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DRESDEN UNIT 3 CYCLE 8 RELOAD ANALYSIS

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EXXON NUCLEAR COMPANY, Inc.

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EXXON NUCLEAR COMPANY, Inc.

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1.0 INTRODUCTION

This report describes the analyses performed by Exxon Nuclear Company (ENC) in support of the Cycle 8 (XN-1) reload for Dresden Unit 3, which is scheduled to commence operation in the Spring of 1982. Dresden 3 is the first BWR/3 to be licensed on the basis of ENC analyses. Some of these analyses may be referenced in subsequent ENC jet pump BWR submittals as generic to jet pump BWR installations in general and to BWR/3 plants in particular.

This report addresses the following areas:

a) Mechanical Design

b) Thermal-hydraulic Design

c) Nuclear Design

d) Anticipated Operational Occurrences

e) Postulated Accidents

f) Operating Limits

2.0 MECHANICAL DESIGN

The mechanical design of BWR fuel fabricated by ENC is described on a generic basis in XN-NF-81-21(P), "Generic Design Report - Mechanical Design for Exxon Nuclear Jet Pump BWR Fuel Assemblies," dated November 1981 (Reference 9.1). This reference document addresses design bases, descriptions and design drawings, design evaluations, and plans for testing, surveillance, and inspection. The design bases and evaluations establish criteria for the determination of fuel system damage and assure (1) that normal operation and anticipated operational occurrences do not result in the violation of any of the established criteria, and (2) that ENC analyses of postulated accidents do not underestimate the number of fuel rod failures.

The mechanical design of ENC XN-1 8x8 fuel is covered by the generic analyses and evaluations in Reference 9.1.

3.0 THERMAL HYDRAULIC DESIGN

This section describes the thermal hydraulic design of the ENC XN-1 8x8 fuel which will be used for Cycle 8 and subsequent cycles in Dresden 3. Design criteria are identified and evaluations in support of those criteria are reported. The evaluations have shown that the subject fuel design satisfies the established criteria in the areas of hydraulic compatibility, thermal margin performance, fuel centerline temperature, rod bow, and bypass flow. In addition, this section reports the MCPR Fuel Cladding Integrity Safety Limit for Dresden 3 Cycle 8, which was determined to be 1.05.

3.1 DESIGN CRITERIA

Primary thermal hydraulic design criteria of Exxon Nuclear Company reload fuel for BWR's which are applicable to Dresden 3 are as follows:

3.1.1 Hydraulic Compatibility

The hydraulic flow resistance of the reload fuel assemblies shall be similar to existing fuel in the reactor so that there is no significant impact on total core flow or the flow distribution among the assemblies in the core.

3.1.2 Thermal Margin Performance

Fuel assembly design shall minimize the likelihood of boiling transition during normal reactor operation and during anticipated operational occurrences. The fuel design shall fall within the limits of applicability of the XN-3 critical power correlation.

3.1.3 Fuel Centerline Temperature

Fuel design and operation shall be such that fuel centerline melting is not expected for anticipated operational occurrences.

3.1.4 Rod Bow

Anticipated magnitudes of rod bow shall not impact thermal margin in BWR fuel designs.

3.1.5 Bypass Flow

Bypass flow from reload fuel assemblies shall match design bypass flow from existing fuel to provide adequate flow in the bypass region.

3.2 HYDRAULIC CHARACTERIZATION

Component hydraulic flow resistances for the ENC XN-1 8x8 reload design and the GE 8x8R design have been determined in single phase flow tests of full scale assemblies. The GE 8x8R design is typical of 8x8 and P8x8R designs for purposes of hydraulic characterizations. Table 3.1 summarizes the component flow resistances for the two designs. In developing this table, the test results have been modified slightly to account for the differences between the tests and actual reactor operation.

The total loss coefficient of the inlet hardware, including the effects of the side entry inlet orifice loss lumped together with the lower tie plate losses, was determined in the flow tests. The loss coefficient of 3.6 for the lower plate shown in Table 3.1 for ENC fuel is adjusted for the geometry of the ENC XN-1 fuel design. The difference in lower tie plate loss coefficients between the ENC and the GE fuel designs is also indicated in Table 3.1. This result reflects the difference measured in the flow tests between the total inlet hardware loss of the GE design and the total inlet hardware loss of the ENC design.

3.3 HYDRAULIC COMPATIBILITY AND THERMAL MARGIN PERFORMANCE

Hydraulic compatibility as it relates to thermal margin performance and the relative thermal margin performance of the ENC XN-1 8x8 reload design and the GE 8x8R design have been determined with detailed thermal hydraulic analyses which have calculated critical power ratios (CPR's) of ENC XN-1 and GE 8x8R fuel types in difference core configurations. The GE 8x8R design is typical of 8x8 and P8x8R designs for purposes of hydraulic compatibility and thermal margin performance. The configurations analyzed are as follows:

Case 1. The Dresden 3 Cycle 8 mixed core configuration in which 224 ENC XN-1 reload fuel assemblies are coresident with existing GE fuel types including 200 GE 8x8R assemblies. The analysis of this configuration provides the relative thermal margin performance of ENC and GE 8x8R fuel designs in a representative mixed core configuration where each fuel type experiences the same core pressure drop.

Case 2. A modified Cycle 8 configuration in which the 224 ENC assemblies are replaced with GE 8x8R assemblies to yield an all GE core. Comparison of results from this case with those of the first case provides the impact of the ENC XN-1 reload on thermal margins for GE fuel.

Case 3. An all ENC configuration (i.e., Dresden 3 Cycle 11). Comparison of results from this case with those of the first case provides the impact of coresident GE fuel on thermal margins for ENC fuel.

The peak assembly powers of both the ENC XN-1 8x8 design and the GE 8x8R design are assumed to be the same in all cases (5.235 MWt) and correspond to an assembly radial peaking factor of 1.50.

The analyses were performed using the methodology described in References 8.6, 8.10 and 8.11 as implemented in ENC's XCOBRA thermal hydraulic program (Reference 9.2). Critical power ratios are calculated within XCOBRA using the XN-3 critical power correlation (Reference 8.9).

Table 3.2 summarizes the input conditions for the analysis. The core loading for the Cycle 8 mixed core configuration case (Case 1) is also defined. The analysis includes consideration of rod-to-rod local peaking impact on critical power for the ENC XN-1 and GE 8x8R fuel types. Figure 3.1 provides the axial power distribution applied to all fuel assemblies in the analysis.

Table 3.3 provides a summary of calculated thermal hydraulic parameters from the Dresden 3 Cycle 8 mixed core configuration case. The differences between ENC XN-1 and GE 8x8R parameters are small. As seen in Table 3.3, the flow to the maximum power ENC assembly is somewhat larger than the flow to the GE 8x8R assembly at the same power level. The higher flow for the ENC XN-1 8x8 design reflects the somewhat lower hydraulic resistance of this design versus the GE 8x8R.

Although the demand curves shown in Figure 3.2 are specific to the Dresden 3 Cycle 8 mixed core case, the relative comparison of ENC XN-1 and GE 8x8R designs is applicable in any core loading where these fuels are coresident. Figure 3.3 shows active region axial average void fraction versus assembly power for the ENC XN-1 and GE 8x8R designs.

Table 3.3 gives CPR values of 1.536 and 1.486 respectively for the maximum power ENC XN-1 and GE 8x8R assemblies in the Dresden 3 Cycle 8 mixed core loading case. The 3% higher operating critical power ratio for the ENC XN-1 design compared to the GE 8x8R design is primarily a result of its greater heat transfer area and higher flow.

3.4 FUEL CENTERLINE TEMPERATURE AT OVERPOWER

Fuel rod centerline temperatures are determined at 120% overpower conditions as a check against the occurrence of calculated centerline melting during anticipated operational occurrences. The conditions at 100% power for maximum temperature in the ENC XN-1 design are a rod exposure of 21,200 MWD/MT and a nominal peak power of 13.93 kw/ft. Fuel temperatures under these conditions envelope maximum temperatures expected at any exposure for ENC XN-1 reload fuel in the Dresden 3 reactor. The peak power at 120% overpower is 16.7 kw/ft. The peak centerline temperature at the 13.93 kw/ft (100%) nominal design condition was calculated to be 3909°F. The peak centerline temperature at 120% overpower (16.7 kw/ft) was calculated to be 4607°F. The melting point of UO_2 decreases 58°F per 10,000 MWD/MT. At 21,100 MWD/MT exposure, the melting point is about 4900°F so that the margin to centerline melt for the 120% overpower design condition is 293°F. All other expected operating conditions have greater margin to fuel centerline melting.

These calculations have been performed with ENC's RODEX2 code.[8.13]

3.5 ROD BOW

Post-irradiation examination of BWR fuel fabricated by ENC has shown that the magnitude of fuel rod bowing is very small. No impact on thermal margins is expected from these small dimensional changes.

3.6 BYPASS FLOW

As shown in Table 3.3, bypass flow for the Dresden 3 end-of-Cycle 8 mixed core loading case is calculated to be 10.4% of the total core flow (98×10^6 lbm/hr). This result is within the usual range of bypass flow for BWR's and represents adequate bypass flow.

3.7 FUEL CLADDING INTEGRITY SAFETY LIMIT

The Fuel Cladding Integrity Safety Limit was calculated using the methodology reported in Reference 8.10. The design basis power distribution histogram shown in Figure 3.4 was used to determine the safety limit. The Monte Carlo procedure used the uncertainties summarized in Table 3.5 to calculate the limit. The detailed analysis is reported in Reference 9.3.

The analysis demonstrates that a Fuel Cladding Integrity Safety Limit of 1.05 provides assurance that during steady state operation at the safety limit, at least 99.9% of the fuel rods in the core would be expected to avoid boiling transition.

Table 3.1 Hydraulic Characterization Comparison Between ENC XN-1 and GE 8x8R

	ENC	GE 8x8R
Lower Tie Plate Loss Coefficient (K_{LTP})	3.60	3.57*
Upper Tie Plate Loss Coefficient	0.378	0.901
Spacer Loss Coefficient	1.734 $R_e^{-.069}$	6.413 $R_e^{-.165}$
Bare Rod Friction Factor	0.208 $R_e^{-.20}$	0.212 $R_e^{-.20}$

---

* This is at Reynolds number of 200,000. More generally, the difference between GE and ENC lower tie plate pressure losses as referenced to the GE 8x8R bare rod flow area is given by:

$$\Delta K_{LTP} = K_{LTP_{GE}} - \frac{A_{BR_{GE}}}{A_{BR_{ENC}}} K_{LTP_{ENC}}$$

$$= 47.59\ R_e^{-.148} - 7.85$$

Table 3.2 Dresden 3 Thermal Hydraulic Design Conditions

Reactor Conditions

Core Power Level (MWt)(100%)	2527
Core Exit Pressure (psia)	1026
Core Inlet Enthalpy (BTU/lbm)	521.8
Total Recirculating Coolant Flow (lbm/hr)	98.0×10^6

Core Loading

	Central Region	Peripheral Region
8x8 (GE), Type 1 (w/o channel seal)	32	36
8x8 (GE), Type 2 (with channel seal)	184	48
8x8R (GE)	200	--
8x8 (ENC)	224	--
Total	640	84

Core Power Distribution

Axial Power Shape	Figure 3.1
Average Bundle Power (MWt)	3.49
Central Region (Average)	3.81
Peripheral Region (Average)	1.09

Table 3.2 Dresden 3 Thermal Hydraulic Design Conditions (Cont.)

<u>Core Power Distribution</u> (Cont.)	
Average Bundle Power by Type (MWt)	
8x8 (GE), Type 1, Central	2.31
8x8 (GE), Type 2, Central	3.19
8x8R (GE)	4.25
8x8 (ENC)	4.13
8x8 (GE), Type 1, Peripheral	1.01
8x8 (GE), Type 2, Peripheral	1.15
Maximum Bundle Power (MWt)	
8x8R (GE)	5.235
8x8 (ENC)	5.235
<u>Fuel Assembly Description</u>	
Hydraulic Resistance Characteristics	Table 3.1
Fuel Rod Diameters (inch)	
8x8 (GE)	.493
8x8R (GE)	[illegible]
8x8 (ENC)	[illegible]

Table 3.2 Dresden 3 Thermal Hydraulic Design Conditions (Cont.)

Fuel Assembly Description (Cont.)	
# Fuel Rods/Assembly	
8x8 (GE)	63
8x8R (GE)	62
8x8 (ENC)	63
# Spacers (all fuel types)	7
Active Fuel Length (feet)	12.1
Total Fuel Rod Length (feet)	13.1

Table 3.3 Thermal Hydraulic Analysis Results
Dresden 3 Cycle 8 - Mixed Core

Core Average Results

Exit Enthalpy (BTU/lbm)	609.8
Exit Quality (active region)	11.1%
Exit Void Fraction	0.583
Axial Average Void Fraction	0.316
Flow Hole Leakage Pressure Drop (psi)	5.6
Bypass Flow	10.4%
Core Pressure Drop (psi)	15.5

Maximum Power Assembly (1.5 Radial Peaking) Results

	ENC 8X8	GE 8X8R
Assembly Flow (lbm/hr)	118.5×10^3	111.6×10^3
Exit Quality (active region)	19.0%	20.4%
Exit Void Fraction	0.742	0.757
Axial Average Void Fraction	0.465	0.482
Critical Power Ratio	1.536	1.486

Table 3.4 Critical Power Ratio Results for Different Core Configuations (Dresden 3, Maximum Power Assemblies with 1.5 Radial Peaking)

		ENC XN-1	GE 8X8R
Case 1	Cycle 8 - Mixed Core	1.536	1.486
Case 2	Modified Cycle 8 GE 8X8R replacing ENC XN-1	--	1.495
Case 3	Cycle 11 - ENC Core	1.509	--

Table 3.5 Uncertainties Considered in the MCPR Safety Limit

Parameter	Standard Deviation*	Reference
Feedwater Flow Rate	0.0176	8.10
Feedwater Temperature	0.0076	8.10
Core Pressure	0.0050	8.10
Total Core Flow Rate	0.0250	8.10
Core Inlet Enthalpy	0.0024	9.2
XN-3 Critical Power Correlation	0.0411	8.9
Assembly Flow Rate	0.0280	8.10
Power Distribution		
Radial Peaking Factor	0.0518	9.2
Local Peaking Factor	0.0246	8.1

*Fraction of nominal value

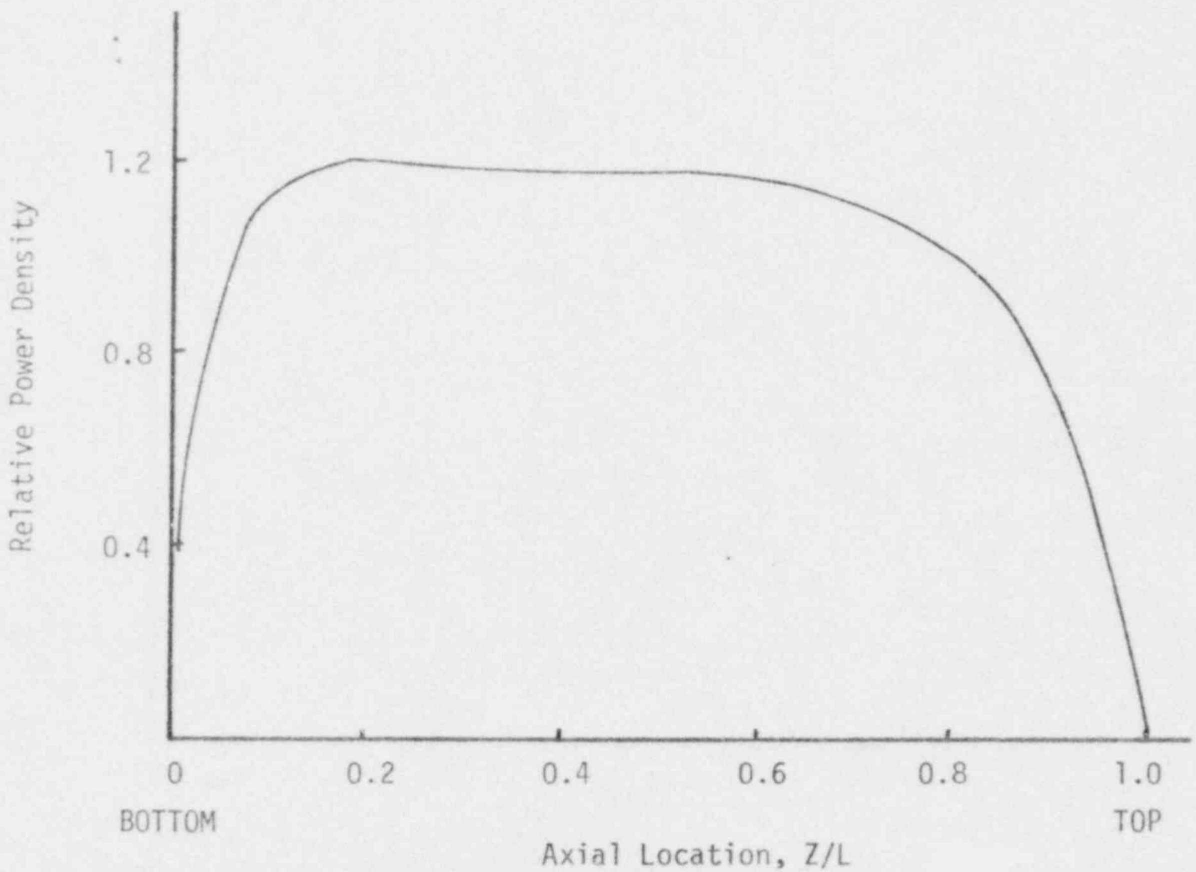

Figure 3.1 Axial Power Profile

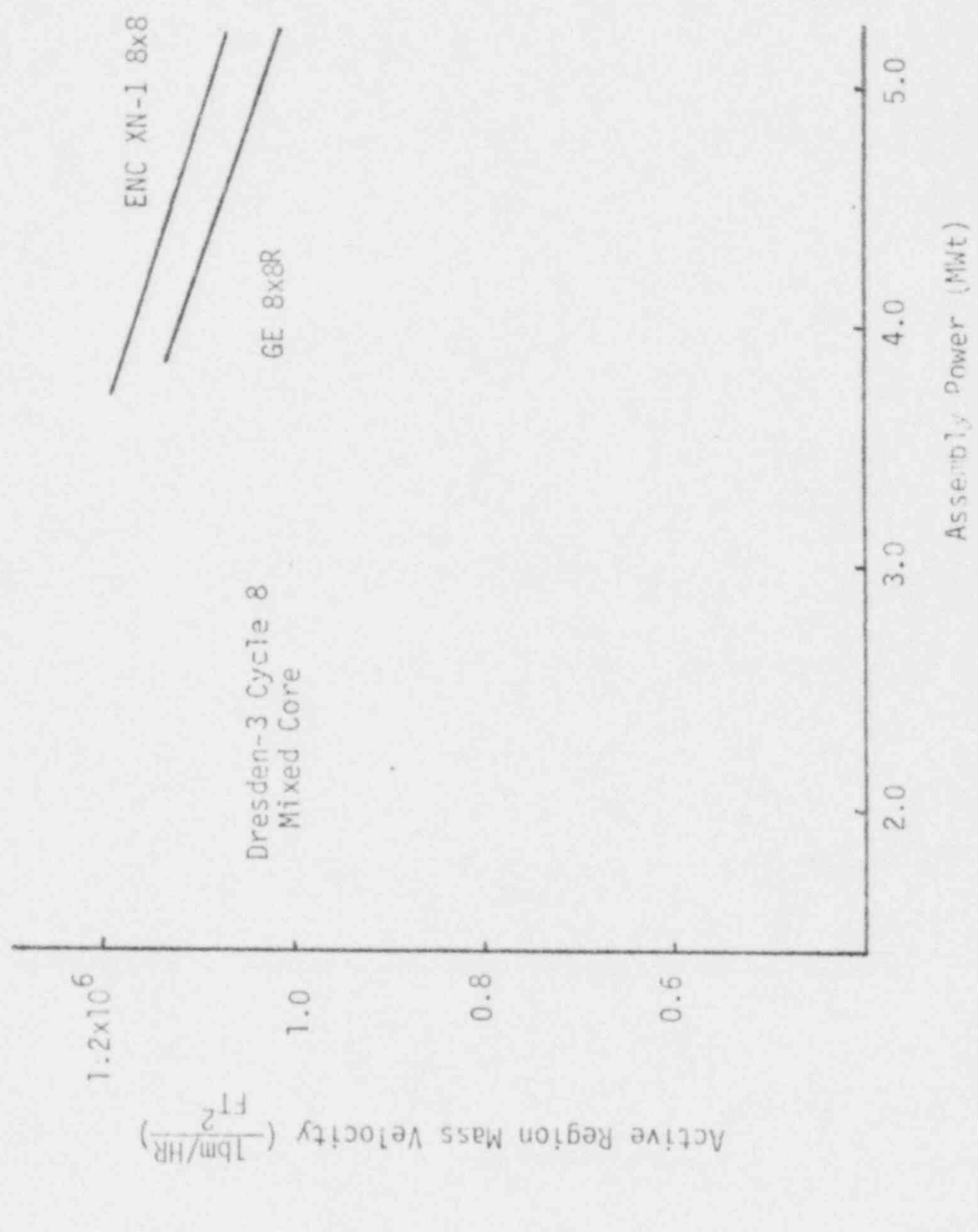

Figure 3.2 Central Orifice Zone Demand Curves

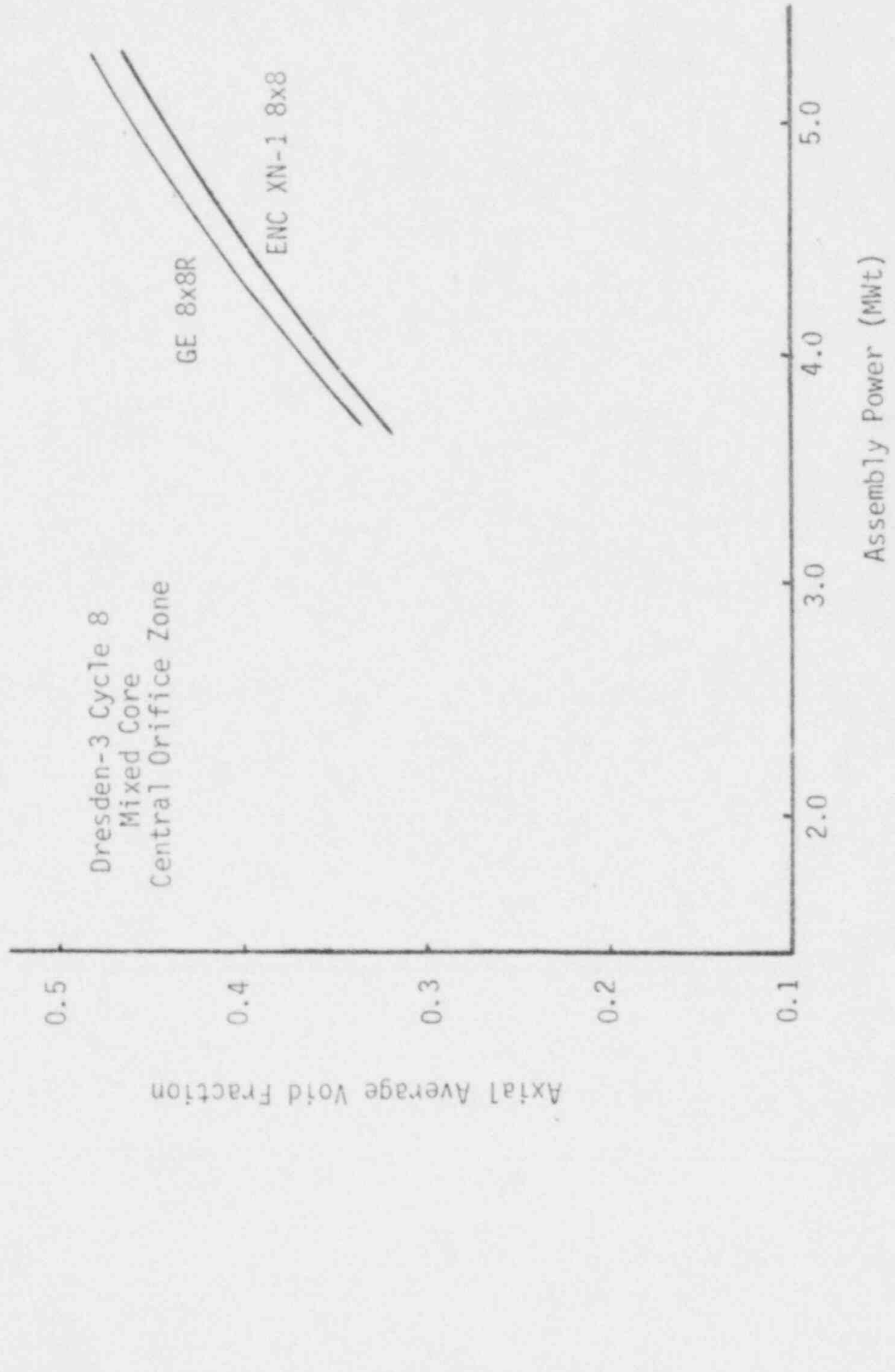

Figure 3.3 Assembly Voids versus Assembly Power

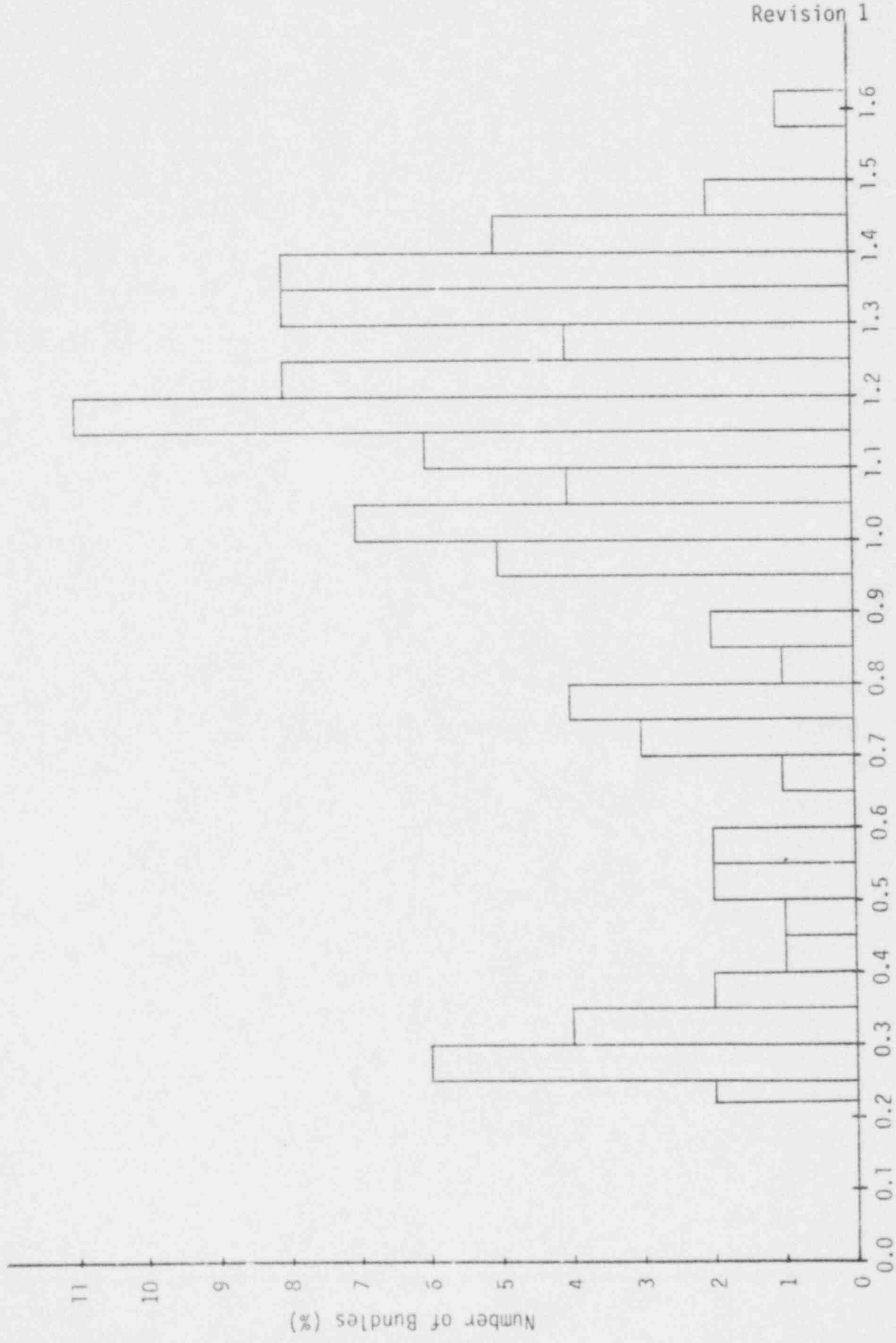

Figure 3.4 Dresden 3 Cycle 8 Safety Limit Radial Power Histogram

4.0 NUCLEAR DESIGN

This section provides a bundle nuclear design evaluation and a core nuclear design report.

4.1 BUNDLE NUCLEAR DESIGN

The results of the neutronic design analysis for ENC's XN-1 8x8 fuel design for Dresden Unit 3 are presented in this section. Analyses were performed using the methodology described in Reference 8.1.

The key nuclear design characteristics for the ENC XN-1 8x8 fuel assembly are summarized below.

- The fuel assembly average enrichment, including a six inch top and bottom natural (0.711 w/o U-235) uranium blanket, is 2.69 w/o U-235. The average enrichment of the central portion of the fuel assembly is 2.87 w/o U-235 and is 133.24 inches in length.
- Five enrichments are utilized to yield a flat local power distribution which results in a balanced design relative to MCPR and MAPLHGR limits.
- The assembly contains five burnable poison rods containing 3.0 w/o Gd_2O_3 blended with 2.25 w/o U-235 to reduce the initial reactivity of the assembly.
- The fuel assembly contains 63 fueled rods and one non-fueled water rod.

4.1.1 Neutronic Design Parameters

The key neutronic design parameters for the ENC Type XN-1 fuel design are presented in Table 4.1.

4.1.2 Enrichment Level and Distribution

The nominal enrichment level (average fissile content) of the enriched lattice of the ENC XN-1 8x8 reload fuel assemblies is 2.87 w/o U-235. The maximum lattice k_∞ in the normal reactor core geometry at peak reactivity is 1.224.

The enrichment distribution of the ENC Type XN-1 reload fuel design was selected on the basis of maintaining a balance between local power peaking factors, assembly reactivity, maximum average planar linear heat generation rate (MAPLHGR), and minimum critical power ratio (MCPR) considerations. The enrichment distribution of the ENC XN-1 8x8 reload design is shown in Figure 4.1.

4.2 CORE NUCLEAR DESIGN

This section provides a description of the core configuration established for Cycle 8 operation of Dresden Uniti 3. Core nuclear design analyses were performed using the methodology reported in Reference 8.1.

4.2.1 Core Configuration

The reference Cycle 8 core loading pattern and fuel assembly inventory are shown in Figure 4.2. All the listed assemblies are irradiated except for those identified as XN-1 8x8, which are unirradiated at the beginning of Cycle 8. No 7x7 fuel remains in the core for Cycle 8.

The nominal end-of-Cycle 7 core average exposure is 21,292 MWD/MTU. The beginning and end of Cycle 8 core average exposures are 12,945 MWD/MTU and 21,127 MWD/MTU respectively. For the Cycle 8 cold shutdown reactivity calculations, the end of Cycle 7 exposure is 20,806 MWD/MTU.

4.2.2 Core Reactivity Characteristics

The calculated BOC8 cold (68°F) core k-effective values at all rods out and all rods in are 1.099 and 0.944 respectively.

The Technical Specifications require the reactor core to be subcritical by 0.25% Δk in the most reactive condition with the strongest control rod stuck out of the core. The most reactive cold shutdown condition for Cycle 8 occurs at BOC. The calculated core k-effective with the strongest rod out is 0.984 resulting in a shutdown margin of 1.6% Δk. The R value for Cycle 8 is 0.04% Δk to account for the effect of B_4C settling in the absorber tubes.

The standby liquid control system is capable of bringing the reactor from full power to a cold shutdown assuming none of the withdrawn control rods can be inserted. With a boron concentration of 600 ppm in the reactor water, the maximum core k_{eff} is 0.944 at cold, xenon free conditions. The calculated shutdown margin (Δk) of the liquid control system is 0.0565 compared to the required Technical Specification minimum value of 0.03.

4.2.3 Control Rod Patterns

Operating control rod patterns are not expected to vary significantly from those typically used in the past.

4.2.4 Core Stability

The stability of the Cycle 8 core was determined analytically. The resultant decay ratios are shown as functions of core power in Figure 4.3. The calculated value of the decay ratio at the intersection of the natural circulation flow line and the 100 percent rod line is 0.45.

Table 4.1 Dresden 3 Reload Batch XN-1
Neutronic Design Values

Fuel Pellet	
Fuel Material	UO_2 Sintered Pellets
Density, g/cc	10.36
% of TD	94.5
Diameter, inches	
3.57 w/o U-235 pellets	0.4055
Others	0.4045
Pellet Axial Height	Ref. 9.1
Dish Volume (total), % of Pellet Volume	
Enriched	1.0
Natural	0.0
Reference Fuel Temperature, °F	
Enriched	915
Natural	740
Fuel Rod	
Fuel Length, inches	145.24
Fuel Stacked Density, g/cc	10.26
Diametral Pellet-to-Clad Gap, inches	
3.57 w/o U-235 fuel	0.0085
Others	0.0095
Cladding Material	Zircaloy-2
Clad I.D., inches	0.414
Clad O.D., inches	0.484
Initial Pressurization	Ref. 9.1

Table 4.1 Dresden 3 Reload Batch XN-1
Neutronic Design Values (cont.)

Fuel Assembly	
Number of Rods, Total	64
Fuel Rods	
Low-Low Enrichment (1.35 w/o)	1
Low Enrichment (1.90 w/o)	4
Medium-Low Enrichment (2.25 w/o)	19
Medium-Low Enrichment with Gd_2O_3 (2.25 w/o + 3.0 w/o Gd_2O_3)	5
Medium Enrichment (3.34 w/o)	16
High Enrichment (3.57 w/o)	18
Inert Water Rod	1
Fuel Rod Pitch, inches	0.641
Fuel Assembly Loading, Kg UO_2	197.3
Fuel Assembly Loading, Kg U	173.9
Core Data	
Number of Fuel Assemblies	724
Rated Thermal Power Level, MW_t	2527
Rated Core Flow, Mlbm/hr	98.0
Core Inlet Subcooling, BTU/lbm	24.6
Moderator Temperature, °F	546
Channel Dimensions	
Thickness, in.	0.080
Internal face-to-face dimension, in.	5.278
Fuel Assembly Cell Dimensions	
Assembly Pitch, in.	6.0
Wide Gap Thickness, in.	0.750
Narrow Gap Thickness, in.	0.374

Table 4.1 Dresden 3 Reload Batch XN-1
Neutronic Design Values (cont.)

Control Rod Data

Total Blade Span, in.	9.750
Total Blade Support Span, in.	1.5630
Blade Thickness, in.	0.3120
Blade Face-to-Face Internal Dimension, in.	0.200
Number of B_4C Rods per Blade	84
B_4C Rod O.D., in.	0.188
B_4C Rod I.D., in.	0.138
Percent of B_4C Theoretical Density	70

L	ML	ML	M	M	M	ML	ML
ML	ML*	M	H	H	ML*	M	ML
ML	M	H	H	H	H	ML*	M
ML	M	H	H	W	H	H	M
ML	M	H	H	H	H	H	M
ML	ML	M	H	H	H	M	ML
L	ML*	ML	M	M	M	ML*	ML
LL	L	ML	ML	ML	ML	ML	L

W I D E (left side)

W I D E

LL --- 1.35 W/O U235
L --- 1.90 W/O U235
ML --- 2.25 W/O U235
M --- 3.34 W/O U235
H --- 3.57 W/O U235
ML* --- 2.25 W/O U235 + 3.00 W/O GD203
W --- INERT WATER ROD

FIGURE 4.1 ENRICHMENT DISTRIBUTION

C1	A2	C1	A2	C1	B2	C1	C1	B2	C1	C1	C1	C1	B4	A4
A2	C1	D0	C1	D0	C1	D0	B2	D0	B2	D0	B2	D0	B2	A4
C1	D0	A2	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	B2	A4
A2	C1	D0	C1	D0	B2	D0	B2	D0	B2	D0	C1	D0	B2	A3
C1	D0	B2	D0	B2	D0	C1	D0	C1	D0	C1	D0	C1	B3	A4
B2	C1	D0	B2	D0	B3	D0	B3	D0	B2	D0	C1	B2	B3	
C1	D0	B2	D0	C1	D0	C1	D0	C1	D0	C1	B3	B3		
C1	B2	D0	B2	D0	B3	D0	B3	D0	C1	D0	B2	A3		
B2	D0	C1	D0	C1	D0	C1	D0	C1	D0	B2	B2	B4		
C1	B2	D0	B2	D0	B2	D0	C1	D0	C1	B2	B4			
C1	D0	C1	D0	C1	D0	C1	D0	B2	B2	A3				
C1	B2	D0	C1	D0	C1	B3	B2	B2	B4					
C1	D0	C1	D0	C1	B2	B3	A3	B4						
B4	B2	B2	B2	B3	B3									
A4	A4	A4	A3	A4										

XY — X = Fuel Type, Y = Cycles Irradiated

Fuel Type	Number of Assemblies	Description
A	72	GE 8x8 2.50 w/o U-235
B	228	GE 8x8 2.62 w/o U-235
C	200	GE P8x8R 2.65 w/o U-235
D	224	XN-1 8x8 2.69 w/o U-235

Figure 4.2 Dresden Unit 3 Cycle 8 Reference Loading Pattern
(One Quarter of Symmetrical Core Loading)

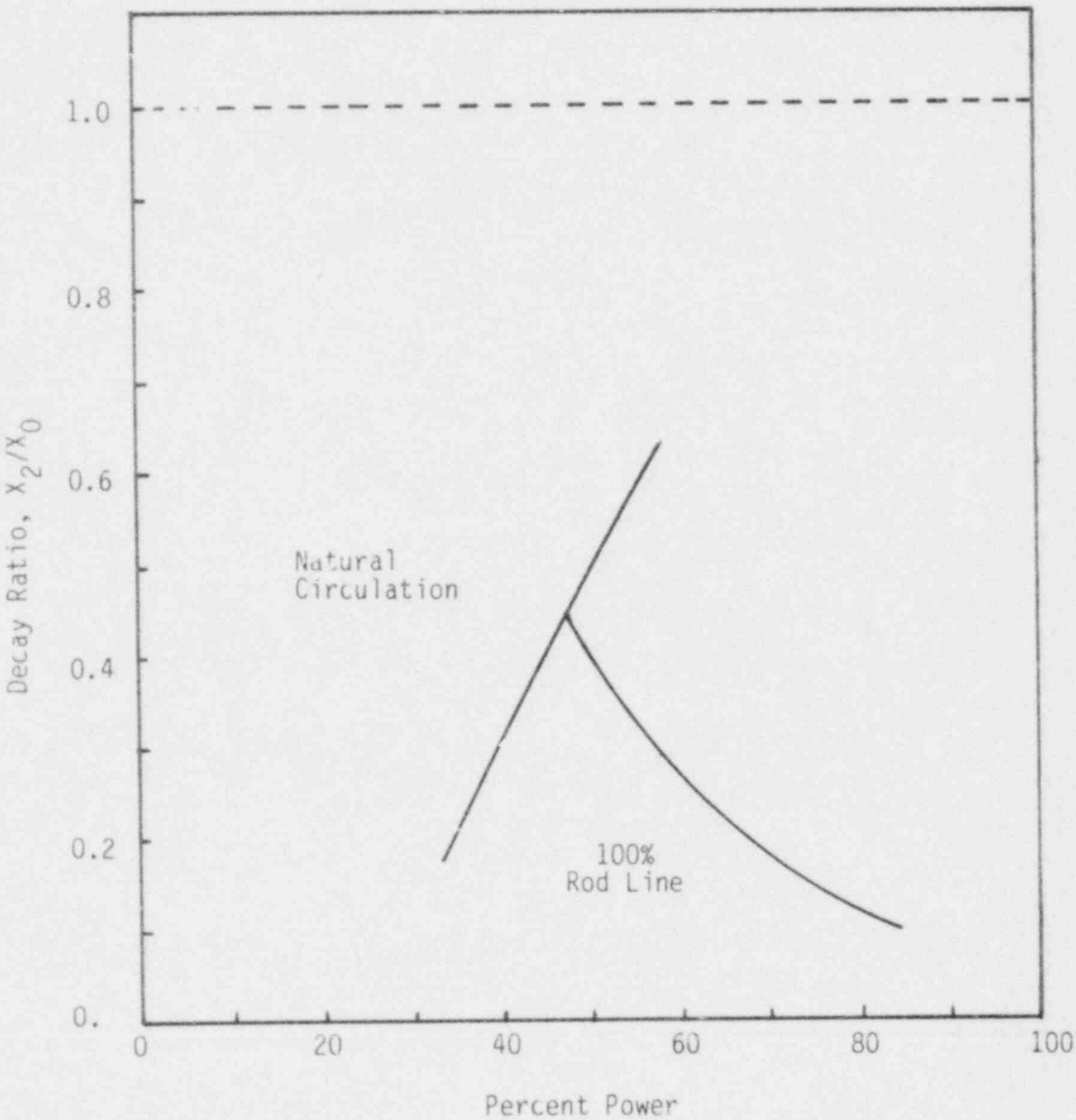

Figure 4.3 Decay Ratio vs. Reactor Power

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Analyses are made to demonstrate the ability of the fuel to perform satisfactorily during infrequent and moderately frequent operational events and to establish appropriate operating limits for the reactor. The generic methodology used for the analysis of these anticipated events has been reported in References 8.1, 8.6, 8.8 and 8.12. The purpose of this section is to report the results of the anticipated operational occurrence analyses performed in support of the operation of Dresden 3 Cycle 8.

5.1 ANALYSES OF PLANT TRANSIENTS AT RATED CONDITIONS

The generator load rejection without bypass transient was determined to be the limiting transient for full power, full flow operation. This determination was made after evaluating a number of transient events for the change in thermal margin associated with them using the input values reported in Table 5.1. The considered transients and the primary results of the analyses are reported in Table 5.2. Uncertainties in the input parameters for these other transients were assumed to be at bounding values. These analyses are reported in detail in Reference 9.3.

The generator load rejection without bypass transient was evaluated to determine thermal margin requirements using the generic statistical methodology described in Reference 8.12. Results of that analysis are reported in Section 5.5.

5.2 ANALYSES FOR REDUCED FLOW OPERATION

The MCPR reduced flow multiplier, k_f, was reevaluated. The combination of the MCPR Operating Limit and the k_f curves as established in Reference 9.5 provides adequate protection of the MCPR Fuel Cladding Integrity Safety Limit during anticipated operational occurrences from all attainable power-flow combinations. Analyses in support of the revised k_f curves are reported in Reference 9.5.

5.3 FUEL LOADING ERROR

The inadvertent loading of a fuel bundle into an incorrect core location and the inadvertent rotation of a fuel bundle 180 degrees from its intended orientation were analyzed using the methodology described in Reference 8.1. The largest calculated ΔCPR for the fuel loading error is 0.16 for both the ENC XN-1 and GE 8x8 fuel types. This ΔCPR was determined by the fuel misloading error analysis; the fuel misorientation resulted in a smaller ΔCPR value.

5.4 CONTROL ROD WITHDRAWAL ERROR

The consequences associated with the inadvertent withdrawal of a high worth control blade until its motion is halted by the rod block was evaluated using the methodology described in Reference 8.1 using the limiting control rod pattern shown in Figure 5.2. The results are reported parametrically with rod block setting in Table 5.3. The rod block monitor setting for Cycle 8 was selected to be 110% as indicated in the table; at this setting, the largest ΔCPR for the rod withdrawal error is 0.15.

5.5 THERMAL MARGIN DETERMINATION

The thermal margin requirements for Cycle 8 operation were determined from the consequences of the generator load rejection without bypass transient using the methodology described in Reference 8.12. The statistical predictor variables selected for the statistical analysis were scram delay time, scram insertion speed, scram reactivity, and void reactivity. Application of the statistical methodology to the parameters of the limiting transient resulted in the ninety-fifth percentile ΔCPR of 0.25; this value was used in determining the MCPR Operating Limit. This methodology was applied to each fuel type resident in the core, and the results are reported by fuel type in Table 5.4. The results for GE 8x8R and P8x8R fuel types were calculated to be identical because of equal values for gap conductance.

The MCPR Operating Limits were determined from the values for the Limiting Transient ΔCPR as noted in Table 5.4 and the Fuel Cladding Integrity Safety Limit as determined in Section 3.7. The operating limit values are reported in Section 7.0. Plant responses to the limiting transient at nominal input conditions are shown in Figures 5.3-5.5.

5.6 ASME OVERPRESSURIZATION ANALYSIS

In accordance with the provisions of the ASME Code, an overpressurization analysis was performed using the COTRANSA plant transient simulation code. The analysis showed that even if failure of the most critical active component were to be assumed (i.e., the scram associated with the closure of the Main Steam Isolation Valve were not to occur and the event were to be terminated by the APRM high flux scram), and if no credit were to be allowed for operation of the four electromatic relief valves, the inadvertent MSIV closure event would not result in reactor vessel pressure exceeding 110% of the design pressure. The maximum vessel pressure observed in the analysis was 1346 psig, which corresponds to 108% of the reactor vessel design pressure. The steam dome pressure associated with the maximum pressure was 1324 psig, which indicates that a pressure safety limit of as high as 1353 psig as measured by the steam space pressure indicator would provide assurance that the 110% of design pressure criterion was not exceeded.

The MSIV closure transient evaluated for ASME code compliance resulted in a higher maximum pressure than that noted for the turbine trip without bypass transient evaluated under the same assumptions.

Table 5.1 Significant Input Parameters to the Transient Analyses

Reactor Thermal Power (MWt)	2527
Steam Dome Pressure (psia)	1020
Feedwater Enthalpy (BTU/lbm)	304.1
Scram Reactivity	Fig. 5.1
90% Scram Insertion Time (sec)	3.5
Nominal Void Reactivity Coefficient (\$/VF)	-15.89
Nominal Doppler Reactivity Coefficient (\$/^oF)	-.0026
Core Average Gap Conductance (BTU/hr-ft^2-^oF)	893
Limiting Assembly Gap Conductance (BTU/hr-ft^2-^oF)	1430

Table 5.2 Analytical Results for Plant Transients at Rated Conditions[1]

Transient	Analytical Model	Exposure	Power	Flow	Maximum Heat Flux	Maximum Neutron Flux	Maximum Pressure[2]	ΔCPR
Generator Load Rejection Without Bypass[4]	COTRANSA	EOC8	100%	100%	114.3%	372%	1294 psia	0.25[3]
Loss of 145°F Feedwater Heating	PTSBWR3	EOC8	100%	100%	119.2%	120%	1056 psia	0.16
Feedwater Controller Failure	COTRANSA	EOC8	100%	100%	116.8%	293%	1214 psia	0.21

NOTES:

1. Values associated with bounding uncertainty values; applicable to all fuel types present in core
2. Maximum transient pressure calculated for lower plenum
3. Statistically determined change in MCPR; applicable to all fuel types present in core
4. Parameters associated with nominal uncertainty values reported for Generator Load Rejection Without Bypass

Table 5.3 Control Rod Withdrawal Error Analyses

Rod Block Reading, %	Withdrawn Rod Position, Feet	ΔCPR GE 8x8	ΔCPR ENC 8x8
105	3.5	0.09	0.09
106	4.0	0.10	0.11
107	4.5	0.12	0.13
108	4.5	0.12	0.13
109	5.0	0.14	0.15
110*	5.0	0.14	0.15

*Selected rod block monitor setting for Cycle 8

Table 5.4 Limiting Transient ΔCPR for Resident Fuel Types

Transient	ΔCPR		
	8x8(XN-1)	8x8R(GE)*	8x8(GE)
Generator Load Rejection (w/o bypass)	.25	.25	.25
Increase in Feedwater Flow	.21	.21	.21
Loss of Feedwater Heating	.16	.16	.16

*Typical value for 8x8R and P8x8R fuel types.

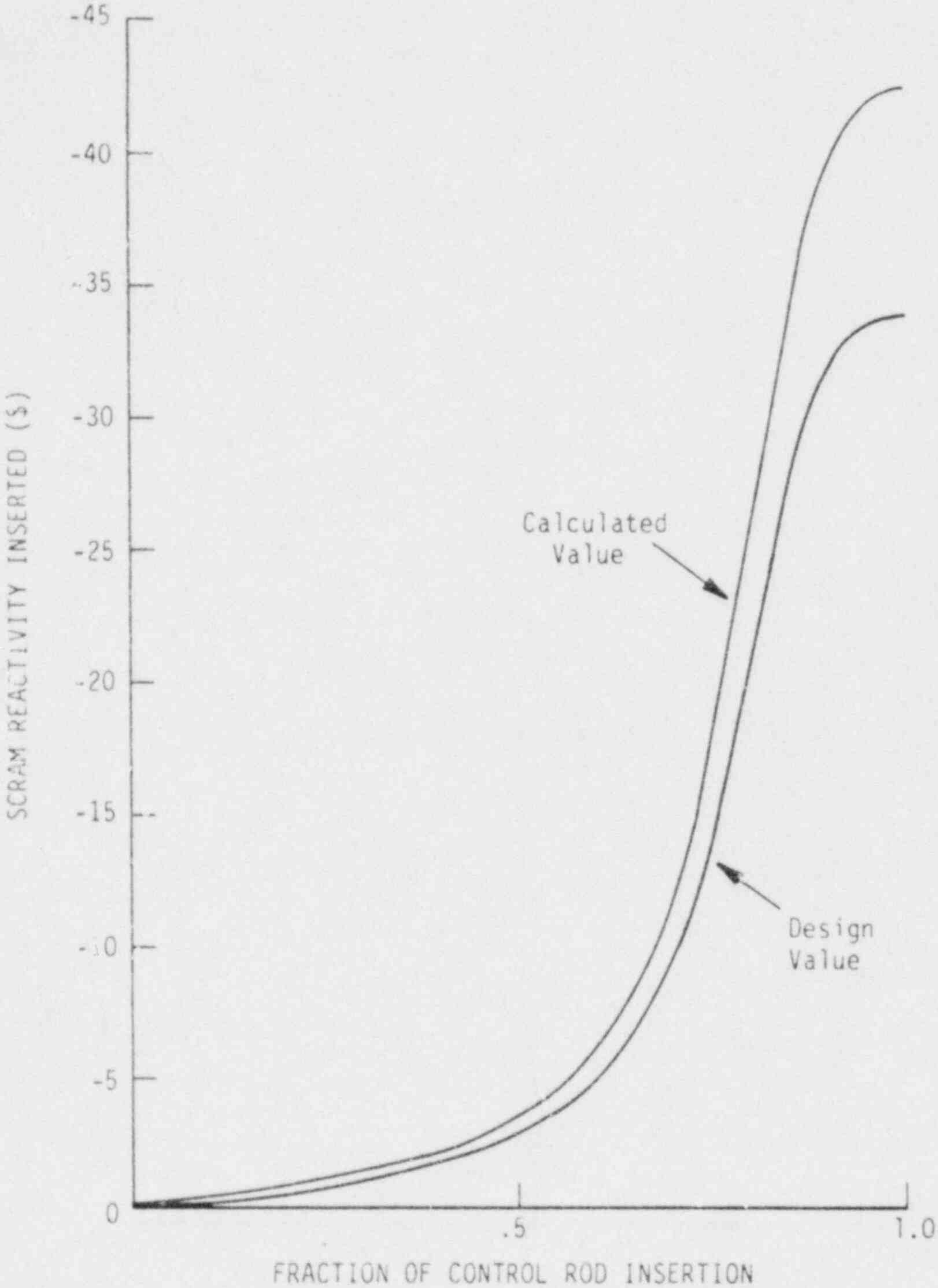

Figure 5.1 Scram Reactivity Used in the PTSBWR3 Analyses

	30	34	38	42	46	50	54	58
31		30		40				
27	24		6		4		24	
23						36		
19			0*		10			
15				38		24		
11	30		24		30			
07		34						
03								

Note: *Control Rod Being Withdrawn, Rod Positions in Notches, Full In = 0, Full Out = Blank or 48

Figure 5.2 Starting Control Rod Pattern for Control Rod Withdrawal Analysis

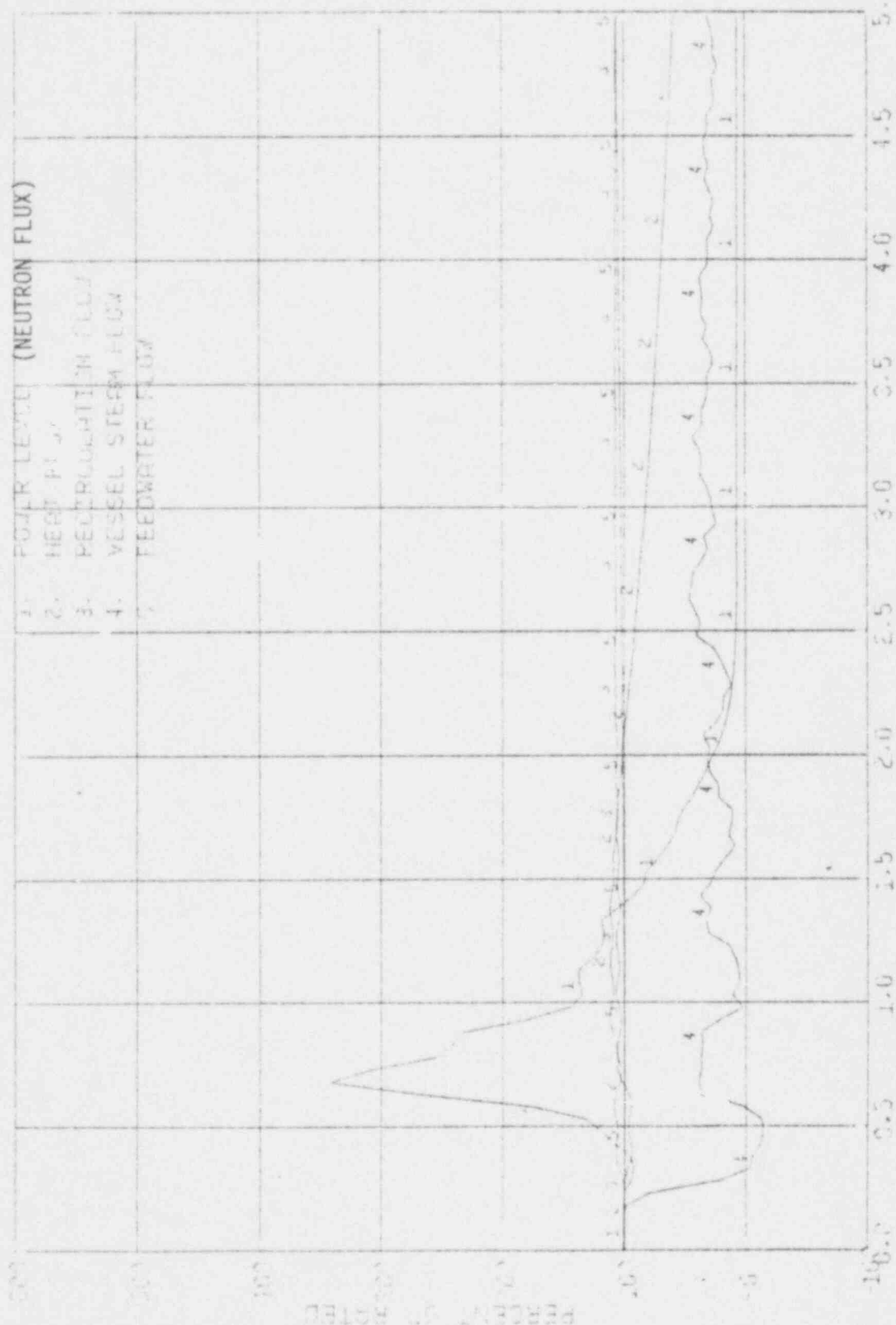

Figure 5.3 Generator Load Rejection w/o Bypass

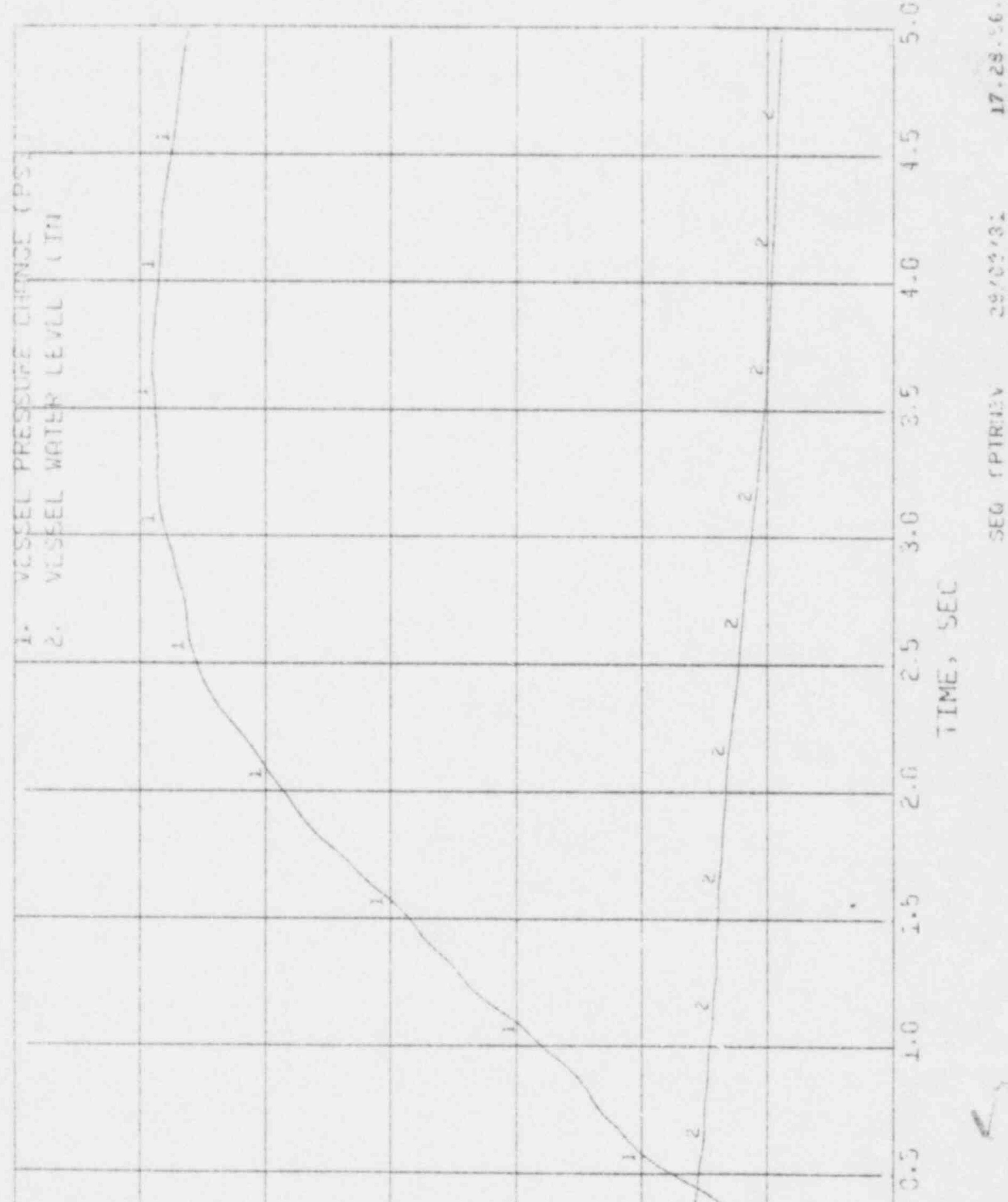

Figure 5.4 Generator Load Rejection w/o Bypass

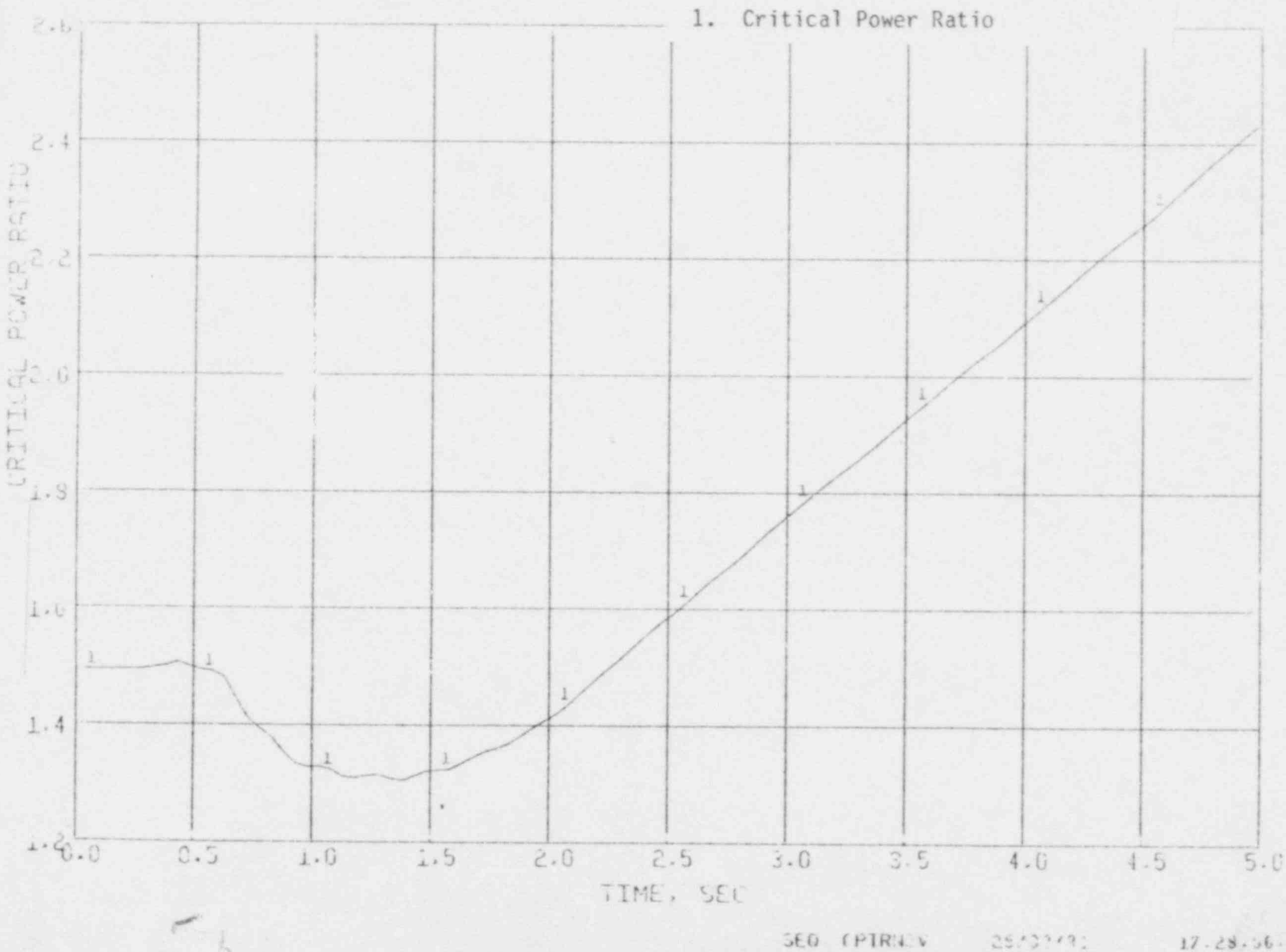

Figure 5.5 Generator Load Rejection w/o Bypass

6.0 POSTULATED ACCIDENTS

This section describes the analyses that were undertaken to assure the capability of the reload core to withstand the effects of postulated accidents.

6.1 LOSS OF COOLANT ACCIDENT ANALYSES

This subsection describes the analyses performed in accordance with 10CFR50.46 and Appendix K to 10CFR50.

6.1.1 Break Spectrum Analysis

A spectrum of potential LOCA break locations and sizes was reported in Reference 8.14. The conclusion in that document was that the limiting break was the double-ended guillotine recirculation line break at the suction to the recirculation pump. That potential break was selected for further analysis as the limiting break.

6.1.2 Limiting Break Analysis

The selected limiting break was analyzed using the methodology reported in References 8.2, 8.3, 8.4, 8.5 and 8.7. The fuel rod stored energy was evaluated with the methodology reported in Reference 8.13. The detailed ECCS analysis was reported in Reference 9.4.

The results of the ECCS analysis are reported in summary form in Table 6.1, which contains values for MAPLHGR, peak cladding temperature and peak local oxidation for several representative exposure points in the life of the fuel. These results are used to define the MAPLHGR operating limits for fuel type XN-1 8x8 in Section 7.0 and are valid for fuel type XN-1 8x8 within the specified exposure limits so long as it remains in the Dresden Unit 3 core.

6.2 CONTROL ROD DROP ACCIDENT

The control rod drop accident was performed on a generic basis in Reference 8.1. The pertinent parametric values and the resultant deposited enthalpy are reported in Table 6.2. The calculated deposited enthalpy of 151 calories/gram is well within the allowable 280 calories/gram specified in the Dresden Unit 3 Technical Specifications.

Table 6.1 Summary of Results of ECCS Analysis

Bundle Average Exposure (MWD/MT)	MAPLHGR (kw/ft)	Peak Cladding Temperature (°F)	Peak Local Metal-Water Reaction (%)
0	13.0	1879	0.8
10,000	13.0	1942	1.0
15,000	13.0	2089	2.8
18,000	12.85	2134	4.1
20,000	12.6	2156	4.1
25,000	11.95	2088	3.4
30,000	11.2	1937	2.1
35,000	10.45	1819	1.4

Table 6.2 Rod Drop Accident Analysis

Dropped Control Rod Worth	= 11.2 mk
Doppler Coefficient (773 ^{o}F)	= -10.2×10^{-6} $\frac{1}{k} \frac{\Delta k}{\Delta T(^{o}F)}$
$\beta_{effective}$	= 0.0058
Four Bundle Local Peaking ($P4B_L$)	= 1.129
Maximum Deposited Fuel Rod Enthalpy	= 151 calories/gram

7.0 OPERATING LIMITS

This section summarizes the results of the analyses reported in the earlier sections of this report by collecting the proper operating limits as indicated by the analyses.

7.1 LIMITING SAFETY SYSTEM SETTINGS

7.1.1 Fuel Cladding Integrity Safety Limit (Specification 1.1.A)

Operation with a Minimum Critical Power Ratio of less than 1.05 shall constitute violation of the Fuel Cladding Integrity Safety Limit.

7.1.2 Steam Dome Pressure Safety Limit (Specification 1.2)

The reactor coolant system pressure shall not exceed 1353 psig at any time when irradiated fuel is present in the reactor vessel.

7.2 LIMITING CONDITIONS FOR OPERATION

7.2.1 Average Planar LHGR (Specification 3.5.I)

During steady state power operation, the Average Planar Linear Heat Generation Rate (APLHGR) of all the rods in any fuel assembly of type ENC XN-1 8x8 at any axial location shall not exceed the value indicated by the exposure-dependent function shown in Figure 7.1.

APLHGR limitations for other fuel types resident in the core are not changed by this report.

7.2.2 Minimum Critical Power Ratio (Specification 3.5.K)

During steady state operation, MCPR shall be greater than or equal to the values given in Table 7.1 times the appropriate value of k_f.

Table 7.1 MCPR Operating Limits

Fuel Type	MCPR Operating Limit
XN-1 8x8	1.30
8x8R/P8x8R	1.30
8x8	1.30

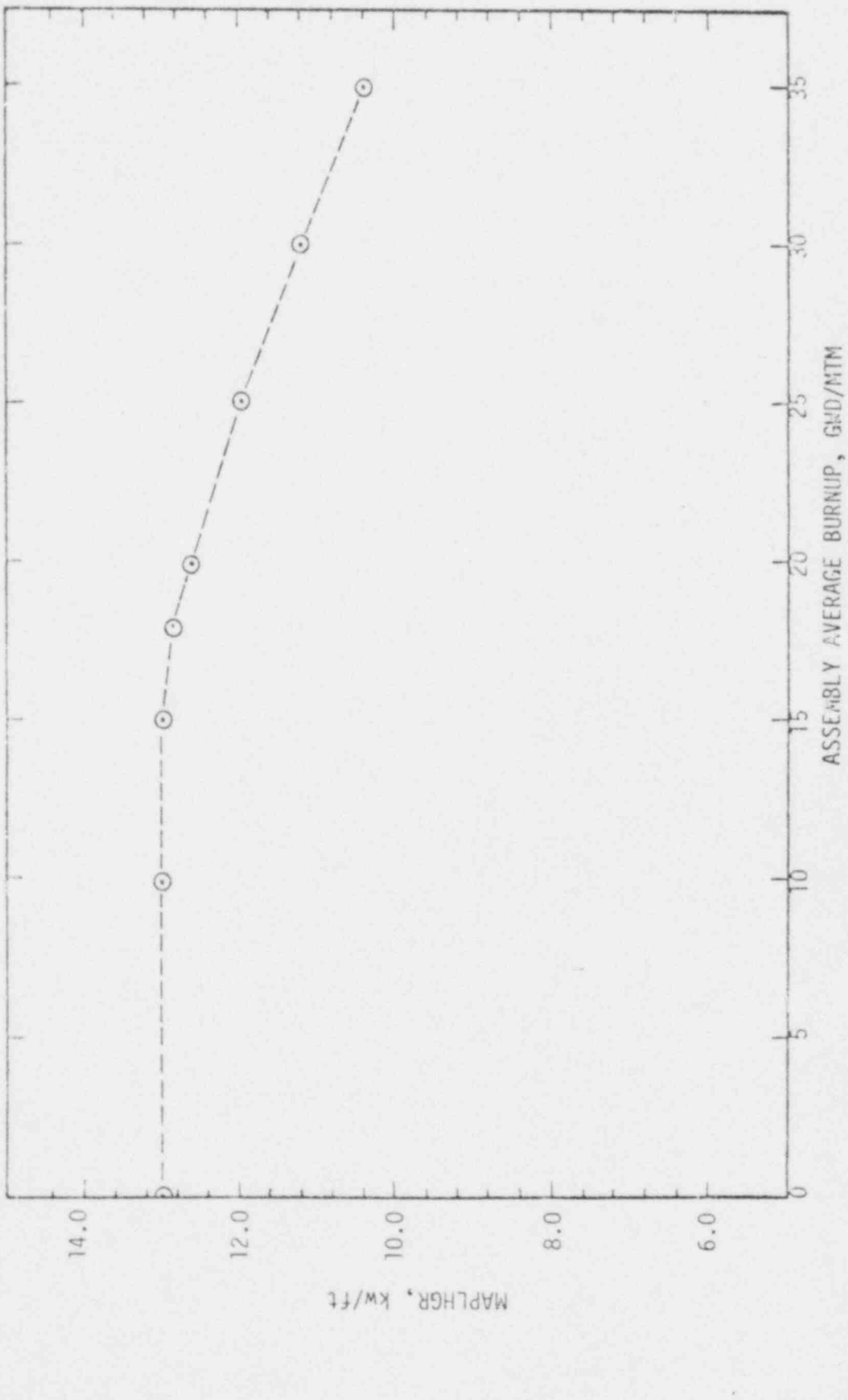

Figure 7.1 Dresden Unit 3 MAPLHGR vs. Assembly Average Burnup for Fuel Type XN-1 8x8

8.0 REFERENCES FOR EXXON NUCLEAR METHODOLOGY FOR BOILING WATER REACTORS

The following reports describe the ENC methodology for the analysis of jet-pump boiling water reactors. They are incorporated in this submittal by reference.

8.1 XN-NF-80-19(P), Volume 1 (Supplements 1 and 2), May 1980
Exxon Nuclear Methodology for Boiling Water Reactors
Neutronics Methods for Design and Analysis

8.2 XN-NF-80-19(P), Volume 2, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
EXEM: ECCS Evaluation Model, Summary Description

8.3 XN-NF-80-19(P), Volume 2A, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
RELAX: A RELAP4 Based Computer Code for Calculating Blowdown Phenomena

8.4 XN-NF-80-19(P), Volume 2B, Revision 1, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
FLEX: A Computer Code for Jet Pump BWR Refill and Reflood Analysis

8.5 XN-NF-80-19(P), Volume 2C, June 1981
Exxon Nuclear Methodology for Boiling Water Reactors
Verification and Qualification of EXEM

8.6 XN-NF-80-19(P), Volume 3, Revision 1, April 1981
Exxon Nuclear Methodology for Boiling Water Reactors
THERMEX: Thermal Limits Methodology, Summary Description

8.7 XN-CC-33(A), Revision 1, November 1975
HUXY: A Generalized Multirod Heatup Code with 10CFR50 Appendix K Heatup Option

8.8 XN-NF-79-71(P), Revision 2, November 1981
Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors

8.9 XN-NF-512(P), Revision 1, March 1981
The XN-3 Critical Power Correlation

8.10 XN-NF-524(P), November 1979
Exxon Nuclear Critical Power Methodology for Boiling Water Reactors

8.11 XN-NF-79-59(P), October 1979
Methodology for Calculation of Pressure Drop in BWR Fuel Assemblies

8.12 XN-NF-81-22(P), September 1981
Generic Statistical Uncertainty Analysis Methodology

8.13 XN-NF-81-58(P), August 1981
RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model

8.14 XN-NF-81-71(P), October 1981
Generic Jet-Pump BWR3 LOCA Analysis Using the ENC EXEM Evaluation Model

9.0 ADDITIONAL REFERENCES

Although not specifically identified as part of the Exxon Nuclear Methodology for Boiling Water Reactors listed in Section 8.0, the following documents provide background information.

9.1 S. F. Gaines, "Generic Design Report - Mechanical Design for Exxon Nuclear Jet Pump BWR Fuel Assemblies," XN-NF-81-21(P), November 1981.

9.2 T. W. Patten, "XCOBRA Code User's Manual," XN-NF-CC-43(P), Revision 1, January 1980.

9.3 R. H. Kelley, "Dresden 3 Cycle 8 Plant Transient Analysis Report," XN-NF-81-78, Revision 1, December 1981.

9.4 J. E. Krajicek, "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model--MAPLHGR Results," XN-NF-81-75(P), October 1981.

9.5 R. H. Kelley, "Dresden Unit 3 Analyses for Reduced Flow Operation," XN-NF-81-84(P), November 1981.

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DRESDEN UNIT 3 CYCLE 8 RELOAD ANALYSIS

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