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Revision 1

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DRESDEN UNIT 2 CYCLE 9 RELOAD ANALYSIS

Mechanical, Thermal Hydraulic, and Nuclear Design Analyses for ENC XN-1 Reload Fuel

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EXXON NUCLEAR COMPANY, Inc.

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NOVEMBER 1982

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

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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-1) reload for Dresden Unit 2, which is scheduled to commence operation in the Spring of 1983. Dresden 2 is the second BWR/3 to be licensed on the basis of ENC analyses and is a sister plant to Dresden 3, which began operation with ENC fuel in April 1982. All limits and analyses reported herein are consistent with their counterparts in the Dresden 3 documentation.

The Cycle 9 core will comprise 224 unirradiated reload fuel assemblies fabricated by ENC, 384 once- and twice-irradiated Type 8x8R assemblies fabricated by General Electric Company (G.E.), and 116 G.E. Type 8x8 fuel assemblies irradiated from three to five cycles each. Except as noted below, 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 will involve the use of four Lead Test Assemblies (LTAs) placed in symmetrical, non-limiting locations in the core. The LTAs, [ ]* are described in Appendix A. Operating limits for the LTAs are also given in Appendix A.

*Brackets identify ENC proprietary information.

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 for the Dresden 2 Type XN-1 fuel	
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

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Core Exposure at BOC9, MWD/MT	13,096
Core Exposure at EOC9, MWD/MT	20,826
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BOC9 Cold K-effective, All Rods Out	1.111
BOC9 Cold K-effective, All Rods In	.958
BOC9 Cold K-effective, Strongest Rod Out	.989
Technical Specification R-Value	.04%(2)
SBLC Reactivity, 70°F, 600 ppm	.950

(1) Nominal Value/Value Used in Shutdown Reactivity Calculations.
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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 Tr
Maximum Pressure	1349.3	~~1351.5~~ psig
Maximum Sensed Pressure	1324.9	~~1327.2~~ psig

5.4 Control Rod Withdrawal Error (CRWE)

Starting Control Rod Pattern for Analysis	Figure 5.1

Rod Block Setting	Distance Withdrawn	ΔCPR ENC 8x8	ΔCPR GE 8x8,8x8R
106	4.5 ft.	0.10	0.08
107	5.0	0.11	0.09
108	5.5	0.12	0.10
109	6.0	0.13	0.11
110(1)	6.0	0.13	0.11

5.5 Fuel Loading Error

	ENC 8x8	GE 8x8,8x8R
ΔCPR	0.14	0.14

5.6 Determination of Thermal Margins — Table 5.1

MCPR Operating Limits at Rated Conditions

Fuel Type	MCPR Operating Limit
ENC XN-1 8x8	1.31
GE 8x8, 8x8R	1.31

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

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

Bundle Average Burnup (MWD/MT)	MAPLHGR	Peak Clad Temperature (^{o}F)	Peak Local MWR (%)
0	13.0 kw/ft	1900	0.8
12,000	13.0	1856	0.7

6.2 CONTROL ROD DROP ACCIDENT — See XN-NF-80-19, Vol.

Dropped Control Rod Worth	7.8 mk
Doppler Coefficient (773°F)	$-9.8\times10^{-6}\ \frac{1}{k}\frac{\Delta K}{\Delta T}\ (^{o}F)^{-1}$
Effective Delayed Neutron Fraction	0.0055
Four Bundle Local Peaking Factor	1.19
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Bundle Average Exposure	MAPLHGR
0 MWD/MT	13.0 kw/ft
12,000	13.0

7.2.2 Minimum Critical Power Ratio

Fuel Type	MCPR
XN-1 8x8	1.31
GE 8x8, 8x8R	1.31

Reduced Flow MCPR Limits

Automatic Flow Control	Figure 5.3a
All Conditions	Figure 5.3b

7.2.3 Surveillance Requirements

See Appendix B.

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, "Plant Transient Analysis for Dresden Unit 2 Cycle 9," XN-NF-82-84, Revision 1 (November 1982).

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 BWR3 LOCA Analysis Using the ENC EXEM Evaluation Model," XN-NF-81-71(A) (October 1981).

9.6 D. J. Braun and P. J. Valentine, "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model; MAPLHGR Results," XN-NF-82-88, Revision 1 (November 1982).

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

9.8 [ ]

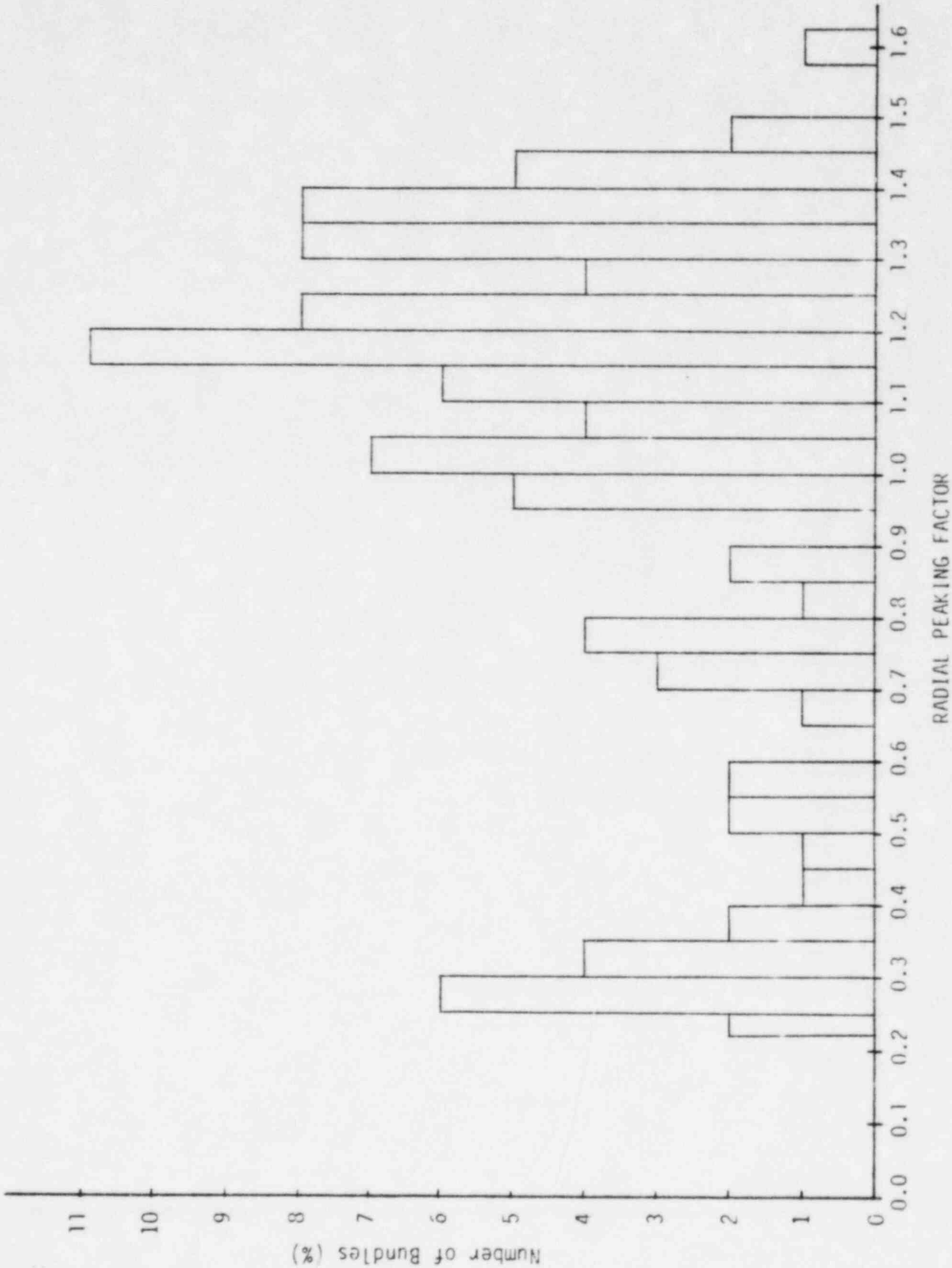


Figure 3.1 Dresden-2 Cycle 9 Safety Limit Radial Power Histogram

L[1] 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 0.96	M 1.06
ML 1.07	M 1.06	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, vertical)

W I D E

Figure 3.2 Dresden-2 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-1 8x8
(Enriched Lattice 3.02 w/o U-235)

C2	D0	C1	C2	C2	D0	C1	C2	C2	D0	C1	C2	C2	C2	A5
D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	D0	A4	B4
C1	D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	C1	B4
C2	C1	D0	C2	C2	C1	D0	C2	C2	C1	D0	C2	D0	C2	A3
C2	D0	C1	C2	C2	D0	C1	C2	C2	D0	C1	D0	C1	A3	A5
D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	D0	C1	C2	A3	
C1	D0	C1	D0	C1	D0	C1	E0	C1	D0	C1	D0	A3		
C2	C1	D0	C2	C2	C1	D0	C2	C2	C1	D0	C2	B4		
C2	D0	C1	C2	C2	D0	C1	C2	C2	D0	C1	A3	A3		
D0	C1	D0	C1	D0	C1	D0	C1	D0	C2	A3	A3			
C1	D0	C1	D0	C1	D0	C1	D0	C1	A3	A5				
C2	C1	D0	C2	D0	C1	D0	C2	A3	A3					
C2	D0	C1	D0	C1	C2	A3	B4	A3						
C2	A3	C1	C2	A3	A3									
A5	B4	B4	A3	A5										

XY

X = Fuel Type
Y = Cycles Irradiated

Fuel Type	Number of Assemblies	Description
A	88	GE 8x8 2.50 w/o U-235
B	28	GE 8x8 2.62 w/o U-235
C	384	GE 8x8R 2.65 w/o U-235
D	220	XN-1 8x8 2.83 w/o U-235
E	4	LTA [ ]

Figure 4.2 Dresden Unit 2 Cycle 9 Reference Loading Pattern (One Quarter of Symmetrical Core Loading)

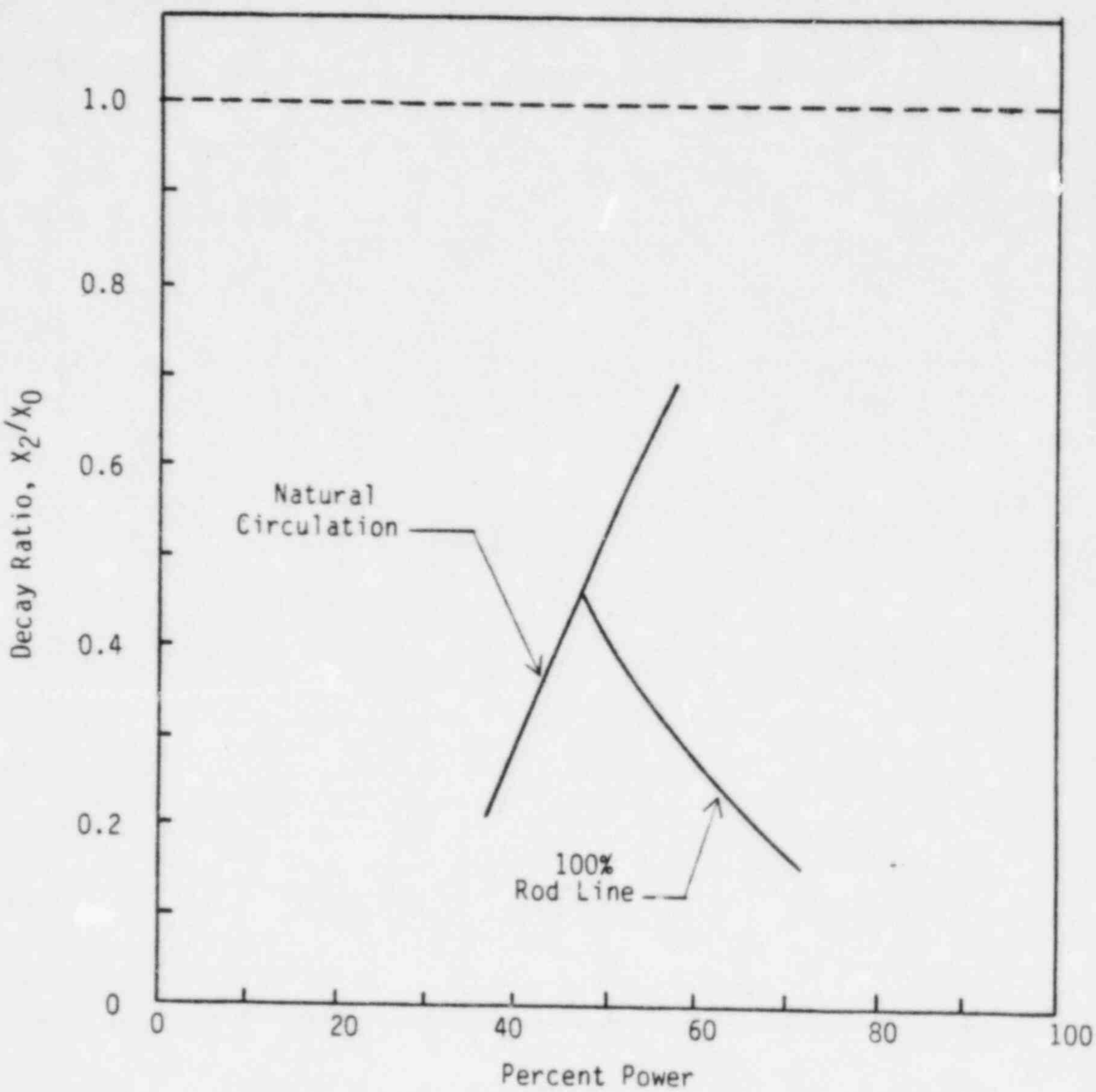
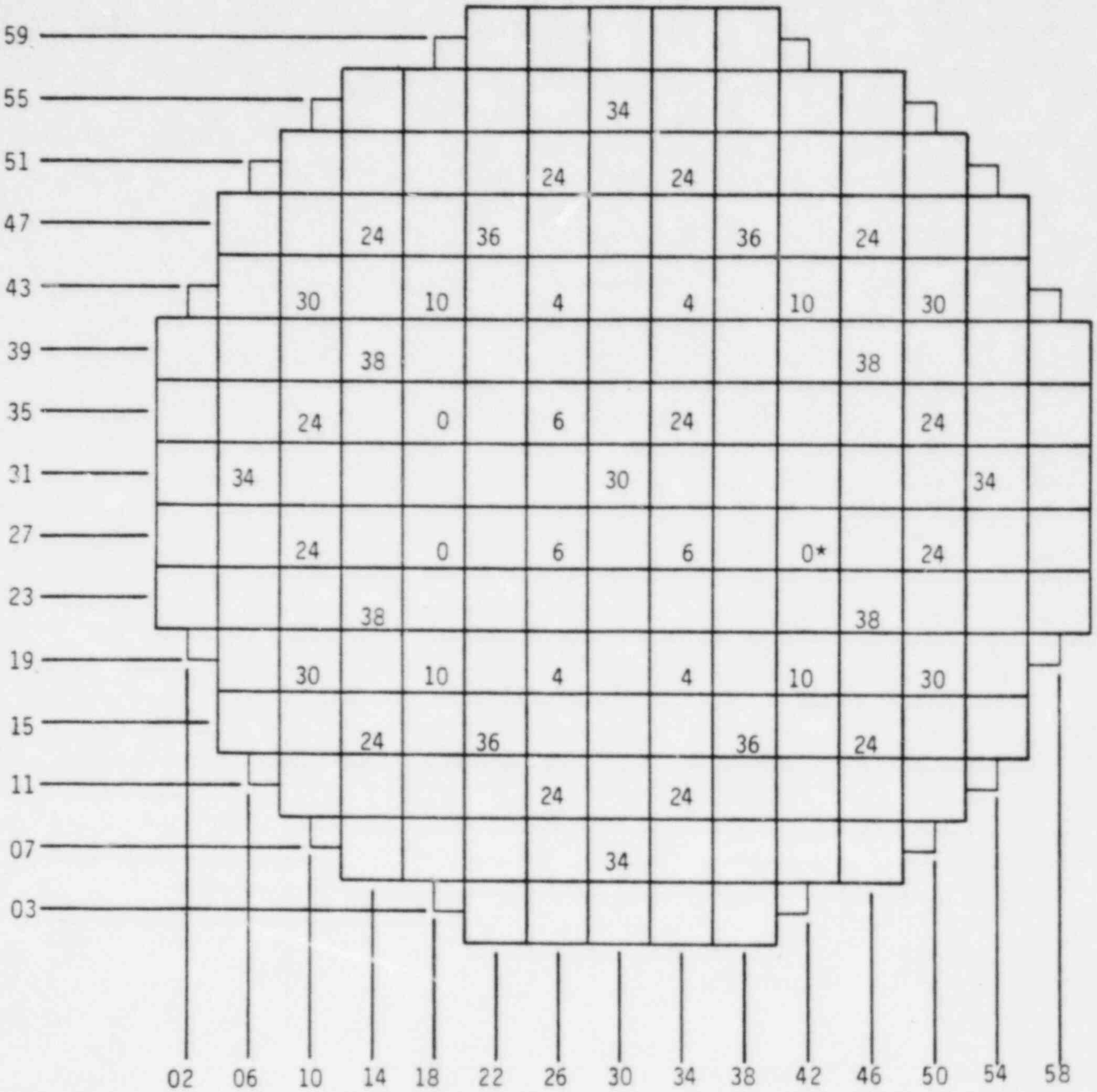


Figure 4.3 Decay Ratio vs. Reactor Power



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

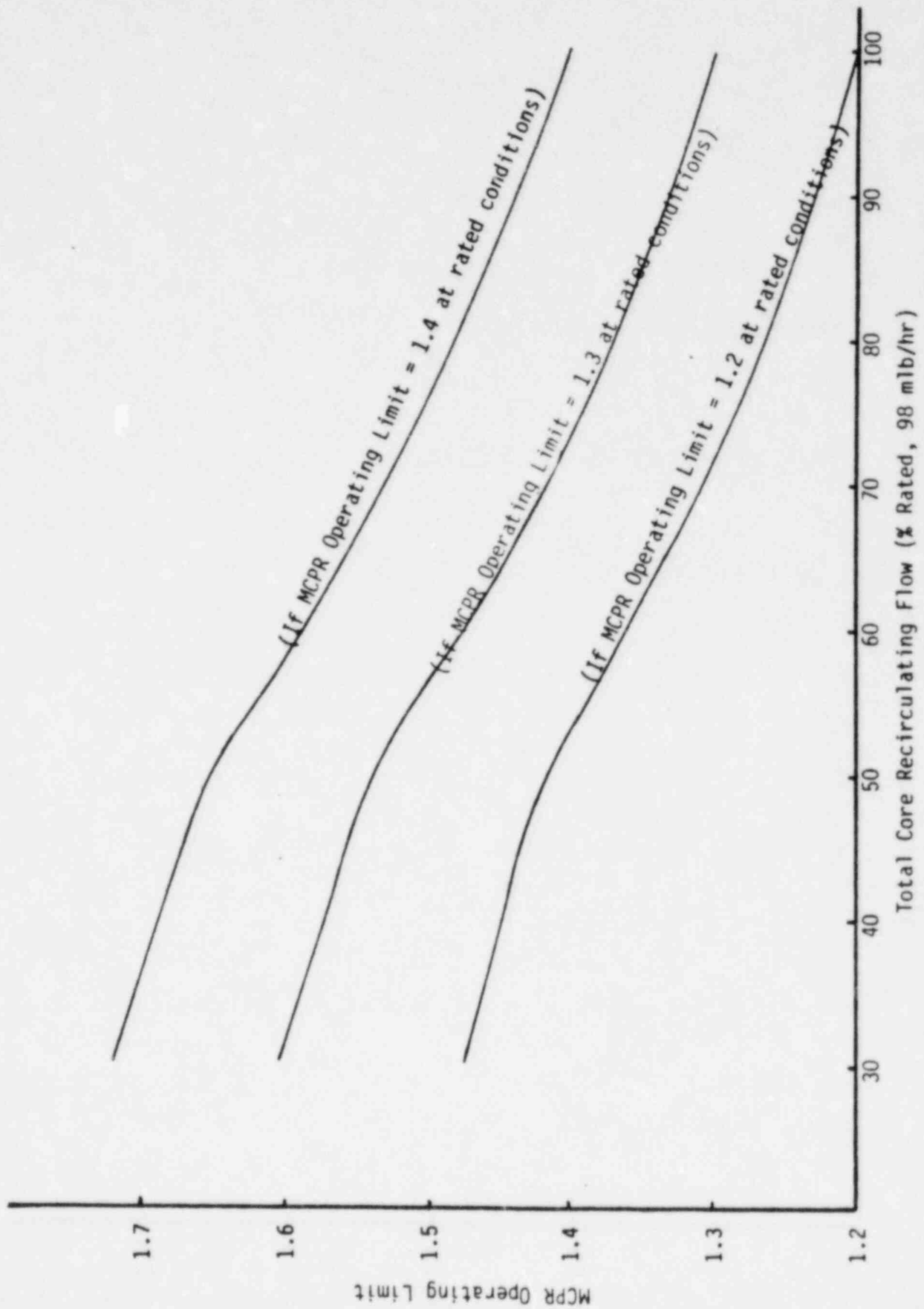


Figure 5.3a MCPR for Automatic Flow Control (AFC)

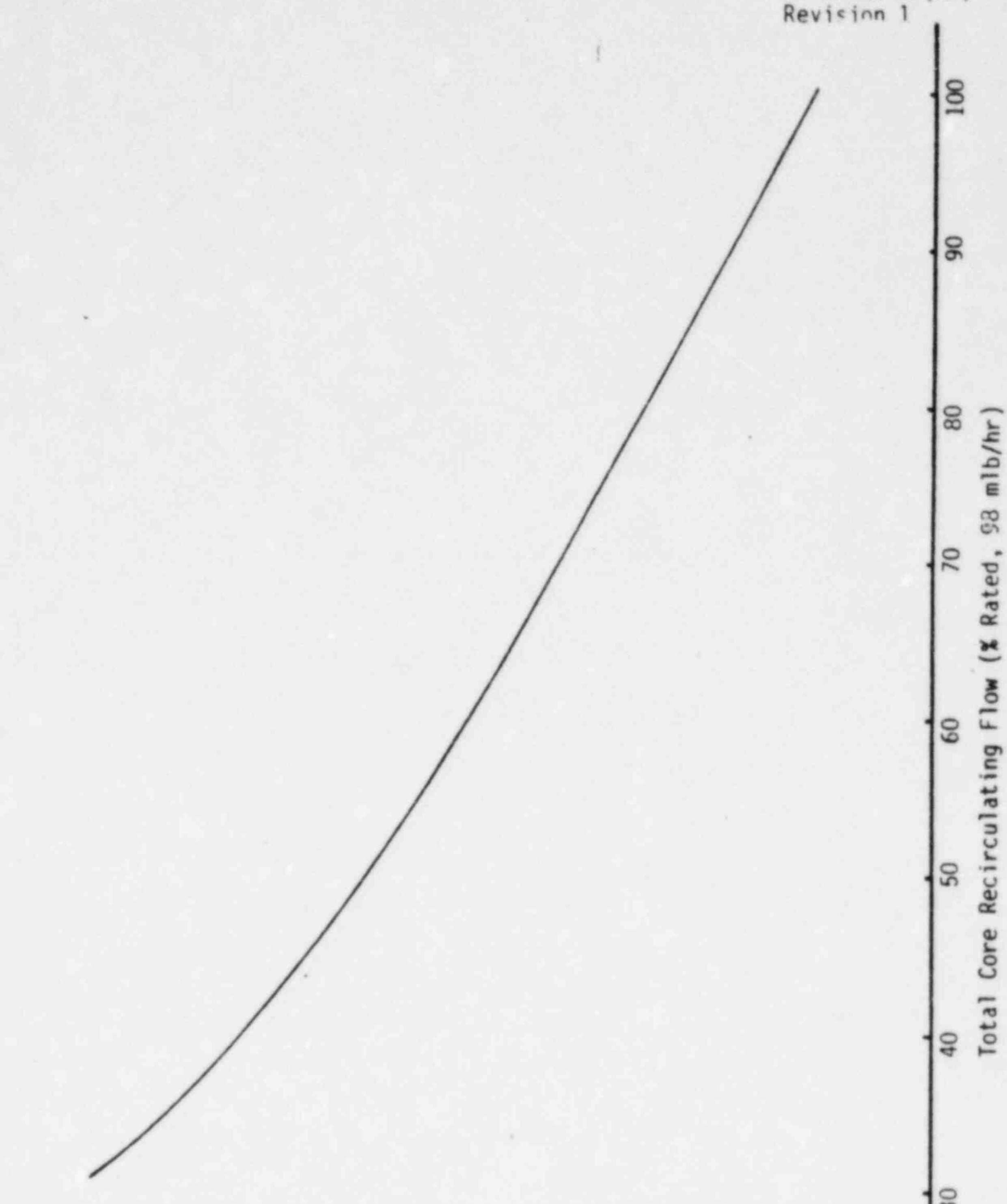


Figure 5.3b MCPR for All Conditions

Table 4.1 Dresden 2 Reload Batch XN-1
Neutronic Design Values

Fuel Pellet	Reference 9.1
Fuel Rod	Reference 9.1
Fuel Assembly	Reference 9.1
Fuel Assembly Loading, $KgUO_2$	197.2
Fuel Assembly Loading, KgU	173.8
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, ^{o}F	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	
Absorber material	B_4C
Total blade span, inch	9.750
Total central support span, inch	1.562
Blade thickness, inch	0.3120

Table 4.1 Dresden 2 Reload Batch XN-1
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%	114.5%	350.3%	1281.1 psig	1.31/1.31
FWCF	COTRANSA	EOC9	100%	100%	116.9%	198.4%	1207.2	1.26/1.26
LFWH	PTSBWR3	EOC9	100%	100%	<120.0%	[illegible]0.1%	1039.9	1.21/1.21
CRWE(1)	XTGBWR	BOC9	100%	100%	-	-	-	1.18/1.16

(1) Rod Block setting of 110% selected for Cycle 9 operation.
(2) Indicated limits for ENC 8x8 fuel/G.E. 8x8 fuel.

APPENDIX A

LEAD TEST ASSEMBLIES

[ ]

A.1 PROGRAM DESCRIPTION

Cycle 9 operation of Dresden 2 will include a lead test assembly program of [ ] fuel assemblies designed and fabricated by Exxon Nuclear Company. The locations of the four lead test assemblies (LTAs) were selected on the basis of allowing adequate exposure accumulation [ ] and not to result in additional operating limitations. The locations of the LTAs are shown in Figure 4.2 of this report.

A.2 FUEL STORAGE CRITICALITY

A comparison of the maximum lattice K_∞ values for the ENC 8x8, ENC LTA, and GE 8x8 fuel designs with and without gadolinia is given in Table A.2. The calculations were performed with the ENC XFYRE code. As shown, the ENC 8x8 fuel with no gadolinia has the highest reactivity.

The high density spent fuel storage racks were analyzed by NSC and were found to meet the fuel storage K_{eff} requirements of 0.95 for the limiting ENC 8x8 3.02 w/o U-235 fuel design with no gadolinia(1). Based on the bundle reactivity comparisons as shown in Table A.1, the high density storage racks are acceptable for storage of the ENC LTA fuel with considerable reactivity margin when the gadolinia is considered.

Previous analyses have demonstrated that the dry storage racks and the spent fuel storage racks are acceptable for storage of the GE 8x8 (2.82 w/o U-235) Cycle 8 reload fuel. Since the ENC fuel with gadolinia has a lower maximum K_∞ than the GE fuel with gadolinia, the ENC 8x8 and LTA fuel may be

---

(1) Wong, Kin W. (testimony), Atomic Safety and Licensing Board, January 21, 1982.

stored in the dry storage and spent fuel storage racks without compromising the technical specification fuel storage requirements.

A.3 STABILITY ANALYSIS

The inclusion of four LTAs does not impact the core stability analysis reported in Section 4.2 of the body of this report. The LTA fuel was analyzed for channel hydrodynamic stability with a resulting maximum decay ratio of 0.29.

A.4 OPERATING LIMITS

Operating limits for the LTAs are established based on consistence with the limits calculated for ENC production fuel. [

]

Extrapolation of the production fuel bundle power limits as determined using ENC's plant transient methodology (Ref. 8.8) results in an operating limit MCPR of 1.35 for the LTA fuel. [

]

Operating limit APLHGR values were determined by extrapolating the 8x8 APLHGR limits from Section 6.1.3 of the body of this report [ ]

[

]

Table A.1 Fuel Assembly Maximum Lattice K_∞ Values

Fuel Design	Maximum K_∞ - No Gadolinia/ (Exposure, MWD/MTU)	Maximum K_∞ - With Gadolinia/ (Exposure, MWD/MTU)
ENC 8x8 3.02 w/o U-235 Five Gd Rods 3.5 w/o Gd_2O_3	1.339 (0)	1.224 (8000)
ENC LTA [ ]	1.334 (0)	1.220 (8000)
GE 8x8 2.82 w/o U-235 (8DRL282L)	1.321 (0)	1.231 (6000)

Table A.2 LTA MAPLHGR Limits

Bundle Average Burnup, MWD/MT	MAPLHGR, kW/ft
0	10.2
12,000	10.2

APPENDIX B

SURVEILLANCE REQUIREMENTS

APPENDIX B - 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 2.74 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 0.07 is required. Between those two values, the MCPR operating limit should be determined by the following formula:

$$MCPR_s = MCPR_a + 0.092T - 0.252$$

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.

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