Attachment 4

ANF-88-069

ADVANCED NUCLEAR FUELS CORPORATION

EXTENDED OPERATING DOMAIN/EQUIPMENT OUT OF SERVICE ANALYSIS FOR DRESDEN UNITS 2 AND 3

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EXTENDED OPERATING DOMAIN/EQUIPMENT OUT OF SERVICE ANALYSIS FOR DRESDEN UNITS 2 AND 3

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1.0 INTRODUCTION

This report describes the plant transient analyses performed by Advanced Nuclear Fuels Corporation (ANF) in support of Increased Core Flow (ICF) to 108% of rated, Feedwater Heaters Out of Service and Final Feedwater Temperature Reduction (FHOOS/FFTR), and Coastdown Operation for Dresden Units 2 and 3. Because Dresden Units 2 and 3 have equivalent physical systems from a transient analysis viewpoint, conclusions drawn from these analyses are generically applicable to both plants for present and future reloads of ANF fuel.

The purpose of this document is to establish that the most limiting condition for operation of the Dresden units is at full power and increased core flow. All future analyses will be performed at this operating condition.

The analyses performed in this document were performed using the same average core plant transient analysis methodology as used to calculate thermal margin requirements for current operation of both Dresden units (Ref. 1 and 2). The approved XCOBRA-T hot channel model determined the limiting change in the Critical Power Ratio (delta CPR).

This analysis supports operation in the expanded power and flow operating map shown in Figure 1.1.

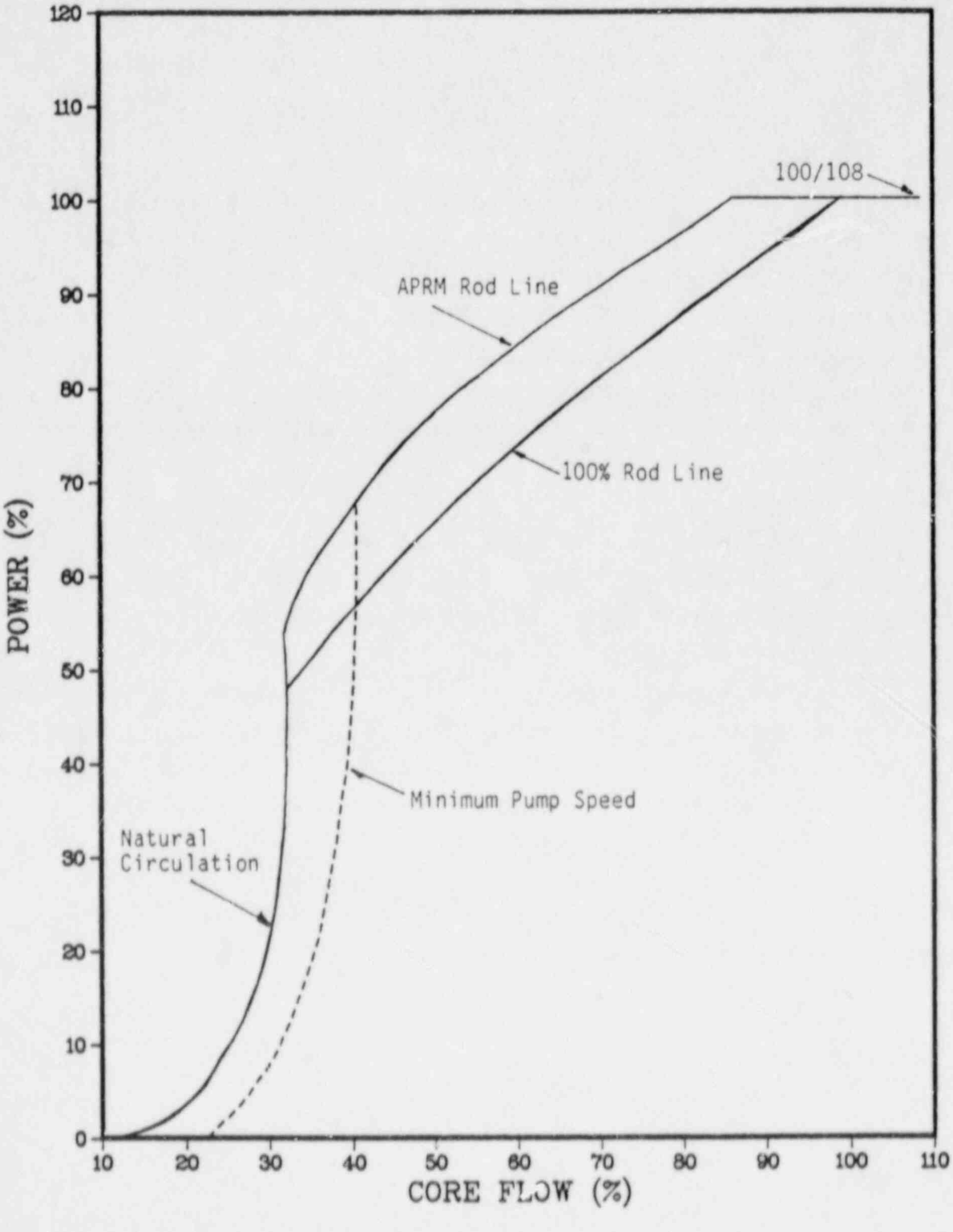


FIGURE 1.1 DRESDEN UNITS 2/3 OPERATING MAP

2.0 SUMMARY

The determination of thermal margin requirements for the Dresden units is based on the consideration of various operational transients. Reference 2 identifies the limiting transients in each general category of events. The most limiting transient events for thermal margin in BWR/3 applications are the generator load rejection without bypass to the condenser, the loss of feedwater heating event, and the feedwater controller failure to maximum demand. The most limiting event for the Dresden units is the generator load rejection without bypass valve operation.

Analyses assure that the MCPR Operating Limit protects all operating domains. The present operating map for the Dresden units allows flows up to 108% of rated. Operation with feedwater heaters out of service or with final feedwater temperature reduction increases operating flexibility. These two phenomena are functionally equivalent and are analyzed with a feedwater temperature reduction up to 100°F. Coastdown operation will extend the end of the operating cycle. Each of these conditions were analyzed to determine the most limiting condition for operation for both Dresden units. By setting the Technical Specification limit on the most limiting condition, all others are bounded.

The most limiting condition for operation was established to be full power and increased core flow at normal feedwater temperature. Therefore, all future analyses will be performed at these operating conditions. No specific plant parameters need be checked on a per cycle basis to assure applicability of this report.

The closure of all main steam isolation valves (MSIVs) is the maximum system pressure event for ASME overpressure. MSIV closure is most limiting without activation of the MSIV position scram and without pressure relief credit for the four electromagnetic relief valves. The results of this analysis indicate that the ICF condition is the most limiting but still within the requirements of the ASME code regarding vessel overpressure.

3.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN AT ICF AND FHOOS/FFTR

3.1 Design Basis

Dresden Units 2 and 3 are sister plants with equivalent physical systems from a transient analysis viewpoint. Both plants contain cores of ANF 8x8 and ANF 9x9 fuel types. Limiting plant transient phenomena are a function of the physical characteristics of the plant rather than specific fuel types. Therefore, the conclusions from these analyses are applicable to both Dresden units. Reactor plant conditions for these analyses are shown in Table 3.1.

The most limiting point in the cycle is when the control rods are fully withdrawn from the core. The thermal margins established for the end of full power capability are conservative for cases where control rods are partially inserted.

3.2 Calculational Model

The average core plant transient methodology previously described in References 2 and 4 as updated in Appendix A of Reference 1 was used for the analysis reported in this document. The delta CPRs were calculated with the approved XCOBRA-T (Ref. 5) hot channel model. A conservative integral power multiplier of 110% applied in XCOBRA-T to pressurization transients accounted for COTRANSA code uncertainties.

The axial power shifts associated with the system overpressurization in the generator load rejection and the feedwater controller failure transients were modeled using the COTRANSA one dimensional core model. RODEX2 (Ref. 3) calculations determined conservative fuel pellet to clad gap conductances based on the Dresden units core configuration.

In accordance with ANF methodology, consistent bounding input is used to evaluate possible limiting transients. From these bounding results, the limiting transient is a generator load rejection without bypass valve operation.

3.3 Anticipated Transients

Reference 2 generically considers eight major categories of system transients. For both Dresden units the limiting events are the generator load rejection without bypass, the feedwater controller failure, and the loss of feedwater heating. The generator load rejection and the feedwater controller failure to maximum demand have been evaluated for effects of increased core flow and FHOOS/FFTR. Analysis of FHOOS and FFTR at increased core flow conservatively bounds FHOOS/FFTR at nominal core flow. These analyses assumed that a relief valve was out of service.

3.3.1 Load Rejection Without Bypass Valve Operation

The load rejection without bypass valve operation (LRWB) is the most limiting of the rapid pressurization transients and of all the system transients for the Dresden units. In the load rejection transient, the abrupt closure of the turbine control valve rapidly stops steam flow. The resulting pressure increase causes a decrease in void level in the core, which in turn creates a power excursion. This excursion is mitigated by Doppler broadening and pressure relief. However, the primary mechanisms for termination of the event are rod insertion and revoiding of the core.

The important parameters for this transient include the power transient (integral power) determined by the void reactivity and the control rod worth. The void reactivity effects the power excursion rate and part of the intrinsic shutdown mechanism. The control rod worth determines the value of scram reactivity. Table 3.2 is a comparison of generator load rejection transients analyzed for a Dresden unit. The ICF case has the highest maximum neutron flux during the transient. Figures 3.1 through 3.4 compare important parameters for the ICF, FHOOS/FFTR, and nominal rated cases. Figure 3.1 shows that the ICF case also has the largest integral power. The total core power produced during the transient for the ICF case was 3883 MW-sec and for the FHOOS/FFTR case it was 3535 MW-sec. For comparison, the nominal case had a core power production of 3640 MW-sec. This is because the ICF case results in a higher positive reactivity insertion rate during the pressurization portion of the transient. This results in the ICF case being the limiting pressurization event. This conclusion is valid for both Dresden Unit 2 and Unit 3.

3.3.2 <u>Feedwater Controller Failure</u>

Failure of the feedwater control system could lead to a maximum increase of feedwater flow into the reactor vessel. The excessive feedwater flow increases the subcooling in the recirculating water returning to the reactor core. This reduction in average moderator temperature results in a core power increase. Eventually, the increasing water level in the downcomer region will reach the high water level trip. The high level trip initiates a turbine trip to prevent water from reaching the turbine. The turbine trip closes the turbine stop valves and the resulting scram arrests the power increase. The pressure pulse resulting from the stop valve closure is mitigated by opening the bypass valves to the condenser.

Figures 3.5 through 3.9 compare important parameters for the ICF, FHOOS/FFTR, and the nominal cases. The total core power produced during the transient for the ICF case was 4421 MW-sec and for the FHOOS/FFTR case it was 3835 MW-sec. For comparison, the nominal case had a core power production of 4200 MW-sec. The FHOOS/FFTR case has a lower integral power because of the lower steam dome pressure at the beginning of the transient. This lower steam dome pressure results in a larger rate of change of the feedwater flow. Therefore, the high reactor vessel water level trip is reached earl[illegible]n in the ICF and nominal cases. Table 3.3 shows that the ICF case also has the maximum peak neutronic power. The ICF case is the limiting feedwater controller failure to maximum demand for both Dresden Units 2 and 3.

TABLE 3.1 REACTOR AND PLANT CONDITIONS

PARAMETER	NOMINAL	FHOOS/FFTR	ICF
Reactor Power (MWt)	2527	2527	2527
Total Recirculating Flow (Mlb/hr)	98.0	105.8	105.8
Core Inlet Enthalpy (Btu/lbm)	522.3	513.0	524.0
Steam Dome Pressure (psia)	1020	1005	1020
Steam Flow (Mlb/hr)	9.8	8.7	9.8
Feedwater Enthalpy (Btu/lb)	312.1	201.4	312.9
Recirculating Pump Flow (Mlb/hr)	17.1	18.5	18.5

TABLE 3.2 LOAD REJECTION WITHOUT BYPASS

	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
Nominal Conditions	350.9	121.0	1259	0.32
Increased Core Flow (ICF)	377.9	121.5	1260	0.33
ICF and Feedwater Heaters Out of Service (FHOOS)	236.6	120.1	1229	0.30

Note: All analyses performed with bounding (not statistically based) input.

TABLE 3.3 FEEDWATER CONTROLLER FAILURE TO MAXIMUM DEMAND

	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
Nominal Conditions	226.3	116.2	1198	0.21
Increased Core Flow (ICF)	240.5	117.0	1199	0.23
ICF and Feedwater Heaters Out of Service (FHOOS)	226.5	117.6	1157	0.21

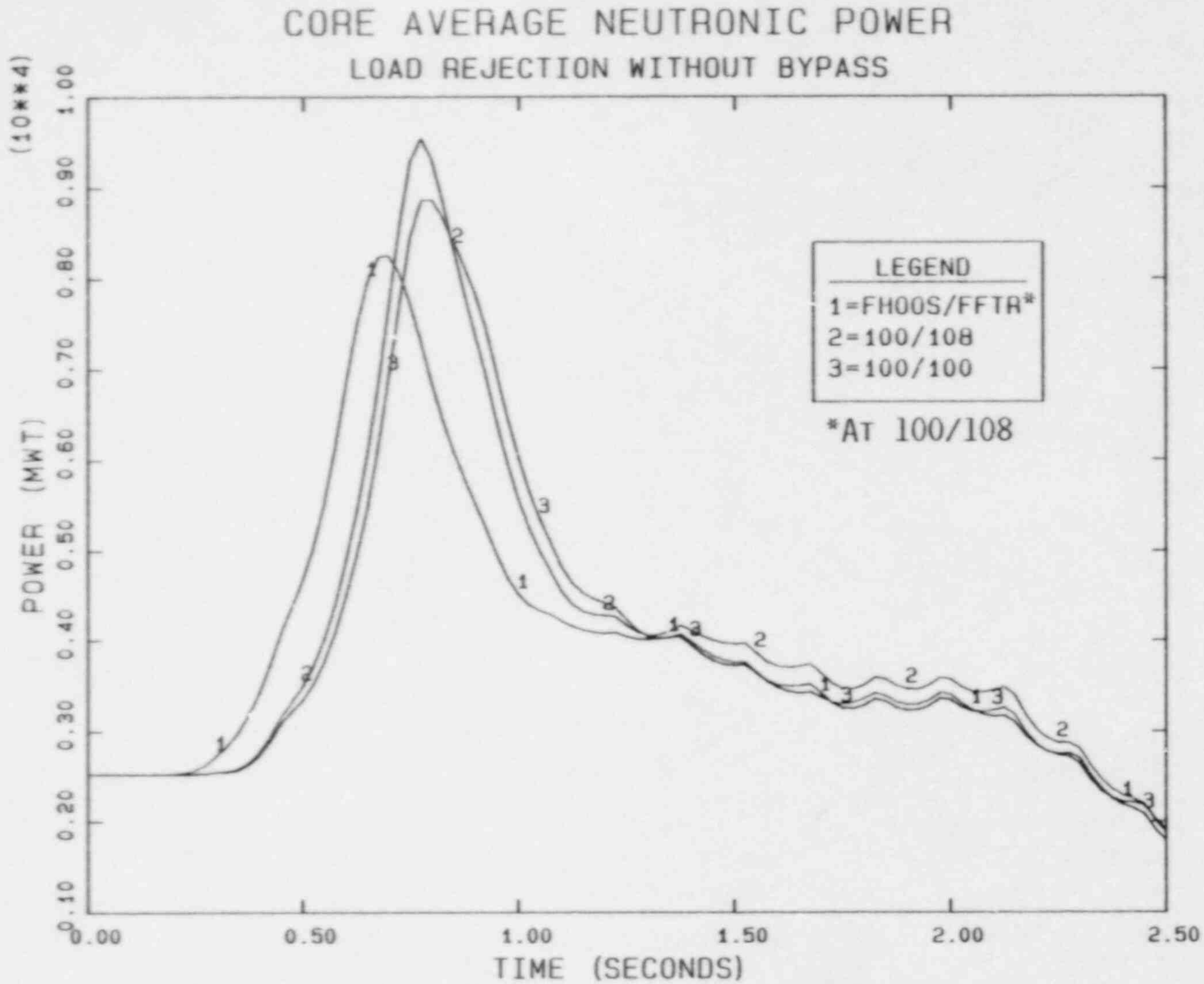


FIGURE 3.1 LOAD REJECTION WITHOUT BYPASS

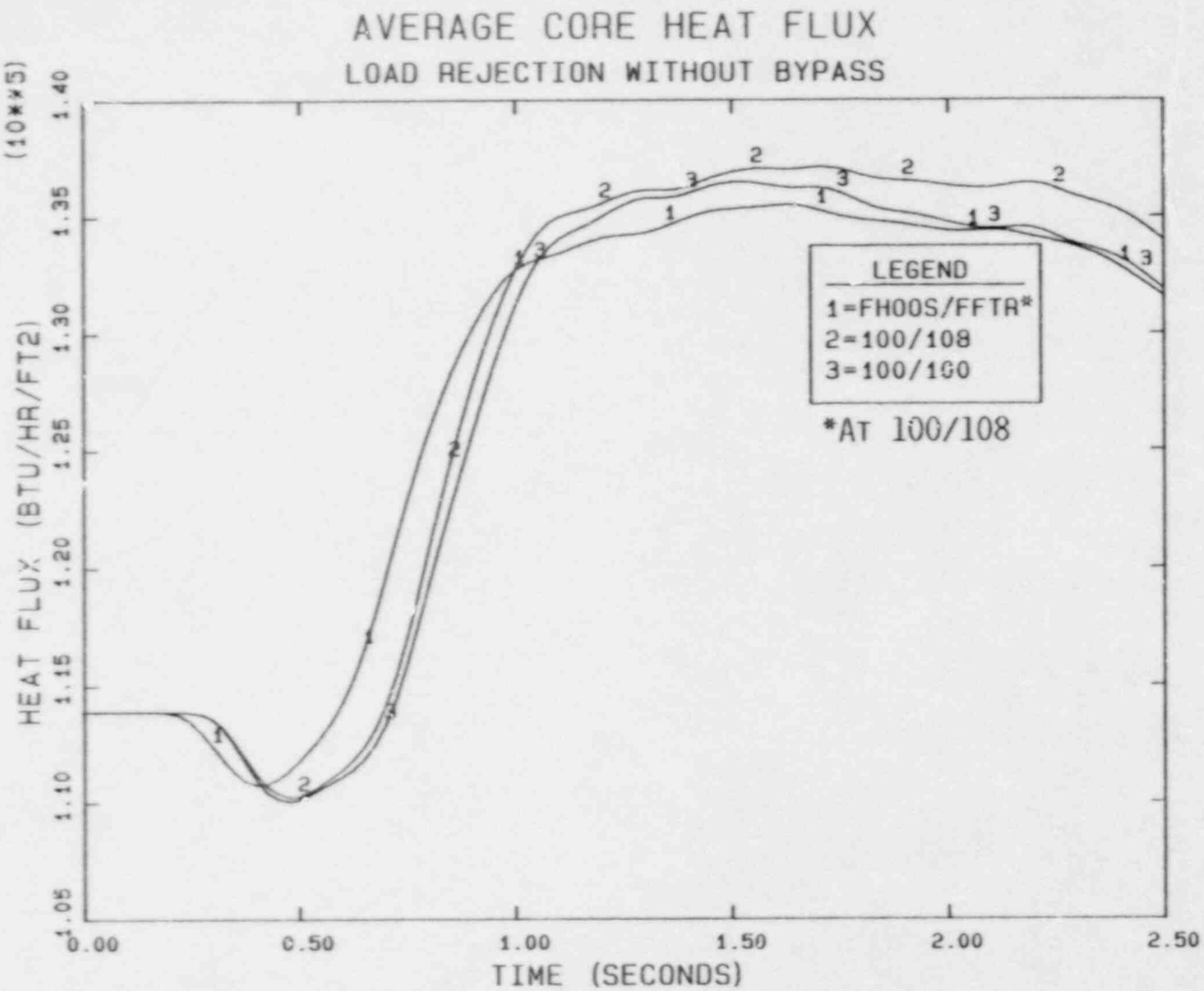


FIGURE 3.2 LOAD REJECTION WITHOUT BYPASS

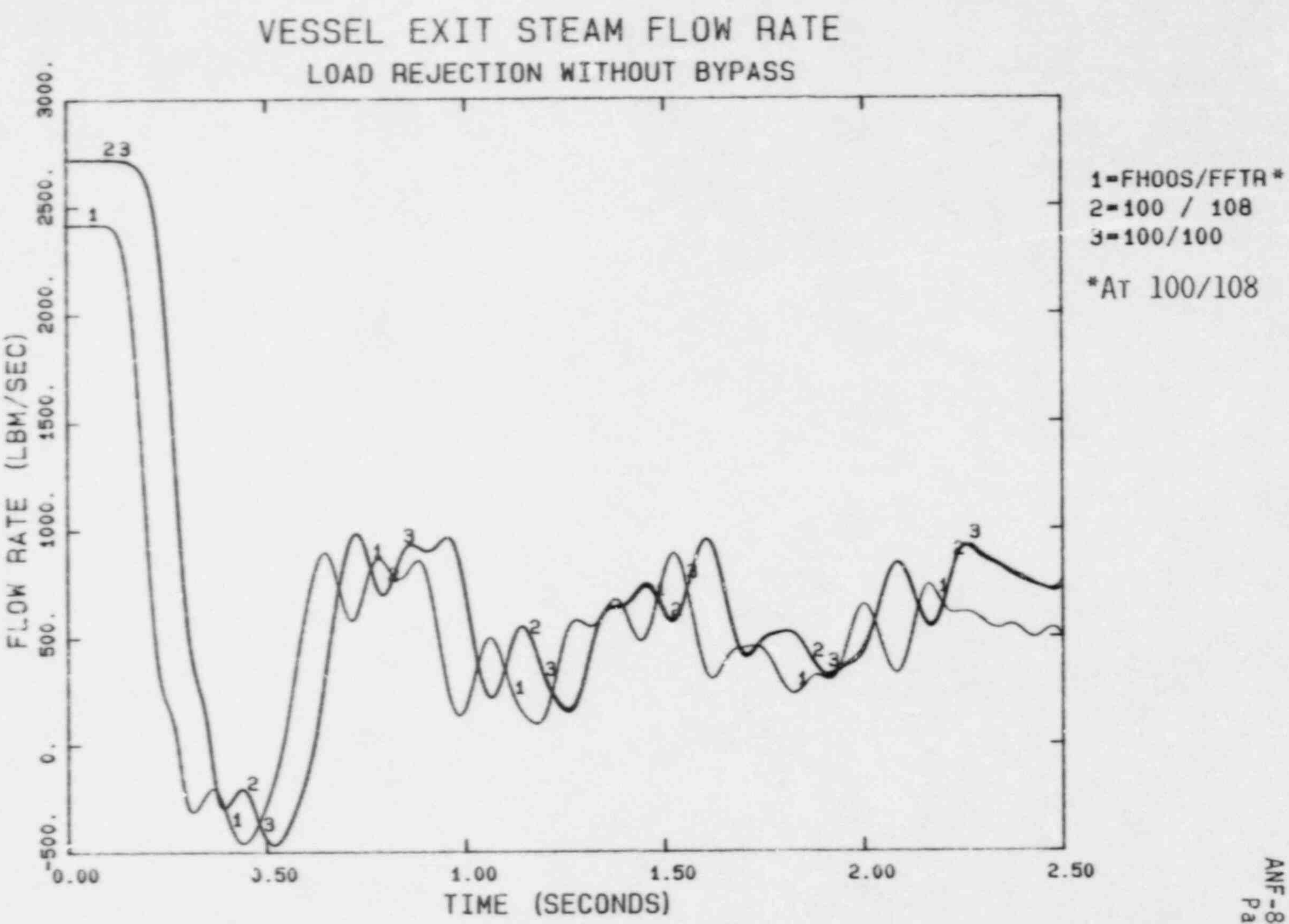


FIGURE 3.3 LOAD REJECTION WITHOUT BYPASS

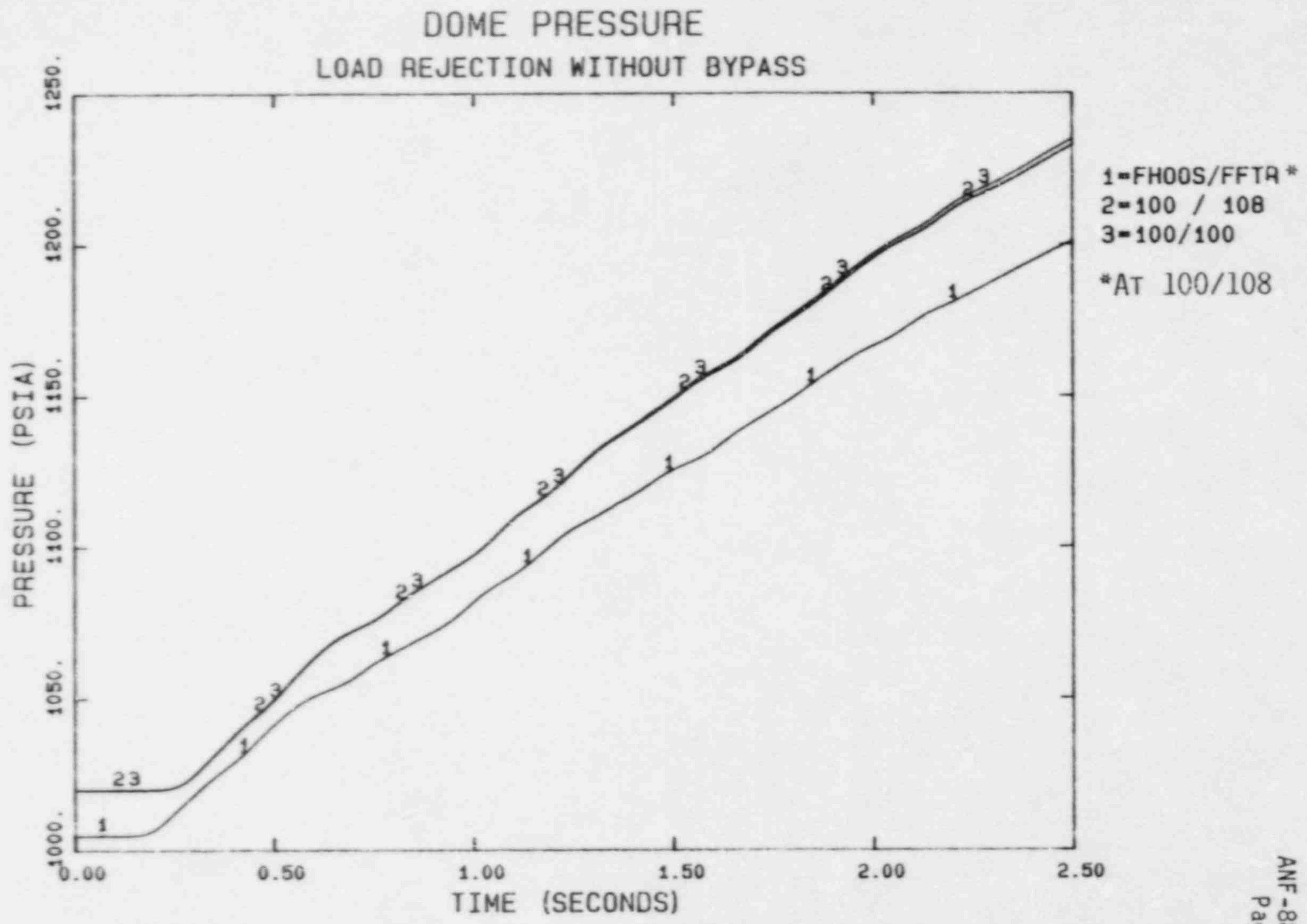


FIGURE 3.4 LOAD REJECTION WITHOUT BYPASS

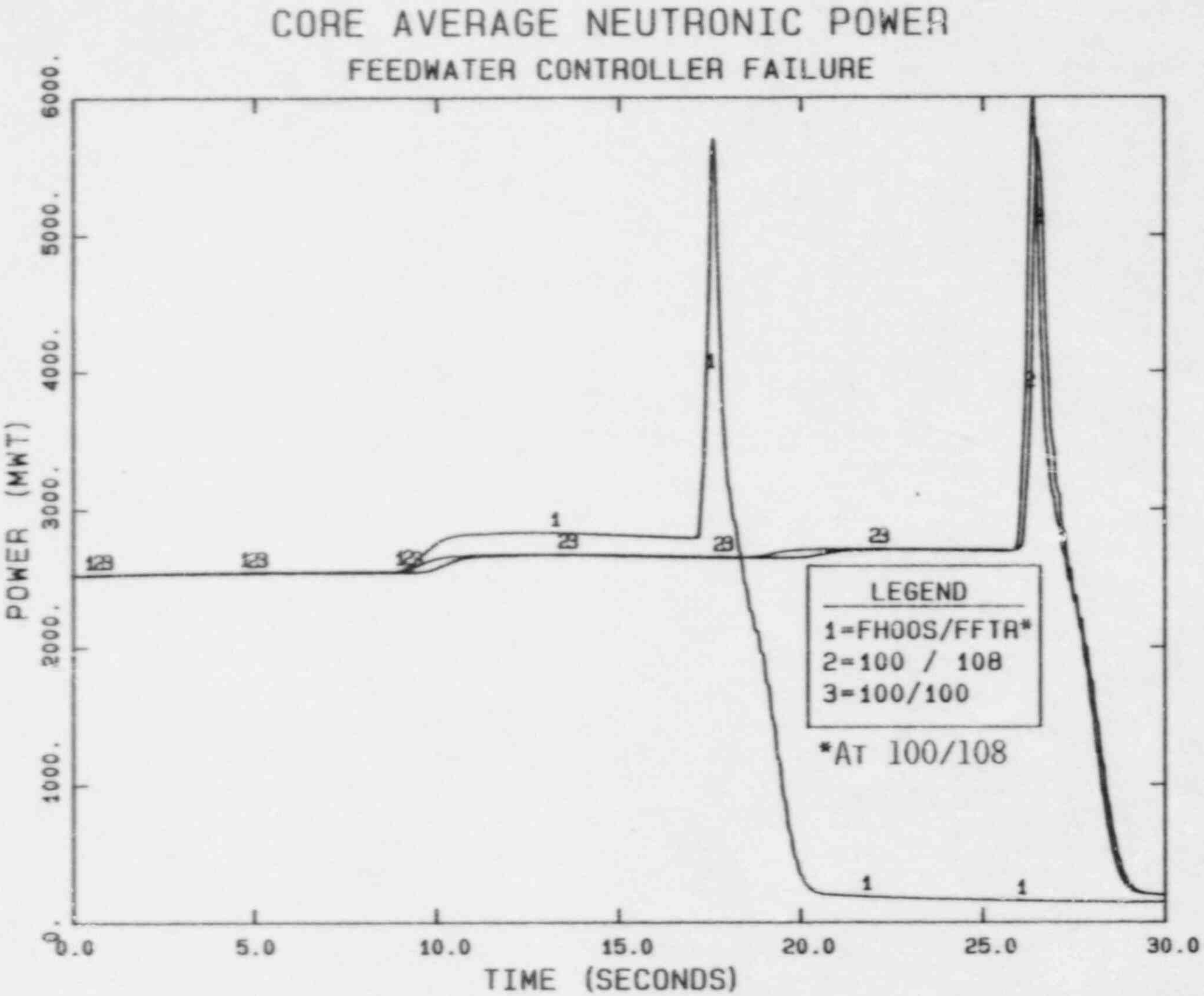


FIGURE 3.5 FEEDWATER CONTROLLER FAILURE

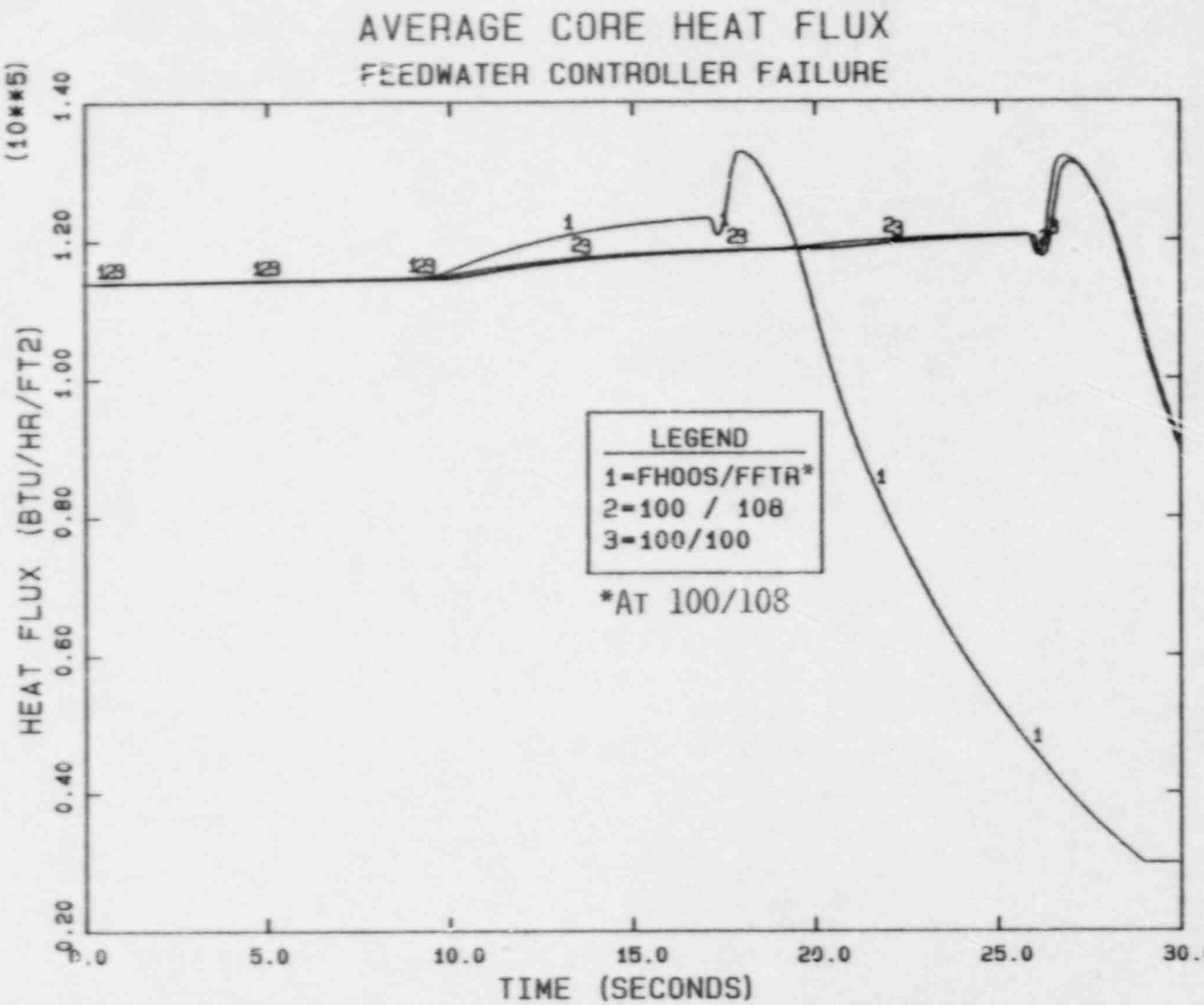


FIGURE 3.6 FEEDWATER CONTROLLER FAILURE

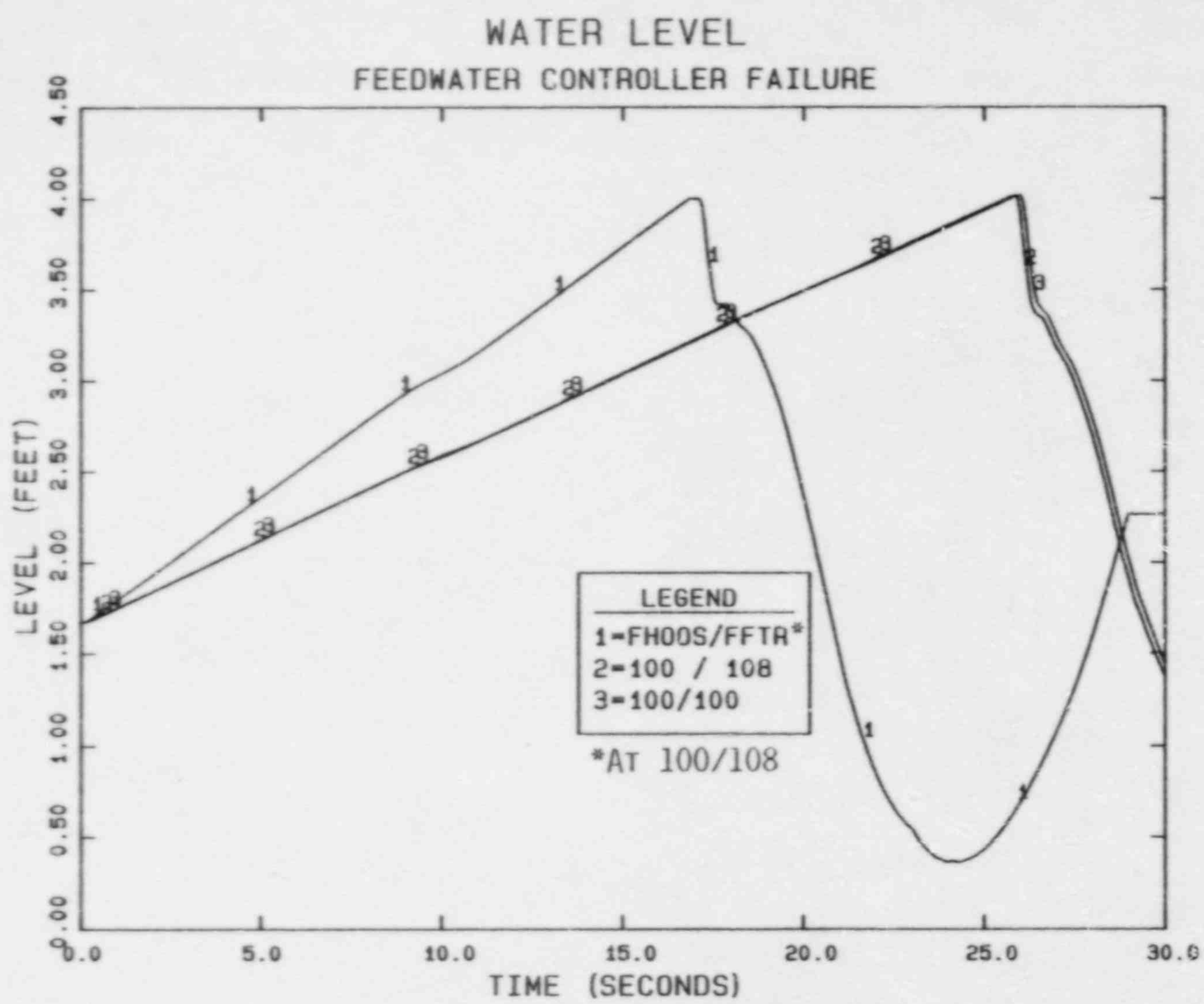


FIGURE 3.7 FEEDWATER CONTROLLER FAILURE

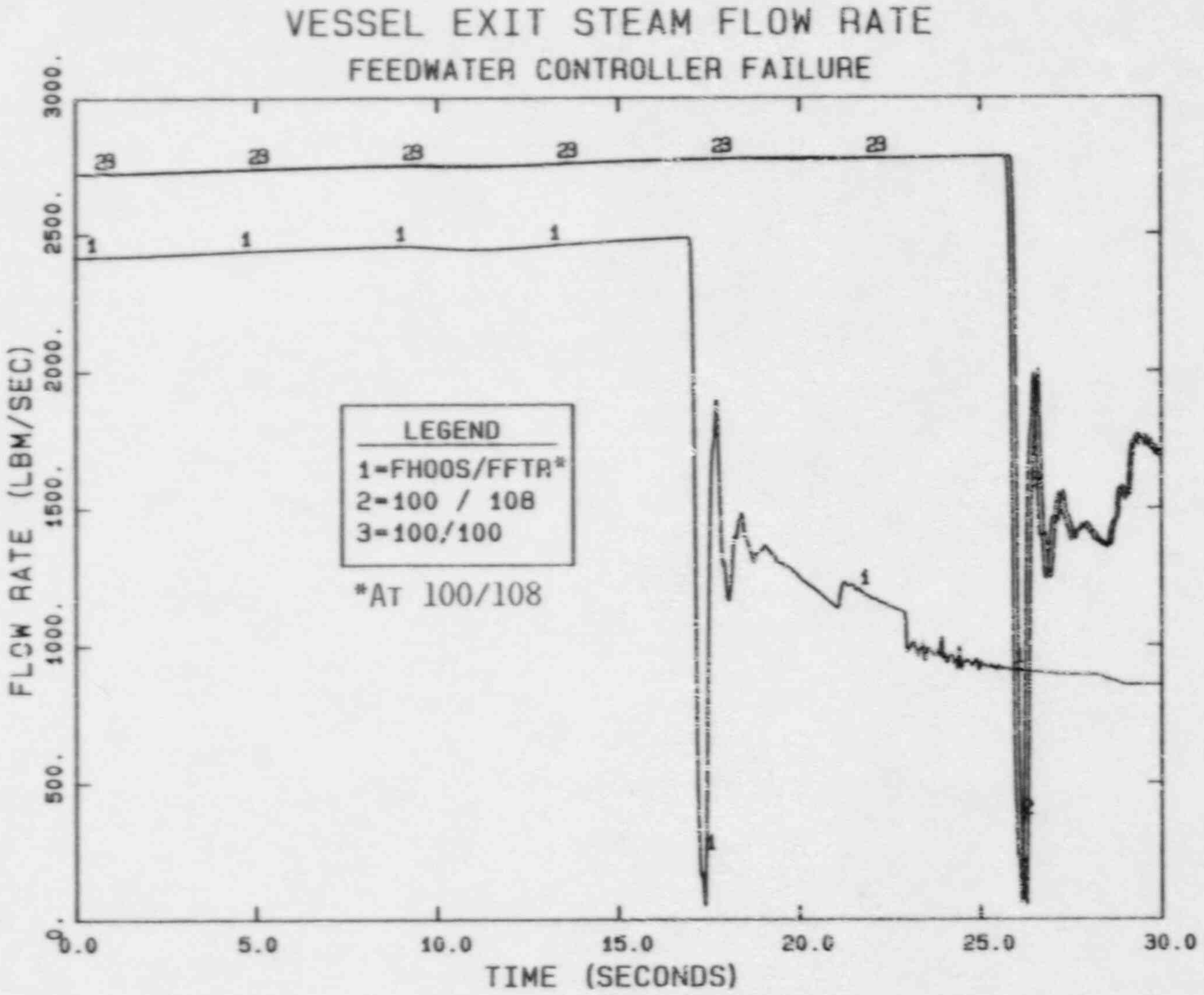


FIGURE 3.8 FEEDWATER CONTROLLER FAILURE

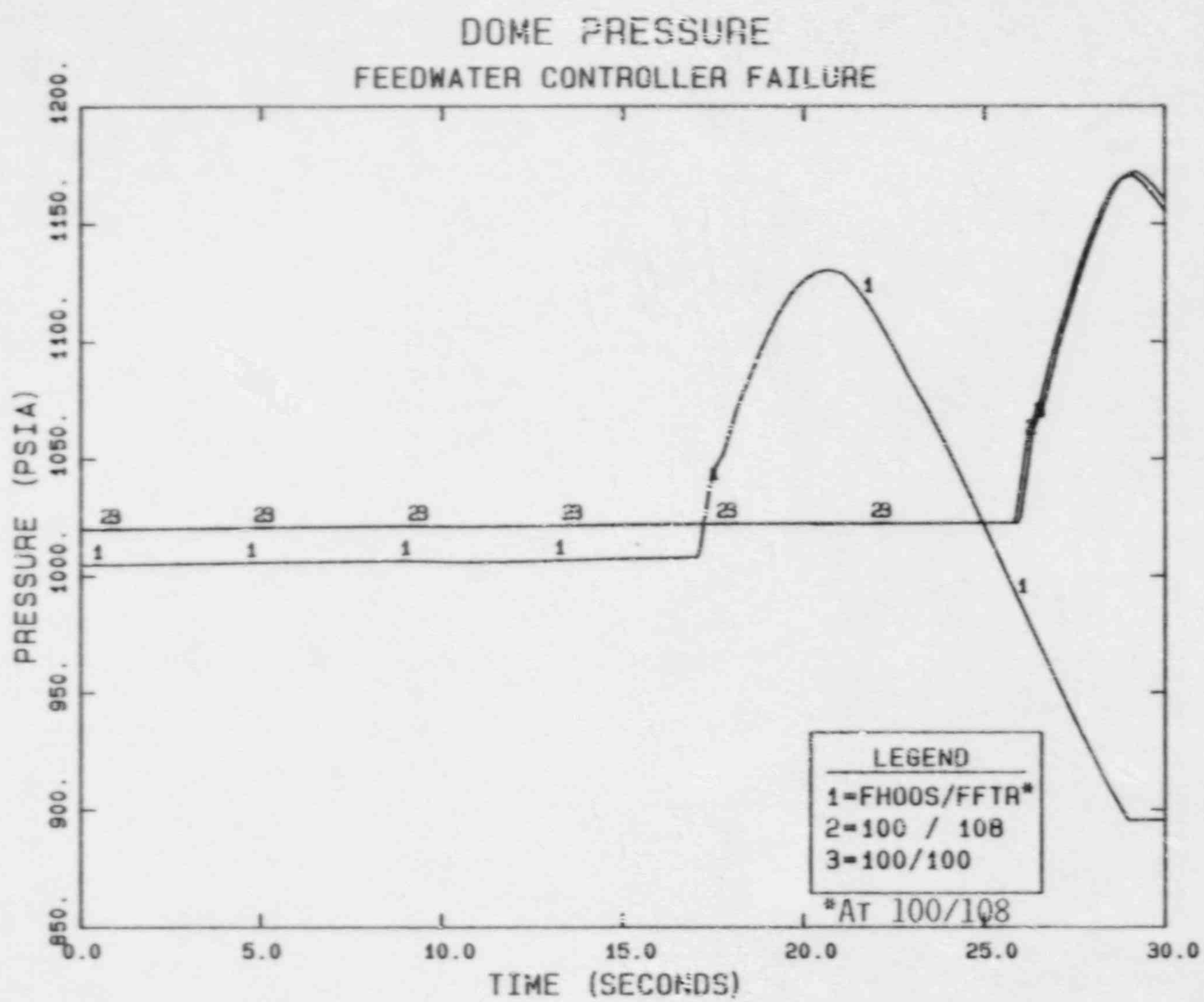


FIGURE 3.9 FEEDWATER CONTROLLER FAILURE

4.0 TRANSIENT ANALYSIS FOR THERMAL MARGIN DURING COASTDOWN

4.1 Design Bases

Economic considerations make operation of the plant past the end of full power capability desirable: this operation is coastdown. Because it occurs at the end of cycle, all rods are fully withdrawn. However, as the power decreases and flow is at its maximum, the axial power distribution shifts toward the top of the core. Two operational modes may exist, hold the reactor dome pressure at its rated condition or allow it to decrease. Both conditions are discussed.

4.2 Calculational Model

The assumptions and models described in Sections 3.1 and 3.2 are applicable to the following discussions. The coastdown analyses used bounding input.

4.3 Anticipated Transients

As discussed in Section 3.3, the limiting events for the Dresden units are the load rejection without bypass and the feedwater controller failure to maximum demand. Because the phenomena that cause these events to be limiting are system related, changing core power will not affect which transients are limiting within each category. Comparisons of the load rejection without bypass and the feedwater controller failure transients show that the load rejection without bypass remains the limiting transient for coastdown operation.

4.3.1 Load Rejection Without Bypass Valve Operation

Section 3.3.1 describes the phenomena occurring during a generator load rejection without bypass valve operation. The previous analyses demonstrated that the ICF case is the most limiting; therefore, this analysis was performed at ICF. Table 4.1 and Table 4.2 are comparisons of generator load rejection transients at increased core flow during coastdown operation allowing dome pressure to vary for Dresden Unit 3 Cycle 11 and for Dresden Unit 2 Cycle 12, respectively. The analysis trend for the two plants is the same; there is a slight increase in delta CPR as the power decreases to 60%.

This trend is counter intuitive. That is, logic would say that because of the lower steam flow the pressurization portion of the transient should be less severe. Analyses at less than rated power and flow have always shown this (Ref. 1). Figures 4.1 through 4.5 present normalized comparisons of the generator load rejection at the 100% power and 80% power with 108% flow cases. The important factor to consider, however, is the axial power shift. Shifting the axial power higher in the core results in a less effective scram. Figure 4.2 shows that on a normalized to initial power basis the 80% power case has a larger power increase in the top of the core than the 100% power case. This results in the larger delta CPR when the hot channel is forced to reach critical heat flux. However, it should be noted that as the power decreases during coastdown, the margin to limits increase. Table 4.3 presents actual core follow data for Dresden Unit 2 Cycle 9. It is seen that although the delta CPR increases by 3.0% as the power decreases to 80%, the margin increases almost 11% for about the same power change. Therefore, no special limits are required for coastdown operation.

Table 4.4 presents a comparison of the LRWB transient performed at 80% power and 108% flow at rated dome pressure and reduced dome pressure. The reduced dome pressure in fact has a slightly larger delta CPR than the rated pressure case (0.005), but this is not seen due to the conservative rounding up of all delta CPR results. Therefore, maintaining the coastdown limits based on the reduced pressure case bounds the rated pressure case.

4.3.2 Feedwater Controller Failure

Section 3.3.2 describes the feedwater controller failure to maximum demand. The analyses for coastdown were performed at ICF and ICF with FHOOS/FFTR. The results of these analyses are presented in Table 4.5. Because the bypass valve operation mitigates the impact of the pressurization event, the feedwater controller failure is bounded by the load rejection without bypass for all cases.

TABLE 4.1 LOAD REJECTION WITHOUT BYPASS DURING COASTDOWN
DRESDEN UNIT 3 CYCLE 11

Power/Flow	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
100%/108%	377.9	121.5	1260	0.33
80%/108%	286.4	98.1	1195	0.34
60%/108%	192.3	74.2	1125	0.35
40%/108%	97.4	48.7	1062	0.31

TABLE 4.2 LOAD REJECTION WITHOUT BYPASS DURING COASTDOWN
DRESDEN UNIT 2 CYCLE 12

Power/Flow	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
100%/108%	392.5	120.3	1305	0.33
80%/108%	301.6	97.1	1194	0.34
60%/108%	194.2	73.8	1124	0.34
40%/108%	98.4	48.2	1062	0.30

TABLE 4.3 DRESDEN 2 CYCLE 9 COASTDOWN CORE FOLLOW DATA

MWd/MT	MWt	Dome Pressure	Flow (Mlb/hr)	CPR
7597	2334	1008 psia	98.0	1.65
7831	2188	1020 psia	97.8	1.74
8016	2131	1016 psia	97.7	1.79
8351	2050	1019 psia	93.3	1.83

TABLE 4.4 LOAD REJECTION WITHOUT BYPASS DURING COASTDOWN 80% POWER/108% FLOW

Dome Pressure	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
1020 psia	261.0	97.7	1219	0.332
992 psia	286.4	98.1	1195	0.337

TABLE 4.5 FEEDWATER CONTROLLER FAILURE DURING COASTDOWN

Power/Flow	Maximum Neutron Flux % of Rated	Maximum Core Average Heat Flux % of Rated	Maximum Vessel Pressure (psia)	Limiting Fuel ΔCPR
100%/108%	240.5	117.0	1199	0.23
80%/108%	193.5	94.5	1120	0.23
60%/108%	116.3	65.8	1043	0.14
40%/108%	57.6	42.2	993	0.06
100%/108%*	226.5	117.6	1157	0.21
80%/108%*	193.8	94.7	1084	0.23
60%/108%*	102.5	64.4	1024	0.09
40%/108%*	53.7	41.9	990	0.04

*With FHOOS/FFTR

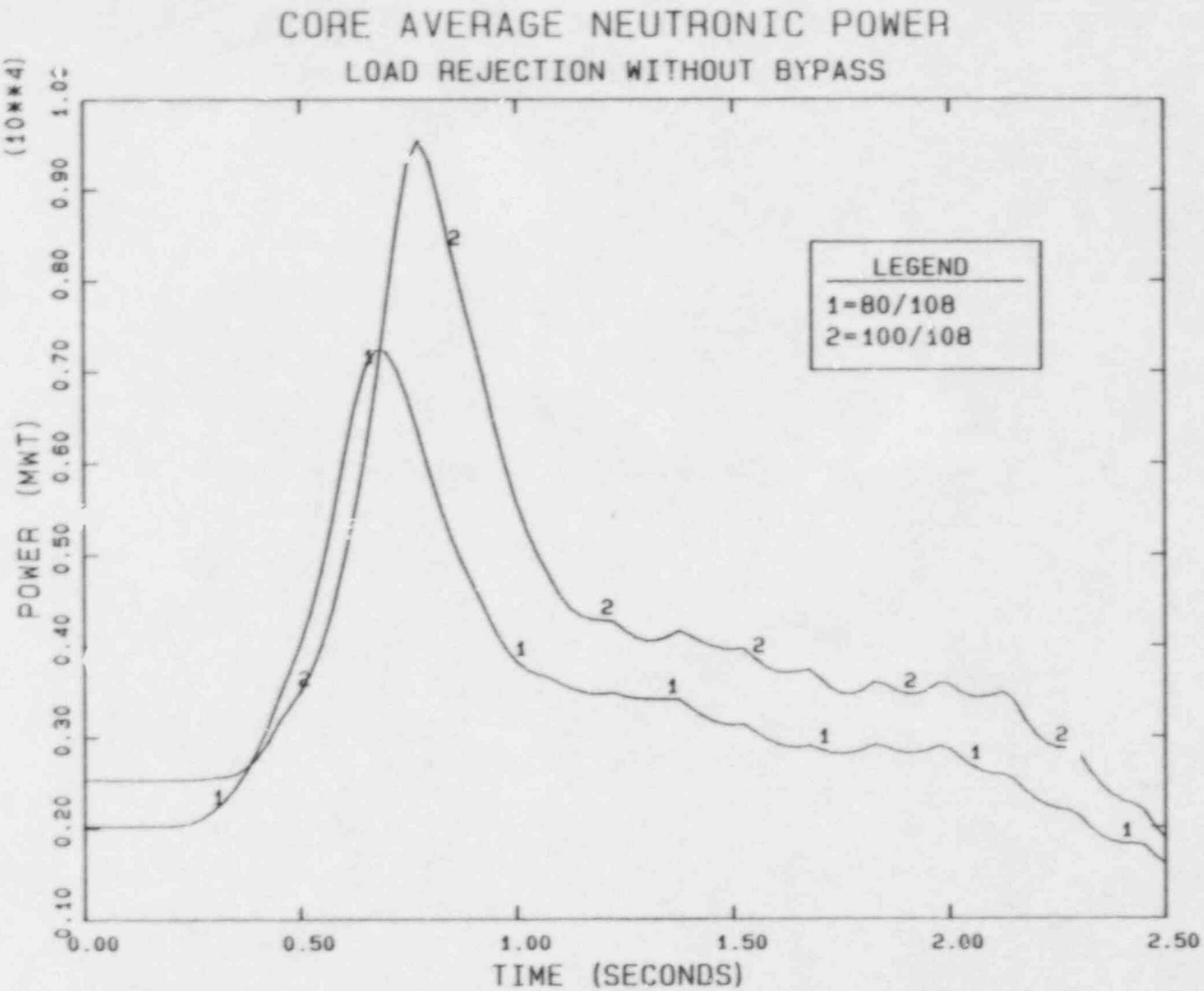
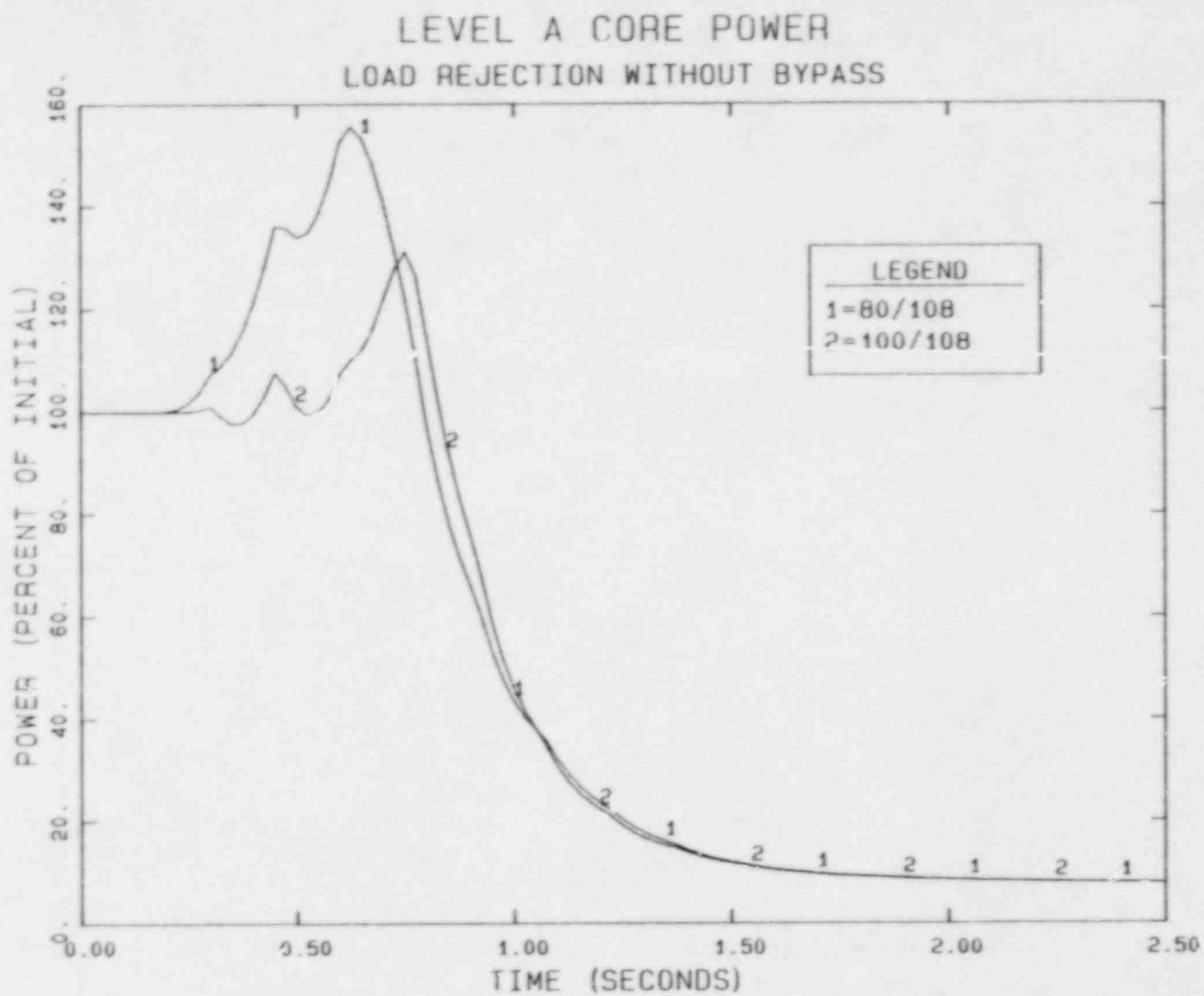


FIGURE 4.1 LOAD REJECTION WITHOUT BYPASS



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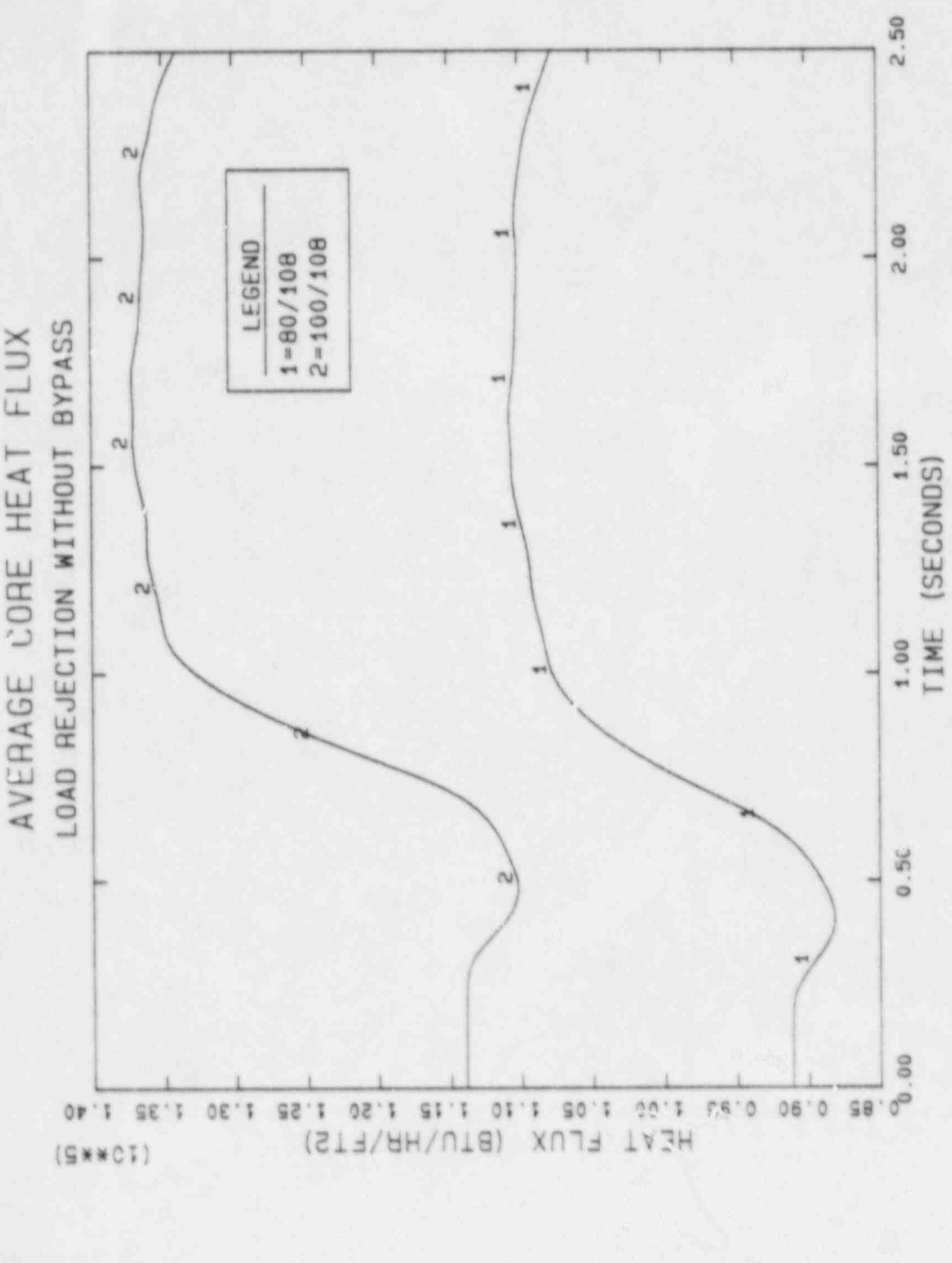

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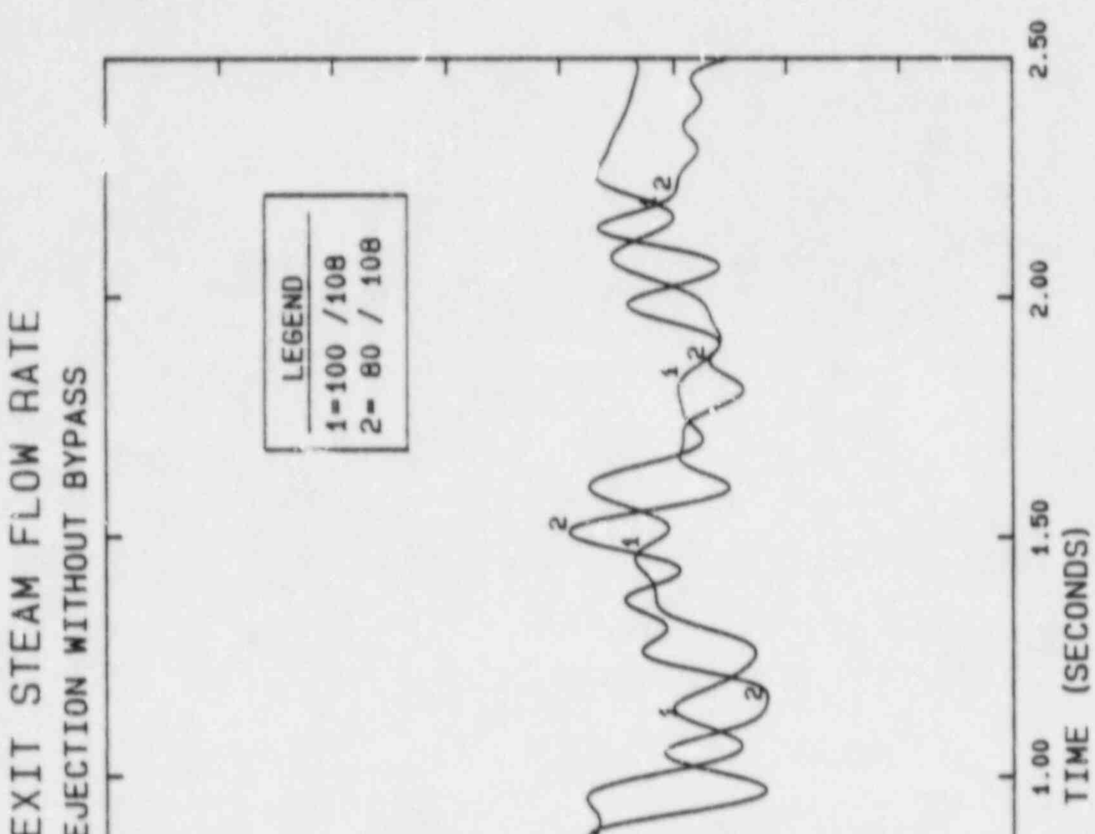

FIGURE 4.1 LOAD REJECTION WITHOUT BYPASS

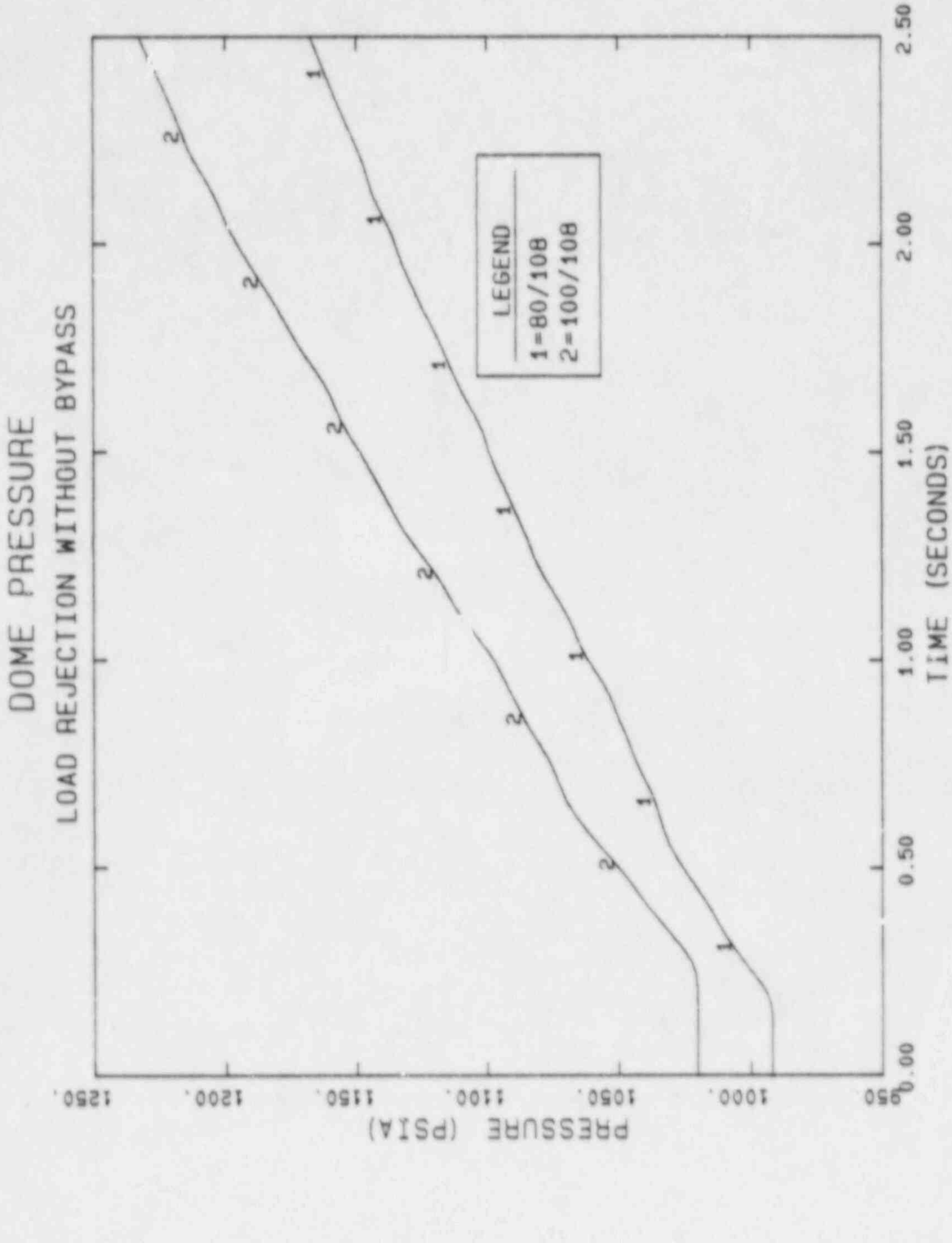

FIGURE 4.5 LOAD REJECTION WITHOUT BYPASS

5.0 TRANSIENT ANALYSIS FOR LOSS OF FEEDWATER HEATING

5.1 Design Basis

Plant transient analysis for the Dresden Units has shown that the most limiting event for an increase of recirculating vessel coolant subcooling is the loss of feedwater heating transient. The reactor plant conditions for the analysis are the nominal conditions shown in Table 3.1.

5.2 Calculational Model

The core average plant transient methodology described in References 2 and 4 as updated in Appendix A of Reference 1 was used for the analysis of the loss of feedwater heating transient. The PTSBWR point kinetics model was used for core and system response. The fuel pellet to clad gap conductance values used in the analysis are based on RODEX2 for Dresden core configurations. Because of the slow nature of this event, the delta CPRs are determined using a quasi-steady-state analysis with XCOBRA.

This analysis is then compared to the XTGBWR analysis for the loss of feedwater heating.

5.3 Loss Of Feedwater Heating Transient

The loss of feedwater heating leads to a gradual subcooling of the water in the lower plenum. Core power slowly increases to the overpower trip setpoint. The gradual power change allows the fuel thermal response to maintain pace with the increase in neutron flux. This analysis conservatively assumed that the feedwater temperature dropped 200°F over a two-minute period. Void reactivity is assumed to be 25% more negative than the nominal value, which results in a maximum value of power and heat flux. Scram performance is assumed to be 20% less than the nominal value.

Table 5.1 presents a summary of the loss of feedwater heating analysis results using PTSBWR. The Dresden Unit 2 Cycle 12 analysis also was performed using the three dimensional nodal core simulator code XTGBWR (Ref. 6). The result of this analysis is a delta CPR of 0.14. As is seen from Table 5.1, if the delta CPR for the loss of feedwater heating is set to 0.20 for both Dresden Units, all past and future cycles will be bounded.

TABLE 5.1 LOSS OF FEEDWATER HEATING

Dresden Unit	Cycle	Void Reactivity($/void fraction)*	Limiting ΔCPR
3	8	-15.89	0.16
3	9	-15.81	0.16
3	10	-15.14	0.20
3	11	-16.55	0.19
2	9	-16.40	0.16
2	10	-16.40	0.20
2	11	-15.14	0.19
2	12	-16.78	0.18

*Nominal Value

6.0 MAXIMUM OVERPRESSURIZATION ANALYSIS

This section describes the analysis of the maximum overpressurization accident performed to assure maximum vessel pressure will not exceed 110% of the design value for compliance with the ASME code.

6.1 Design Basis

The evaluation of the maximum pressurization event was performed with the reactor conditions summarized in Table 3.1. The same bounding conditions as those used in the transient analysis were assumed. In addition, further conservatism was added by not allowing the operation of the four power actuated relief valves as required by the ASME.

6.2 Pressurization Events

The general categories of expected maximum pressurization events are partial or total isolation of the turbine or containment and loss of offsite power. Generally, the condition in which the greatest energy is generated within the smallest confinement will result in the maximum pressurization.

Previous analyses have determined that the maximum pressurization transients for the Dresden units is the inadvertent closure of all MSIVs with failure of direct scram (Ref. 7). The position scram, which commands reactor shutdown upon MSIV movement, mitigates the effects of this event to the point that it does not contribute to the determination of thermal margins. Delaying the scram until the high pressure trip setpoint is reached results in a substantially more severe transient.

Although the closure rate of the MSIVs is substantially slower than that of the turbine stop or control valves, the compressibility of the fluid in the steam lines provides significant damping of the compression wave associated with the turbine trip events to the point that the slower MSIV closure without the direct scram results in nearly as severe a compression wave. Once the containment is isolated, the subsequent core power production must be contained in a smaller volume than the associated turbine trip events.

Analyses have demonstrated that the containment isolation event under these conservative assumptions results in a higher overpressure than total isolation of the turbine.

6.3 Closure Of All Main Steam Isolation Valves

This calculation assumed that all four steam lines were isolated at the containment boundary within three seconds. The valve characteristics and steam compressibility combine to delay the arrival of the compression wave at the core until approximately three seconds from the initiation of the MSIV stroke.

Table 6.1 presents the results of the comparison case for ICF, ICF and FHOOS, and full power part flow. The most limiting conditions for this event are the increased core flow and normal feedwater temperature case. This conclusion is valid for both Dresden units.

TABLE 6.1 ASME OVERPRESSURE EVENT

Power/Flow %	Maximum Neutronic Flux (% rated)	Maximum Heat Flux (% rated)	Maximum Vessel Pressure (psig)
100/108	439.0	133.9	1324
100/108 FHOOS	360.5	129.2	1304
100/87	412.2	129.2	1324

7.0 REFERENCES

1. T. H. Keheley, "Dresden Unit 3 Cycle 10 Plant Transient Analysis," XN-NF-85-62, Exxon Nuclear Co., Inc., Richland, WA 99352, September 1985.

2. R. H. Kelley, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71(P), Revision 2 (as supplemented), Exxon Nuclear Co., Inc., Richland, WA 99352, November 1981.

3. K. R. Merckx, "RODEX2 Fuel Rod Thermal Mechanical Response Evaluation Model," XN-NF-81-58(A), Revision 2, Exxon Nuclear Co., Inc., Richland, WA 99352, March 1984.

4. J. A. White, "Exxon Nuclear Methodology For Boiling Water Reactors; THERMEX Thermal Limits Methodology; Summary Description," XN-NF-80-19(P)(A), Volume 3, Revision 2, Exxon Nuclear Co., Inc., Richland, WA 99352, January 1987.

5. "XCOBRA-T: A Computer Code for BWR Transient Thermal-Hydraulic Core Analysis," XN-NF-84-105(P), Volume 1 and Supplements 1 and 2, Exxon Nuclear Co., Inc., Richland, WA 99352, June 1985.

6. D. J. Braun, "Dresden Unit 2 Cycle 12 Plant Transient Analysis," ANF-88-052, Advanced Nuclear Fuels Corporation, Richland, WA 99352, May 1988.

7. R. H. Kelley, "Dresden Unit 3 Cycle 8 Plant Transient Analysis Report," XN-NF-81-78, Revision 1, Exxon Nuclear Co., Inc., Richland, WA 99352, December 1981.

ANF-88-069
Issue Date: 7/13/88

EXTENDED OPERATING DOMAIN/EQUIPMENT

OUT OF SERVICE ANALYSIS FOR DRESDEN

UNITS 2 AND 3

DISTRIBUTION

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Attachment 5

GE Nuclear Energy

July 26, 1988
REP:88-161

cc: W. F. Naughton

Mr. R. A. Roehl
Supervising Fuel Buyer
COMMONWEALTH EDISON COMPANY
Fuel Department, 234 E
P. O. Box 767
Chicago, IL 60690

SUBJECT: Correction to Dresden 2 Cycle 12 Alternate Water Chemistry LTA's MAPLHGR Curve

REFERENCES:
1. Test and Inspection Agreement between Commonwealth Edison Company and General Electric Company dated May 28, 1975.
2. Letter from R. A. Roehl to R. E. Parr, "Dresden 2 Cycle 12 Alternate Water Chemistry LTA's Exposure Limits," RAR:88-193, May 19, 1988.
3. Letter from R. E. Parr to R. A. Roehl, (Telecopied July 19, 1988) "Reevaluation of Dresden 2 Cycle 12 Alternate Water Chemistry LTA's Exposure Limits," REP:88-159, July 19, 1988.
4. Loss Of Coolant Accident Analysis For Quad Cities Units 1 & 2 and Dresden Units 2 & 3, NEDO-24146, Rev 1, April 1979.

ATTACHMENTS: Corrected MAPLHGR Curve for Dresden-2 Alternate Water Chemistry LTA Bundles

Dear Mr. Roehl:

The attached MAPLHGR curve is the corrected version of the composite limiting MAPLHGR curve for the Dresden HWC LTA's. Please replace the MAPLHGR curve which was sent to you in Reference 3 with the attached curve. The corrected MAPLHGR curve is consistent with the GE LOCA analysis for the Dresden Unit 2 HWC LTA's (Reference 4).

Please note that the attached information is proprietary to the General Electric Company and should be controlled pursuant to Article XVIII (Proprietary Data And Access) of the January 6, 1986 Contract.

Very truly yours,

J. S. Worthington for

R. E. Parr
Senior Fuel Project Manager
Edison Projects
M/C 174; (408) 925-6526

GENERAL ELECTRIC COMPANY
PROPRIETARY INFORMATION

MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE
DRESDEN UNIT 2 CYCLE 12
LTA BUNDLES LY5455, LY5456, LY5457, LY5458

AVERAGE PLANAR EXPOSURE (GWd/ST)	MAPLHGR (KW/FT)
0.2	11.5
1.0	11.6
5.0	11.9
10.0	12.1
15.0	12.1
20.0	11.9
25.0	11.3
30.0	10.7
35.0	10.2
41.6	8.8

Attachment 6

Discussion of Previous SER Topics

The following discusses several topics raised in previous generic and reload SERs for Dresden and is based on information provided by Advanced Nuclear Fuels.

1. Rod Bow Considerations

   During the review of the previous reload submittal, the NRC SER placed an exposure cap on ANF 8x8 and 9x9 fuel due to rod bow considerations. The limit was set at 30,000 MWd/Mt for ANF 8x8 fuel and 23,000 MWd/Mt for ANF 9x9 fuel (batch average exposure). This condition was eliminated for both ANF 8x8 and 9x9 fuel when ANF received the NRC SER for XN-NF-82-06, Supplement 1, Revision 2 in May of 1988. This SER, which references mechanical design analyses of 35,000 MWd/Mt peak assembly exposure for ANF 8x8 and 40,000 MWd/Mt peak assembly exposure for ANF 9x9 fuel in XN-NF-85-67(P)(A), Revision 1, establishes ANF's currently approved mechanical design limits.

2. Extended Burnup Approval Status (XN-NX-82-06)

   Qualification of Exxon Nuclear Fuel for Extended Burnup (XN-NF-82-06(P)(A) w/ Supplements 2, 4, and 5, Revision 1) was approved in July of 1986 by the NRC. Extended burnup qualification for 9x9 fuel (XN-NF-82-06(P)(A) Supplement 1, Revision 2) was approved in May of 1988 by the NRC.

3. Conditions on Critical Power Methods (XN-NF-524, Rev. 1)

   In the Safety Evaluation prepared by the Core Performance Branch covering Revision 1 to XN-NF-524(P)(A), "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors", the NRC Staff identified four conditions to be met during the application of the subject methology under the generic approval granted by the SER. The steps taken during the analysis to assure compliance with these conditions are described below.

   CONDITION 1: Each plant specific application must contain the data used to generate the uncertainties employed in the methology.

   The uncertainties used in the Dresden Unit 2 MCPR safety limit calculations are the same as the uncertainties which have been used in pervious Dresden analyses. The two loop uncertainties are discussed below; the single loop uncertainties are the same except as described in the Cycle 11 reports.

   Plant measurement uncertainties which are not fuel-dependent were taken from approved NSSS supplier generic documents applicable to Dresden. As identified in XN-NF-524(P)(A), Revision 1, specific uncertainty values used in the analysis were a feedwater flow rate uncertainty of 1.76%, a feedwater temperature uncertainty of 0.76%, a core pressure uncertainty of 0.5%, and a total core flow rate uncertainty of 2.5%. The generic core inlet temperature uncertainty of 2.0% was conservatively replaced with an uncertainty of 2.4% on the core inlet enthalpy. These approved values were used as one-sigma uncertainties consistent with NEDO-24011. The nominal values, uncertainties, and statistical treatment of these measured plant parameters are summarized in Table 1.

The uncertainties associated with the XN-3 Critical Power Correlation are based on data contained in XN-NF-512(P)(A), Revision 1, and XN-NF-734(P)(A). The safety limit analysis was based on a one-sigma uncertainty value of 4.11% for the XN-3 correlation, consistent with the source documents noted above and with XN-NF-80-19(P)(A), Volume 4, Revision 1, which provides a generic description of the overall reload analysis. The correlation statistics were developed from the ANF's CHF data base, which includes test geometries which encompass the Dresden 8x8 and 9x9 fuel designs. XN-NF-734(P)(A) was issued explicitly to validate the XN-3 statistics for application to 9x9 fuel.

Power distribution measurement uncertainties are based on data contained in XN-NF-80-19(P)(A), Volume 1. The safety limit calculation was based on one-sigma uncertainies of 5.28% on radial peaking factor and 2.46% on local peaking factor consistent with the reference report. These uncertainties were developed based on analytical predictions of measured data for BWR fuel. The same methods used for the analytical predictions were used for the nuclear design analyses for Dresden; hence the generic uncertainty values are applicable to Dresden.

The correlation and power distribution measurement uncertainties and their statistical treatment for the Dresden analysis are summarized in Table 2.

CONDITION 2: All plant parameters that are not statistically convoluted must be placed at their limiting value.

In the performance of plant transient analyses, ANF uses design values for major process parameters for consistency with the FSAR analyses which are superseded by the ANF transient analyses. Design values are established by the plant designer as conservative predictions of the boundaries of the plant operating envelope, and may not be accurate predictions of actual plant operation. These values are used to assure a conservative calculation of the transient effects. Nominal values are best-estimate predictions of plant operating conditions. The use of nominal conditions is appropriate for the statistically treated parameters in the Monte Carlo analysis.

Input to the Monte Carlo calculation consists of three major classifications of data: heat balance information, power distribution information, and fuel geometric information.

Heat balance information consists of feedwater temperature and flow rate, core pressure and total flow rate, and core inlet enthalpy. All of these variables are considered statistically in the Monte Carlo analysis.

Power distribution information is taken from the fuel management analysis and consists of radial, axial, and local peaking factors. Radial and local peaking factors are considered statistically in the Monte Carlo analysis. For power distributions characterized by bottom-peaked core average axial power shapes, a limiting center-peaked axial distribution is used.

Fuel geometric information consists of fuel dimensions and hydraulic demand curves. Small variations in fuel dimensions with manufacturing tolerances are considered in the ANF pressure drop methodology and contribute to the flow distribution uncertainty. Hydraulic demand curves are used to determine fuel assembly flow rates as a function of bundle power; individual assembly flow rates are treated statistically in the Monte Carlo analysis.

CONDITION 3: Each application should demonstrate that the uncertainties in plant parameters are treated with at least a 95% probability at a 95% confidence level in accordance with Acceptance Criterion 1.0 of Standard Review Plan Section 4.4

The magnitude and nature of the uncertainties used in the Monte Carlo analysis have been established generically during the Staff review of ANF topical report XN-NF-524(P)(A), Revision 1, "Exxon Nuclear Critical Power Methodology for Boiling Water Reactors". A detailed review of the XN-3 correlation statistics was included in the review of ANF topical report XN-NF-80-19(P)(A), Volume 1, "Neutronics Methods for Design and Analysis". The conclusion that these uncertainties may be conservatively treated as normally distributed was addressed during the generic review.

Uncertainties in the measurement of plant parameters were taken from the NSSS supplier's generic reload submittal. Based on the Staff's approval of these uncertainties for use in the MCPR safety limit calculation, the ANF analyses used the published values as 95% confidence statistics. Process measurement uncertainties are generally characterized by a normal distribution; therefore, a normal distribution was used in the ANF analysis.

The Monte Carlo analysis was performed to demonstrate that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition at a confidence level of 95%. This conclusion conservatively assures that the boiling transition limitation will be protected during anticipated operational occurrences in which the MCPR safety limit is protected. The referenced Standard Review Plan section identifies this method as an acceptable approach to the 95/95 treatment of uncertainties.

CONDITION 4: Each application must present a goodness-of-fit analysis for the fitting of the Pearson curve in order to assure that the number of Monte Carlo trials used in establishing the safety limit MCPR are sufficient.

In the original ANF MCPR safety limit methodology, the first four statistical moments of the Monte Carlo output were used to define an output frequency distribution through fitting of Pearson functions. This approach was take to minimize the number of trials necessary in the Monte Carlo analysis. Revision 1 to XN-NF-524 abandoned this approach in favor of a distribution-independent method of assigning tolerance limits. The new approach required a larger number of Monte Carlo trials, but the end result was a conclusion which was independent of the Pearson functions.

Since the statistical analysis involved no fitting of standard functions to the Monte Carlo output, no goodness-of-fit analysis was provided. In the case of the Dresden analysis, 500 Monte Carlo trials were provided. In the non-parametric tables, an expected value may be established at a confidence level of 95% with as few as 50 trials.

Table 1

Plant Measurement Uncertainties

Parameter	Units	Nominal Value	Uncertainty % Nominal	Statistical Treatment
Feedwater Flowrate	Mlbm/hr	12.41*	1.76	Convoluted
Feedwater Temperature	deg F	340.1	0.76	Convoluted
Core Pressure	psia	1035	0.50	Convoluted
Total Core Flow	MLBM/hr	98.0	2.50	Convoluted
Core Inlet Temperature			0.20	Replaced by core inlet enthalpy
Core Inlet Enthalpy	Btu/lbm	522.3	0.24	Convoluted
Core Power	MW	3200*		Allowed to vary with heat balance

* Feedwater flowrate and core power were increased above design values to attain desired core MCPR for safety limit evaluation, consistent with XN-NF-524(P)(A), Revision 1.

Table 2

Fuel-Related Uncertainties

Parameter	Source Document	Uncertainty % Nominal	Statistical Treatment
XN-3 Correlation	XN-NF-512(P)(A) XN-NF-734(P)(A)	4.11	Convoluted
Radial Peaking Factor	XN-NF-80-19(P)(A) Volume 1	5.28	Convoluted
Local Peaking Factor	XN-NF-80-19(P)(A) Volume 1	2.46	Convoluted
Axial Peaking Factor	XN-NF-80-19(P)(A) Volume 1	2.99	Limiting Value
Assembly Flowrate	XN-NF-79-59(P)(A)	2.80	Convoluted