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DRESDEN UNIT 3 CYCLE 9 RELOAD ANALYSIS

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EXXON NUCLEAR COMPANY, Inc.

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DRESDEN UNIT 3 CYCLE 9 RELOAD ANALYSIS

Mechanical, Thermal Hydraulic, and Nuclear Design Analyses

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1.0 INTRODUCTION

This report presents the results of analyses performed by Exxon Nuclear Company (ENC) in support of the Cycle 9 (XN-2) reload for Dresden Unit 3, which is scheduled to commence operation near the end of 1983.

The Cycle 9 core will comprise 184 unirradiated Type XN-2 reload fuel assemblies fabricated by ENC, 224 once-irradiated Type XN-1 fuel assemblies, and 316 General Electric 8x8 assemblies. The ENC-fabricated assemblies are as described in XN-NF-81-21 (Reference 9.1). The core configuration is described in Section 4.0 of this report.

Cycle 9 operation of Dresden Unit 3 will include the use of eight developmental control blades fabricated by ASEA-ATOM. These blades are described in Appendix B.

This report is intended to be used in conjunction with XN-NF-80-19, Volume 4, "Application of the ENC Methodology to BWR Reloads," which describes the analyses which were performed in generation of the results reported in this document.

2.0 FUEL MECHANICAL DESIGN ANALYSIS

Applicable Fuel Design Report — Reference 9.1

The power history depicted in Figure 5.10 of Reference 9.1 bounds the expected power history of the Dresden 3 Type XN-2 fuel.

The XN-2 fuel design is as described in Reference 9.1 except for the pellet axial height in most of the fuel rods, which is slightly shorter than the generic design for improved resistance to pellet-cladding interaction.

Fuel Centerline Temperature	
Exposure at Minimum Margin Point	21,200 MWD/MT
Centerline Temperature at 120% Overpower	4607°F
Melting Point of Fuel	4900°F
Margin to Centerline Melting	293°F

3.0 THERMAL HYDRAULIC DESIGN ANALYSIS

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Core Exposure at BOC9, MWD/MT	14,082
Core Exposure at EOC9, MWD/MT	21,000
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BOC9 Cold K-effective, All Rods In	.949
BOC9 Cold K-effective, Strongest Rod Out	.985
Technical Specification R-Value	.0004(2)
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(1) Nominal Value/Value Used in Shutdown Reactivity Calculations.
(2) Accounts for B_4C Settling in Control Rod Tubes (Maximum K-effective with strongest rod withdrawn occurs at BOC9).

5.0 ANTICIPATED OPERATIONAL OCCURRENCES

Applicable Generic Transient Analysis Report — Reference 9.2

5.1 Analysis of Plant Transients at Rated Conditions — Reference 9.3

Limiting Transients:(1)

Generator Load Rejection Without Bypass (LRWB)

Loss of Feedwater Heating (LFWH)

Feedwater Controller Failure - Maximum Demand (FWCF)

5.2 Analyses for Reduced Flow Operation — Reference 9.4

Limiting Transient: Recirculation Flow Increase

5.3 ASME Overpressurization Analysis — Reference 9.3

Event	MSIV Closure
Single Failure	MSIV Position Scram Trip
Maximum Pressure	1347.6 psig
Maximum Sensed Pressure	1323.1 psig

5.4 Control Rod Withdrawal Error (CRWE)

Starting Control Rod Pattern for Analysis — Figure 5.1

Rod Block Setting	Distance Withdrawn	ΔCPR
106	4.5 ft.	.11
107	4.5	.11
108	5.5	.14
109	5.5	.14
110(2)	6.5	.16

5.5 Fuel Loading Error

ΔCPR — 0.19

(1) Results of Limiting Transient Evaluations reported in Section 5.6.
(2) Rod Block setting of 110% selected for Cycle 9 operation.

5.6 Determination of Thermal Margins — Table 5.1

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Operating Limit
ENC 8x8	1.30
GE 8x8, 8x8R	1.30

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Limiting Break: Double-Ended Guillotine Break
Recirculation Pump Suction Line
1.0 Break Coefficient

Assembly Average Burnup (MWD/MT)	MAPLHGR (kW/ft)	Peak Local MWR (%)	Peak Clad Temperature (^{o}F)
0.	13.0	.8	1879
10,000	13.0	1.0	1942
15,000	13.0	1.7	2123
18,000	12.85	1.9	2159
20,000	12.6	1.5	2074
25,000	11.95	1.2	2011
30,000	11.2	4.5	2153
35,000	10.45	1.4	1808

6.2 CONTROL ROD DROP ACCIDENT(1)	See XN-NF-80-19, Vol. 1
Dropped Control Rod Worth	0.0062
Doppler Coefficient (773°F)	-10.18×10^{-6} 1/K $\Delta K/\Delta T$, $(^{o}F)^{-1}$
Effective Delayed Neutron Fraction	0.0055
Four Bundle Local Peaking Factor	1.232
Maximum Deposited Fuel Rod Enthalpy	85

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Bundle Average Exposure (MWD/MT)	MAPLHGR (kW/ft)
0	13.0
10,000	13.0
15,000	13.0
18,000	12.85
20,000	12.6
25,000	11.95
30,000	11.2
35,000	10.45

(1) Reported for limiting G.E. control blade. See Appendix B for evaluation of ASEA-ATOM development control blade.

7.2.2 Minimum Critical Power Ratio

Fuel Type	MCPR
ENC XN-1, XN-2	1.30
GE 8x8, 8x8R	1.30

Reduced Flow MCPR Limits

Automatic Flow Control	Figure 5.3a
All Conditions	Figure 5.3b

7.2.3 Surveillance Requirements

See Appendix A.

9.0 ADDITIONAL REFERENCES

9.1 S. F. Gaines, "Generic Mechanical Design for Exxon Nuclear Jet Pump BWR Reload Fuel," XN-NF-81-21(A), Revision 1 (January 1982).

9.2 R. H. Kelley, "Exxon Nuclear Plant Transient Methodology for Boiling Water Reactors," XN-NF-79-71, Revision 2 (November 1981).

9.3 R. H. Kelley, "Dresden Unit 3 Cycle 9 Plant Transient Analysis," XN-NF-83-58 (July 1983).

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

9.5 J. E. Krajicek, "Generic Jet Pump BWR/3 LOCA Analysis Using the ENC EXEM Evaluation Model," XN-NF-81-71(A) (October 1981).

9.6 J. E. Krajicek, "Dresden Unit 3 LOCA Analysis Using the ENC EXEM Evaluation Model; MAPLHGR Results," XN-NF-81-75(P) (November 1981) and Supplement 1 (July 1983).

9.7 J. C. Chandler, "Dresden Unit 3 Cycle 8 Reload Analysis," XN-NF-81-76, Revision 1 (December 1981).

Table 4.1 Dresden 3 Reload Batch XN-2
Neutronic Design Values

Fuel Pellet	Reference 9.1
Fuel Rod	Reference 9.1
Fuel Assembly	Reference 9.1
Fuel Assembly Loading, KgUO_2	196.5
Fuel Assembly Loading, KgU	173.2
Core Data	
Number of fuel assemblies	724
Rated thermal power, MW	2527
Rated core flow, 10^6 lbm/hr	98.0
Core inlet subcooling, BTU/lbm	24.6
Moderator temperature, ^oF	546
Channel thickness, inch	0.080
Channel inside face-to-face dimension, inch	5.278
Fuel assembly pitch, inch	6.0
Wide water gap thickness, inch	0.750
Narrow water gap thickness, inch	0.374
Control Rod Data(1)	
Absorber material	B_4C
Total blade span, inch	9.750
Total central support span, inch	1.562
Blade thickness, inch	0.3120

(1) Applies to G.E. fabricated control blades. Refer to Appendix B for description of developmental contorl blades fabricated by ASEA-ATOM.

Table 4.1 Design 3 Reload Batch XN-2
Neutronic Design Values (Cont.)

Blade face-to-face internal dimension, inch	0.200
Absorber rods per blade	84
Absorber rod outside diameter, inch	0.188
Absorber rod inside diameter, inch	0.138
Absorber density, % of theoretical	70

Table 5.1 Determination of Thermal Margins

Event	Model	Exposure	Power	Flow	Maximum Heat Flux	Maximum Power	Maximum Pressure	Indicated MCPR Limit[2]
LRWB	COTRANSA	EOC9	100%	100%	112.5%[3]	300%[3]	1273[3] psig	1.30[4]
FWCF	COTRANSA	EOC9	100%	100%	115.9%	260%	1196	1.26
LFWH	PTSBWR3	EOC9	100%	100%	118.5%	120%	1039	1.21
CRWE[1]	XTGBWR	BOC9	100%	100%	-	-	-	1.21

(1) Rod Block setting of 110% selected for Cycle 9 operation.

(2) Indicated limits applicable to both ENC 8x8 fuel and G.E. 8x8 fuel.

(3) Nominal case results; all other events bounding case results.

(4) Statistically determined value; all other events bounding value.

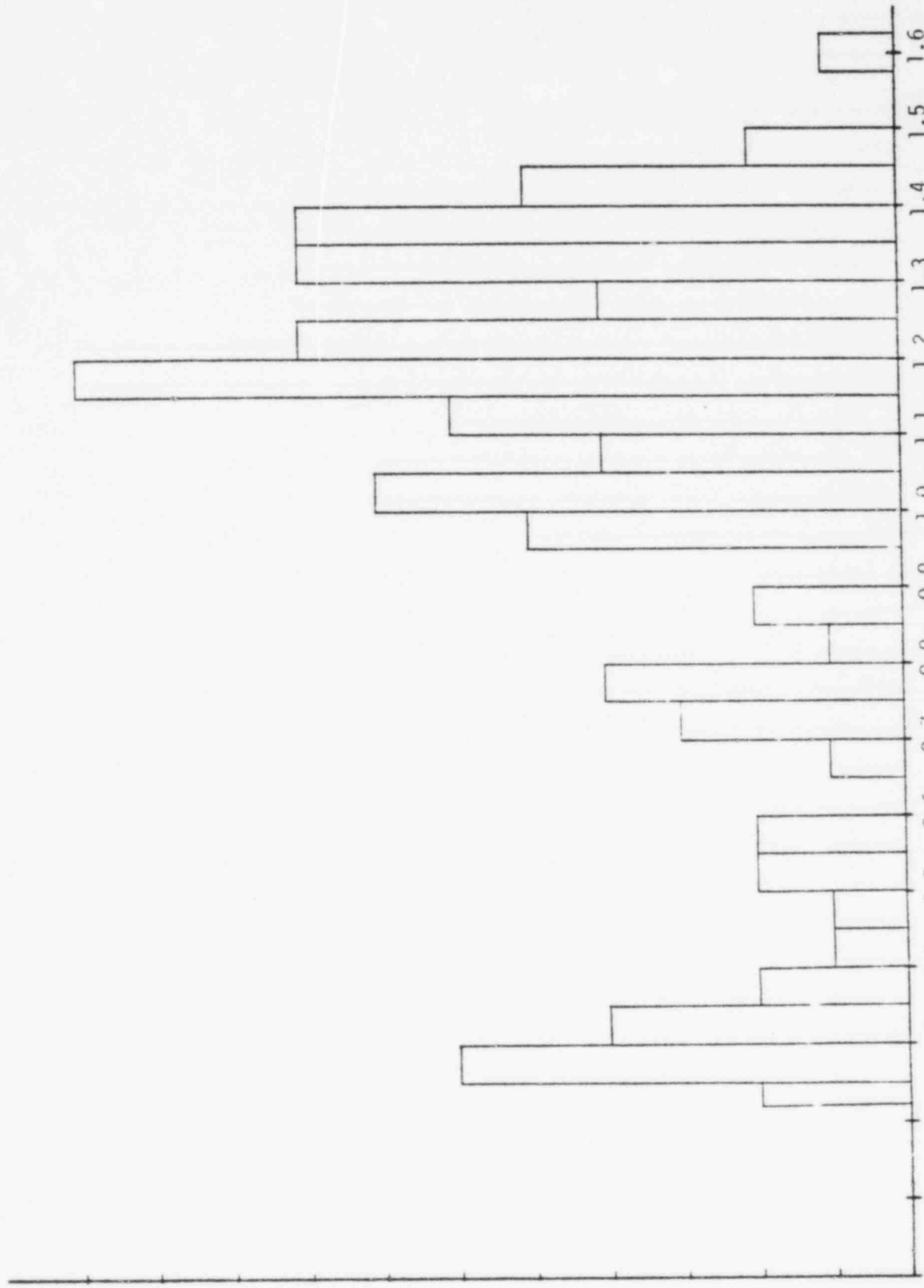


Figure 3.1 Dresden-3 Cycle 9 Safety Limit Radial Power Histogram

(1) L 1.06	ML 1.02	ML 0.92	M 1.08	M 1.06	M 1.08	ML 0.92	ML 1.01
ML 1.08	ML* 0.90	M 1.03	H 1.00	H 0.98	ML* 0.73	M 1.03	ML 0.92
ML 1.03	M 1.09	H 1.00	H 0.94	H 0.95	H 0.96	ML* 0.73	M 1.08
ML 1.01	M 1.06	H 0.97	H 0.94	W 0.00	H 0.95	H 0.98	M 1.06
ML 1.03	M 1.08	H 0.99	H 0.94	H 0.94	H 0.94	H 1.00	M 1.08
ML 1.08	ML 0.93	M 1.03	H 0.99	H 0.97	H 1.00	M 1.03	ML 0.92
L 1.07	ML* 1.01	ML 0.93	M 1.08	M 1.06	M 1.09	ML* 0.90	ML 1.02
LL 1.01	L 1.07	ML 1.08	ML 1.03	ML 1.01	ML 1.03	ML 1.08	L 1.06

W I D E (left side)

W I D E

Figure 3.2 Dresden-3 Cycle 9 Safety Limit Local Peaking

Note 1: Enrichment distribution is explained in Figure 4.1.

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

W I D E

LL --- 1.35 W/O U235
L --- 2.00 W/O U235
ML --- 2.37 W/O U235
M --- 3.51 W/O U235
H --- 3.76 W/O U235
ML* --- 2.37 W/O U235 + 3.50 W/O GD2O3
W --- INERT WATER ROD

FIGURE 4.1 ENRICHMENT DISTRIBUTION FOR FUEL TYPE XN-2 8x8
(Enriched Lattice 3.02 w/o U-235)

C2	G0	F1	C2	C2	G0	F1	C2	C2	G0	F1	C2	C2	F1	E3
G0	C2	G0	F1	F1	F1	G0	F1	F1	F1	G0	F1	G0	D3	E3
F1	G0	F1	G0	C2	G0	C2	G0	C2	G0	F1	G0	F1	F1	E3
C2	F1	G0	C2	C2	F1	F1	C2	C2	F1	G0	C2	G0	C2	A3
C2	F1	C2	C2	C2	G0	C2	C2	C2	G0	F1	G0	F1	E3	E3
G0	F1	G0	F1	G0	F1	G0	F1	G0	F1	G0	C2	E3	A3	
F1	G0	C2	F1	C2	G0	C2	G0	F1	G0	F1	F1	E3		
C2	F1	G0	C2	C2	F1	G0	C2	C2	F1	F1	F1	A3		
C2	F1	C2	C2	C2	G0	F1	C2	C2	G0	C2	C2	E3		
G0	F1	G0	F1	G0	F1	G0	F1	G0	C2	E3	B3			
F1	G0	F1	G0	F1	G0	F1	F1	C2	E3	B3				
C2	F1	G0	C2	G0	C2	F1	F1	C2	B3					
C2	G0	F1	G0	F1	E3	E3	A3	E3						
F1	D3	F1	C2	E3	A3									
E3	E3	E3	A3	E3										

X = Fuel Type

Y = Cycles Irradiated

Fuel Type	Number of Assemblies	Description
A	24	GE 8x8 2.50 w/o U-235 (reinserted assemblies)
B	12	GE 8x8 2.62 w/o U-235 (reinserted assemblies)
C	200	GE 8x8R 2.65 w/o U-235
D	8	GE 8x8 2.50 w/o U-235
E	72	GE 8x8 2.62 w/o U-235
F	224	XN-1 8x8 2.69 w/o U-235
G	184	XN-2 8x8 2.83 w/o U-235

Figure 4.2 Dresden Unit 3 Cycle 9 Reference Loading Pattern By Fuel Type (One Quarter of Symmetrical Core Loading)

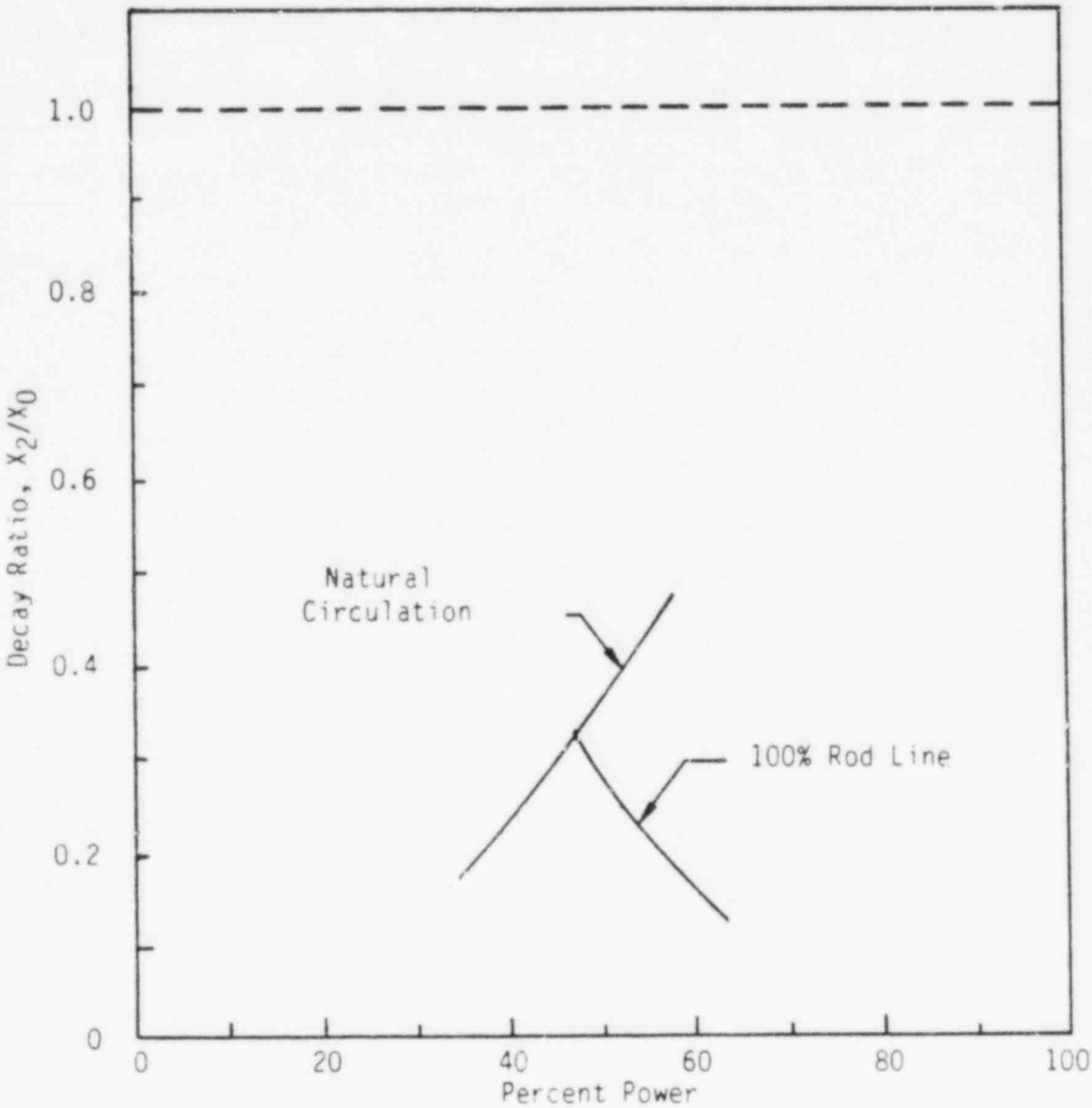
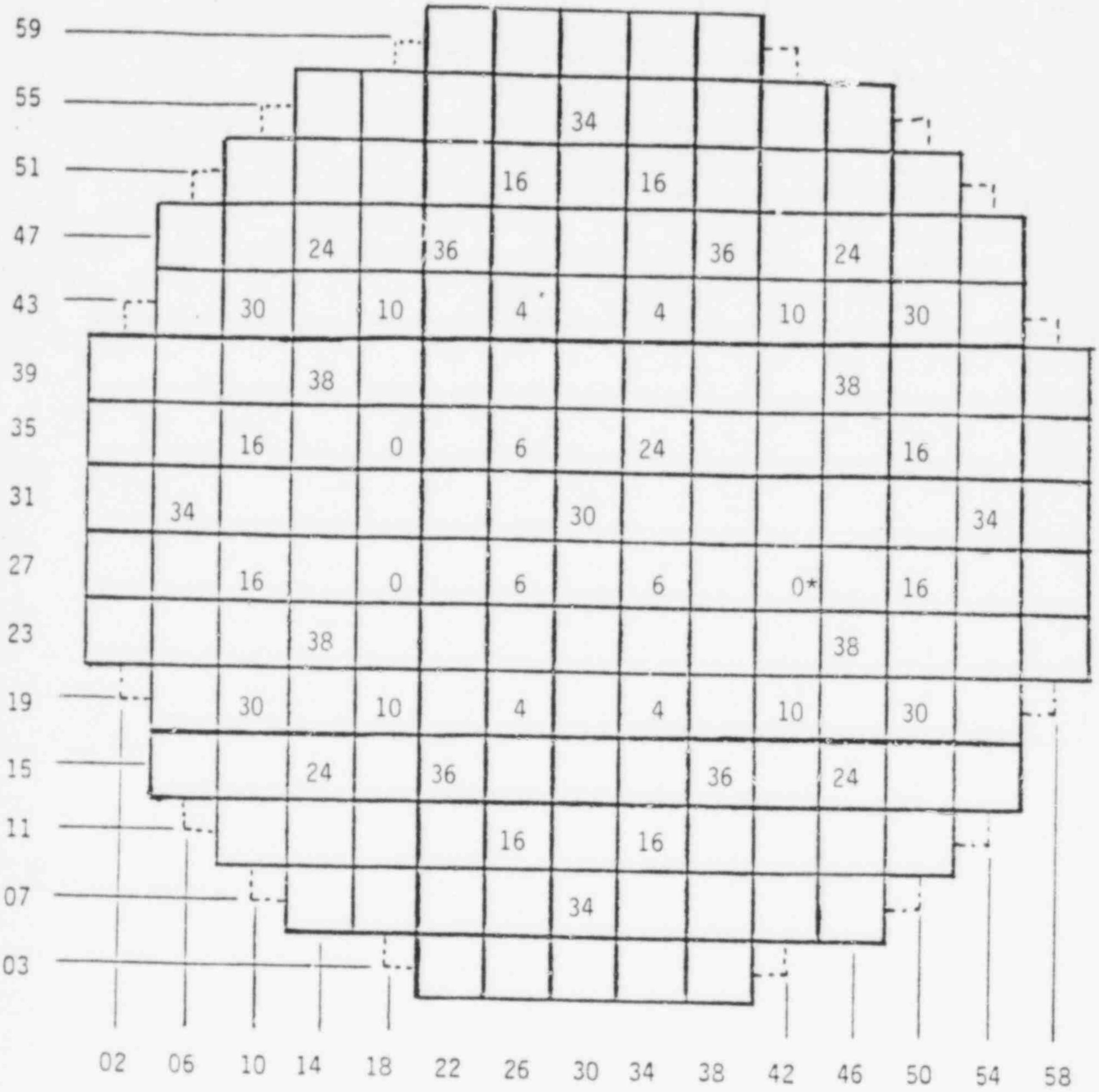


Figure 4.3 Decay Ratio vs. Reactor Power



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

Figure 5.1 Starting Control Rod Pattern for Control Rod Withdrawal Analysis

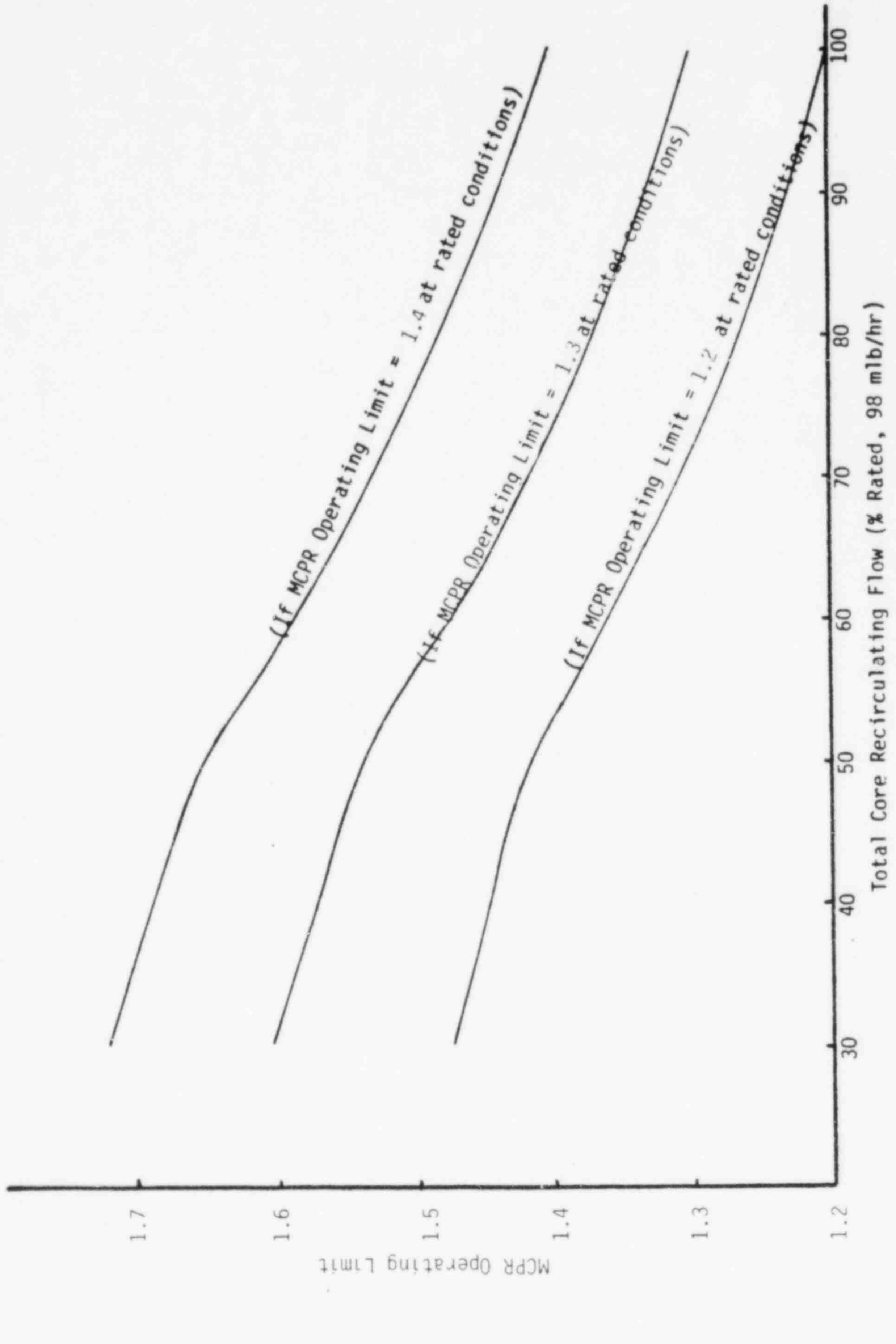


Figure 5.3a MCPR for Automatic Flow Control (AFC)

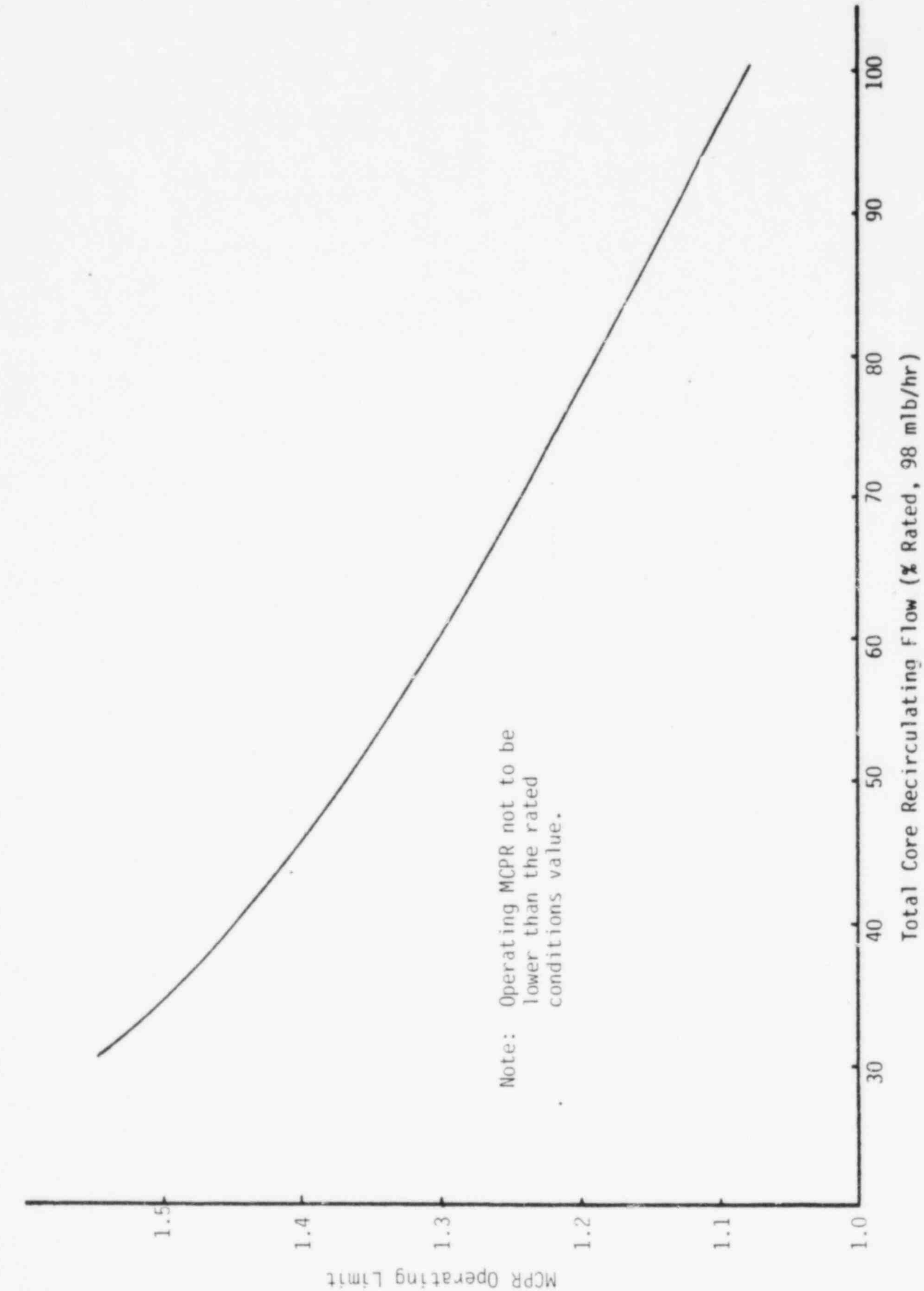


Figure 5.3b MCPR for Automatic and Manual Flow Control Modes

APPENDIX A

SURVEILLANCE REQUIREMENTS

APPENDIX A - SURVEILLANCE REQUIREMENTS

The thermal margin (MCPR) requirements associated with the generator load rejection transient without bypass to the condenser (LRWB) are based on a statistical combination of uncertainties in calculated parameters and measured plant performance in the area of control rod drive performance. The Plant Technical Specifications require that control rod drive performance be monitored on an individual rod basis at regular intervals. This Appendix provides for modification of MCPR operating limits if the measured control rod drive performance falls outside the statistical basis used in the thermal margin calculation.

For a mean control rod insertion time to 90% insertion of 2.58 seconds or less, the MCPR operating limits established by the statistical evaluation of the LRWB transient are valid. For a mean 90% insertion time corresponding to the Technical Specification limit of 3.50 seconds, an additional thermal margin conservatism of 0.05 is required. Between those two values, the MCPR operating limit should be determined by the following formula:

$$MCPR_S = MCPR_a + 0.054T - 0.140$$

where:

$MCPR_S$ = Operating Limit MCPR adjusted for observed scram time statistical behavior;

$MCPR_a$ = Operating Limit MCPR obtained from cycle analysis; and

T = Statistical mean of observed scram insertion times to the 90% insertion point.

APPENDIX B

ASEA-ATOM DEVELOPMENTAL CONTROL BLADES

B.1 PROGRAM DESCRIPTION

Cycle 9 operation for the Dresden Unit 3 nuclear plant will utilize a total of eight (8) ASEA-ATOM (AA) control blades. Four of these blades will contain a single control zone wherein the primary control material will be B_4C. The remaining four blades will contain two control zones: a zone equivalent to that of the first set of AA blades and a second zone containing hafnium as the primary control material. This second zone is located at the top six inches of the control blade.

The fuel management strategy projected for Cycle 9 operation is the Single Rod Sequence (SRS) mode. The core loading consists of grouping higher exposed fuel assemblies around a single control blade withdrawal sequence. It is only these control blades which are scheduled for insertion and used for core control during the cycle. In order to acquire operational experience, the locations of the AA control blades were selected such that the AA control blades constitute part of the single withdrawal sequence. Consequently, the AA control blades will reside in relatively low reactivity (low control rod worth) positions within the core. The locations of the AA control blades are shown in Figure B.1.

B.2 LICENSING ANALYSIS

On an equal fuel assembly basis, the reactivity worth of both types of AA control blades is estimated to be approximately 9% higher than anticipated for the current control blade worths (AA Report #TR-BR-82-98). Hence, the establishment of the Cycle 9 operating limits has included an evaluation of the impact of the AA control blades upon these limits. A description of the assumptions and analyses performed in support of this program is provided.

B.2.1 Plant Transient Analysis

The analyses associated with establishing reactor operating limits as a result of anticipated plant transients utilized a scram reactivity worth associated with the core comprised of all control blades of the current design. As a consequence, the scram reactivity worth does not take credit for the higher worth associated with the AA control blades and envelopes the anticipated scram reactivity associated with the core comprised of the current control blades and the eight AA blades. Hence, the reactor operating limits reported for Cycle 9 are conservative with respect to the utilization of the AA control blades.

B.2.2 Loss-of-Coolant Accident Analysis

The analysis of the postulated loss-of-coolant accident utilizes a scram reactivity worth curve which conservatively envelopes the anticipated scram curve expected for Cycle 9. This curve is subsequently used in determining the operating limits (MAPLHGR) for Cycle 9 and as such conservatively envelopes the scram reactivity curve expected with the utilization of the AA control blades.

B.2.3 Control Rod Withdrawal Error Analysis

The analysis of an inadvertent withdrawal of a control rod (CRWE) was performed utilizing the approved methodology[1] and explicitly modeled the presence of the eight AA control blades. Hence, the limiting control rod pattern was selected including any effect due to the presence of the AA control blades. As was previously indicated, the withdrawn control rod was determined not to be an AA control rod. Hence, the results of Section 5.4 report the limiting CRWE for Cycle 9. This result is primarily due to locating the AA blade in a low reactivity position within the core. Any change in the

location of the AA control blades for reactor operation in cycles subsequent to Cycle 9 will be reported and analyzed for those reactor cycles.

B.2.4 Control Rod Drop Accident Analysis

The approved methodology for the analysis of the dropped control rod(1) correlates the fuel rod deposited enthalpy to four independent parameters: control rod worth, Doppler coefficient, delayed neutron fraction, and four bundle local peaking. Of these four, only the control rod worth is affected by the utilization of the AA control blades. The values of the four independent parameters for Cycle 9 and the highest worth AA control blades are:

Dropped Rod Worth	.0032
Doppler Coefficient	$-10.18\times10^{-6}\ \frac{1}{K}\frac{\Delta K}{\Delta T}$, $(^{o}F)^{-1}$
Delayed Neutron Fraction	0.0055
Four Bundle Local Peaking	1.116

The resulting fuel rod deposited enthalpy is 36 cal/gm, which is well within the current 280 cal/gm limit.

B.2.5 Fuel Bundle Loading Error Analysis

The use of AA developmental control blades has no impact on the analysis of fuel bundle mislocation and misorientation errors.

(1) Safety Evaluation for the Exxon Nuclear Company Topical Report: "Exxon Nuclear Methodology for Boiling Water Reactors - Neutronics Methods for Design and Analysis," [XN-NF-80-19(P)], Volume 1, dated May 1980, Supplements 1 and 2 dated April 1981.

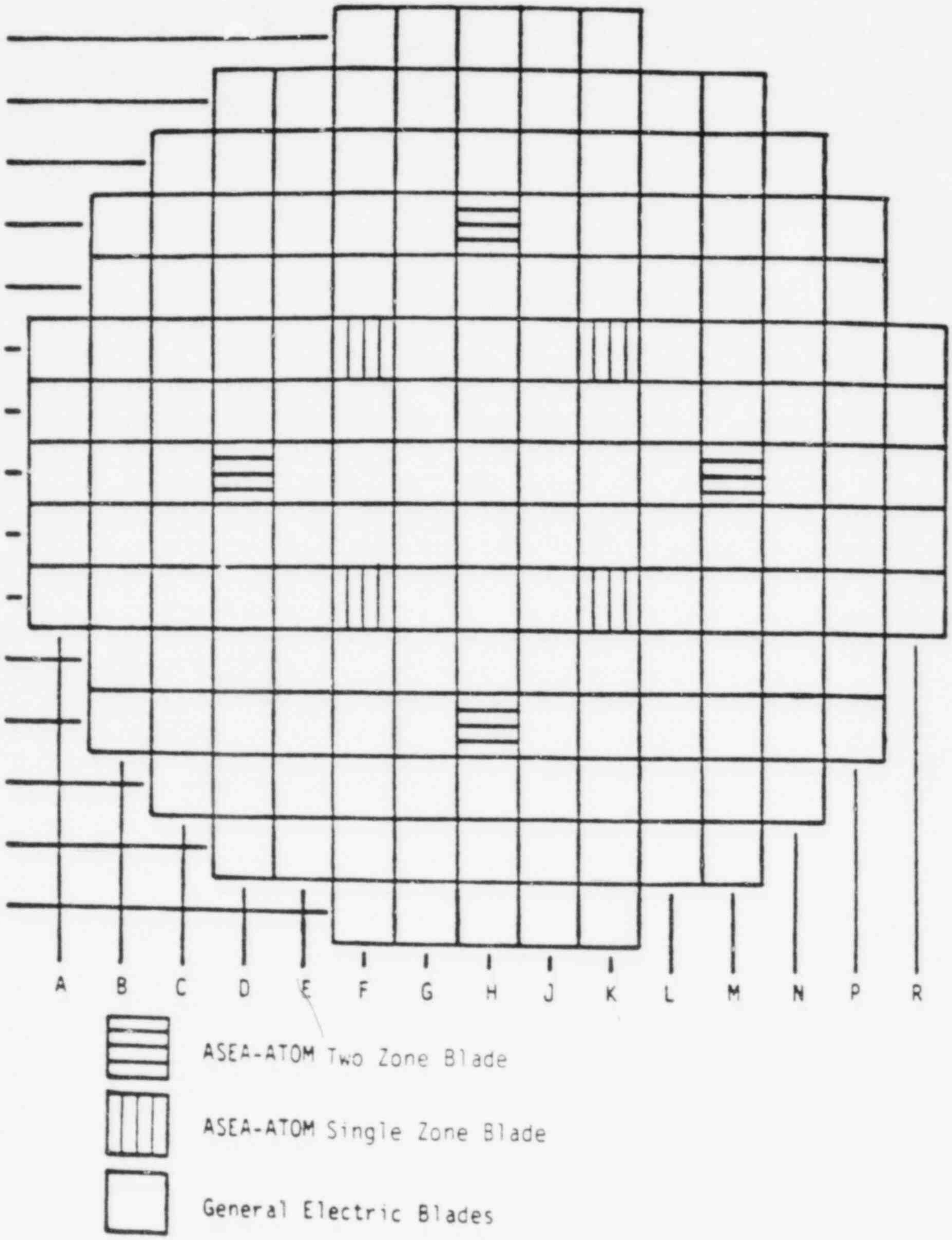


Figure B.1 Control Blade Locations for Dresden Unit 3

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DRESDEN UNIT 3 CYCLE 9 RELOAD ANALYSIS

Distribution

JC Chandler
RE Collingham
GC Cooke
SL Garrett
SE Jensen
WV Kayser
RH Kelley
TL Krysinski
JL Maryott
GF Owsley
RB Stout
RI Wescott
HE Williamson

CECo/LC O'Malley (60)

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