examine the impact of partial feedwater heating operation on the feedwater sparger for PNPP. Six cases were analyzed to determine the number of days allowable per year (for 40 years) for partial feedwater heating operation without exceeding the feedwater sparger fatigue usage factor limit of 1.0. Results of this study are presented in Table 15.D-3. This table indicates the annual average number of days allowable for partial feedwater heating, reducing from normal 420°F to 370°F or to 320°F rated feedwater temperature with an additional 41 end of cycle days at 250°F. For example, the feedwater sparger is designed to operate with 21 days of partial feedwater heating at rated 320°F during a fuel cycle and 41 days of partial feedwater heating at rated 250°F beyond the end of the fuel cycle for every fuel cycle for 40 years. The feedwater sparger is acceptable for partial feedwater heating operation within these limits.

Table 15.D-3

Summary of Feedwater Sparger Fatigue Analysis

Feedwater Temperature reduction to 250°F for 41 days at End of each 18-Month Cycle for 40 Years	Allowable Number of Days per Year** for 40 Years at Feedwater Temperature of	
	370°F	320°F
3 Step*	127	21
7 Step*	144	24
No end of cycle reduction	256	61

*3 Step means ~3 average steps of feedwater temperature reduction from 420°F to 370°F or 320°F.

7 Step means ~7 average steps of feedwater temperature reduction from 420°F to 370°F or 320°F.

**This evaluation assumes 70% capacity factor. Allowable number of days which results in a feedwater sparger fatigue usage factor of 1.0.

15.D.9 Reactor Protection System Setpoints

At reactor power levels where significant amounts of steam are being generated, the fast closure of turbine stop or control valves will result in rapid reactor vessel pressurization. When pressure increases, power increases, especially if the bypass valves fail to open. For this reason, scram occurs on turbine stop valve position and control valve fast closure to provide margin to the core thermal-hydraulic safety limit. At low power levels high neutron flux scram, vessel pressure scram, and other normal scram functions provide sufficient protection. Therefore, below 40% rated power, turbine stop valve and control valve scram functions are bypassed. The 40% NB rated power is sensed through the direct measurement of the turbine first stage pressure.

As feedwater temperature is reduced steam flow decreases. If the core thermal power is maintained with partial feedwater heating, the steam flow change means that the turbine first-stage pressure versus power relationship is altered. Thus, it is necessary to readjust turbine stop and control valve scram bypass setpoints (sensed from turbine first stage pressure) for partial feedwater heating operation. A new setpoint is established for the trip units prior to commencement of each partial feedwater heating operation at each operating cycle.

15.D.10 Miscellaneous Impact Evaluation

15.D.10.1 Feedwater System Piping

The impact of partial feedwater heating operation on the feedwater system piping up to the first feedwater guide lug outside the containment has been evaluated for feedwater temperature at 250°F. Results of the study show that with the additional partial feedwaste heating operations, the feedwater piping fatigue usage factor still meets the allowable limit of 1.0.

15.D.10.2 Impact on Anticipated Transient Without Scram (ATWS)

An impact evaluation performed for PNPP shows that reducing feedwater temperature helps to reduce the consequences of an ATWS event. As a result of reduced feedwater temperature, steam flow and core average void fraction are reduced. This results in lower void coefficient and greater CPR margin which corresponds to milder transients.

15.D.10.3 Annulus Pressurization Load (APL) Impact

A boundary analysis was performed to determine the impact of partial feedwater heating operation on annulus pressurization loads (APL). It is found that partial feedwater heating has a small impact on annulus pressurization loads and is bounded by the normal operation APL limits.

15.D.10.4 Fuel Mechanical Performance

Evaluations were performed to determine the acceptability of PNPP partial feedwater heating operation on GE-6 fuel rod and assembly thermal/mechanical performance. Component pressure differential [illegible] fuel rod overpower values were determined for anticipated operational [illegible]rrences with partial feedwater heating conditions. These values were found [illegible]ounded by those applied in the fuel rod and assembly design bases and there[illegible] PNPP with partial feedwater heating operation is acceptable and co[illegible]istent with the fuel design basis.

15.D.11 References

15.D.11-1 "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" NEDO-24154 October 1978.

15.D.11-2 "Three Dimensional BWR Core Simulator" NEDO-20953-A, January 1977.

15.D.11-3 Letter, J. S. Charnley (GE) to F. J. Miraglia (NRC), "Loss of Feedwater Heating Analysis", July 5, 1983 (MFN-125-83).

15.D.11-4 "Compliance of the General Electric Boiling Water Reactor Fuel Designs to Stability Licensing Criteria" NEDE-22277-P, December 1982.

15.D.11-5 "General Electric Standard Application for Reactor Fuel" NEDE-24011-P-A, January 1982.

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