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PRAIRIE ISLAND UNIT 1 NUCLEAR PLANT CYCLE 5 SAFETY ANALYSIS REPORT

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EXXON NUCLEAR COMPANY, Inc.

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PRAIRIE ISLAND UNIT 1 NUCLEAR PLANT CYCLE 5

SAFETY ANALYSIS REPORT

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1.0 INTRODUCTION AND SUMMARY

The Prairie Island 1 nuclear plant will operate in Cycle 5 beginning in spring of 1979 with one region of fuel supplied by Exxon Nuclear Company (ENC). The composition of the core during Cycle 5 will be 40 ENC assemblies and 81 Westinghouse assemblies. Included in Cycle 5 is a Gadolinia Demonstration Program with 64 fuel pins containing 1 w/o gadolinia. The gadolinia bearing pins are uniformly dispersed among sixteen fuel assemblies.

The characteristics of the fuel and of the reloaded core results in conformance with existing Technical Specification limits regarding shutdown margin provisions and thermal limits. This document provides the thermal hydraulic analysis results for the ENC supplied fuel and the neutronic analysis for the plant during Cycle 5 operation. The ENC fuel design, plant transient analysis, ECCS analysis, and the core rod ejection analysis for the ENC supplied fuel are given in References 1, 2, 3, and 4, respectively.

2.0 OPERATING HISTORY OF THE REFERENCE CYCLE

Prairie Island 1 Cycle 4 has been chosen as the reference cycle with respect to Cycle 5 due to the close resemblance of the neutronic characteristics between these two cycles. The Cycle 4 operation began on April 19, 1978, and as of August 31, 1978 the core has accrued about 4,000 MWD/MT. Cycle 4 loading included 40 fresh Westinghouse fuel assemblies and 81 exposed Westinghouse assemblies.

The measured power peaking factors at hot-full-power, equilibrium xenon conditions, have remained below the Technical Specification limits throughout Cycle 4. Cycle 4 operation has typically been rod free with the control bank positioned in the range of 210 to 220 steps, 228 steps being fully withdrawn. It is anticipated that similar control bank insertions will be used in Cycle 5.

The critical boron concentration as calculated by ENC for Cycle 4 has agreed to within about 30 ppm compared to the observed values (see Figure 2.1). Also the power distribution predicted by ENC has agreed to within ±4.0 percent of the measured values (see Figure 2.2 for typical comparison).

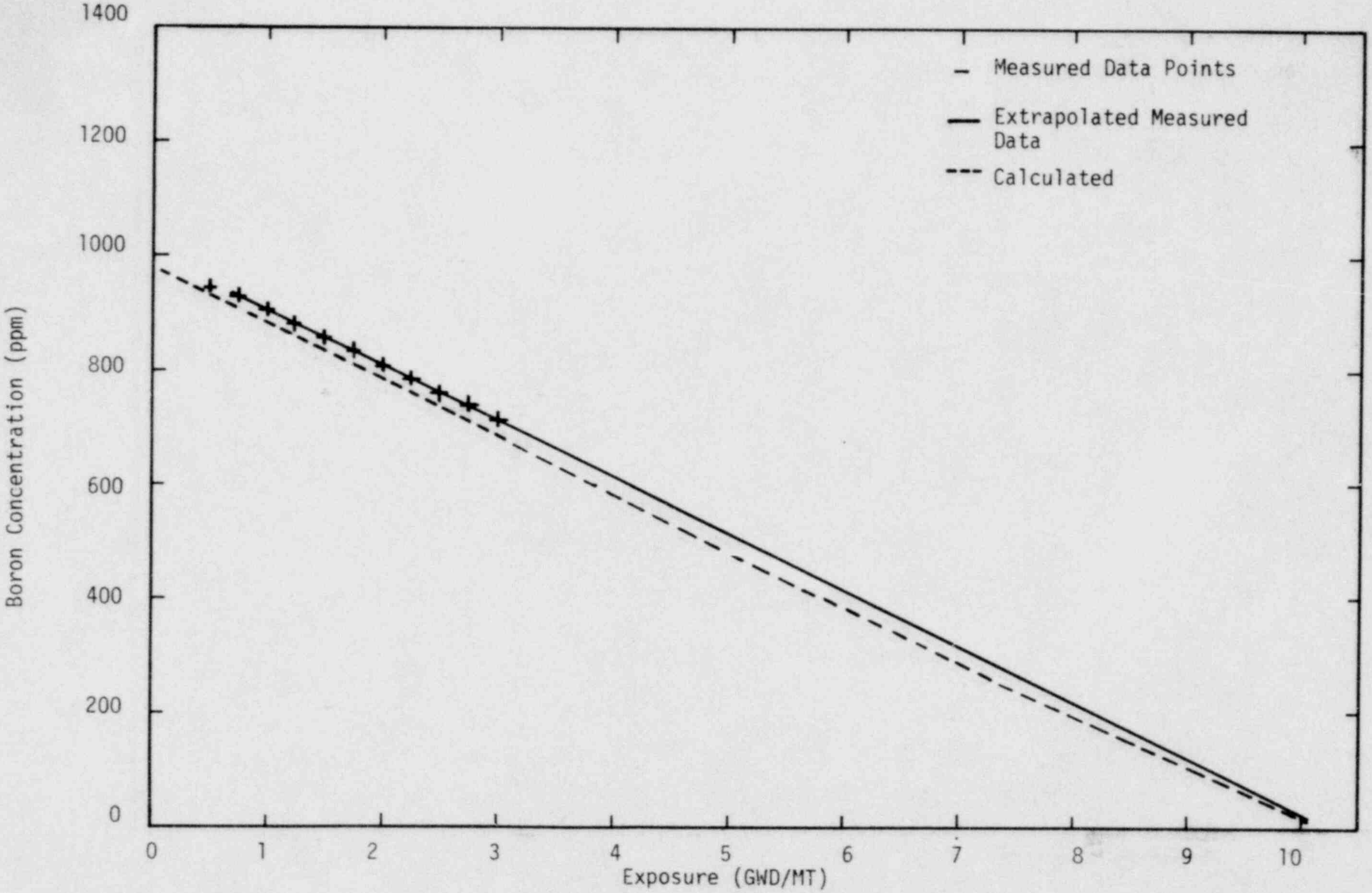

Figure 2.1 Prairie Island 1 Cycle 4 Critical Boron Vs. Exposure

	7	8	9	10	11	12	13
G	.930 .940 1.08	1.123 1.148 2.23	.949 .961 1.26	.932 .925 -0.75	.872 .878 0.69	.884 .848 -4.07	.827 .835 0.97
H	1.127 1.142 1.33	1.145 1.155 0.87	1.053 1.049 -0.38	1.192 1.172 -1.68	.940 .928 -1.28	1.190 1.203 1.09	.696 .708 1.72
I	.946 .954 0.85	1.022 1.010 -1.17	1.237 1.224 -1.05	1.037 1.015 -2.12	1.127 1.149 1.95	1.054 1.039 -1.42	
J	.917 .923 0.65	1.186 1.168 -1.52	1.037 1.011 -2.51	1.200 1.223 1.92	1.052 1.082 2.85	.732 .726 -0.82	
K	.866 .883 1.96	.936 .926 -1.07	1.127 1.139 1.06	1.054 1.071 1.61	.792 .790 -0.25		
L	.880 .889 1.02	1.187 1.204 1.43	1.054 1.029 -2.37	.732 .720 -1.64			
M	.825 .833 0.97	.694 .703 1.30					

Calculated (XTGPWR)
Measured Assembly Power
M-C/C * 100

Calculated A.O. = -1.083%
Measured A.O. = +0.593%

	Calculated	Measured	Percent Difference (m-c)/c * 100
F_Q	1.623	1.620	-0.18%
$F_{\Delta H}$ pin	1.412	1.4215	0.67%
F_z	1.13	1.14	0.88%

Figure 2.2 Prairie Island Unit 1 Power Distribution Comparison to Map 104-11, HFP, 2,000 MWD/MT

3.0 GENERAL DESCRIPTION

The Prairie Island 1 reactor consists of 121 assemblies, each having a 14x14 fuel rod array. Each assembly contains 179 fuel rods, 16 RCC guide tubes, and 1 instrumentation tube. The fuel rods consist of slightly enriched UO_2 pellets inserted into zircaloy tubes. In the case of the gadolinia bearing rods, the gadolinia is uniformly blended with the enriched UO_2. The RCC guide tubes and the instrumentation tube are made of zircaloy. Each ENC assembly contains seven zircaloy spacers with Inconel springs; six of the spacers are located within the active fuel region.

The projected Cycle 5 loading pattern is shown in Figure 3.1 with the assemblies identified by their Cycle 4 location and Batch ID's. The initial enrichments of the various regions are listed in Table 3.1. BOC 5 exposures along with Batch ID's are shown in Figure 3.2. The core consists of 40 fresh ENC assemblies at 3.40 w/o loaded on the periphery and 81 Westinghouse assemblies scatter-loaded in the center portion of the core. Included in Cycle 5 is a Gadolinia Demonstration Program with 64 fuel pins containing 1 w/o gadolinia. The gadolinia bearing pins are uniformly dispersed among sixteen fuel assemblies. Pertinent fuel assembly parameters for the Cycle 5 fuel are depicted in Table 3.1.

Table 3.1 Prairie Island 1 Cycle 5 Fuel Assembly Design Parameters

	Batch				
	4	5	6	7A	7B*
Enrichment, wt% U-235	2.78	3.30	3.30	3.40	3.40
Number of Assemblies	1	40	40	24	16
Pellet Density % TD	94.40	94.40	94.50	94.00	94.00
Pellet-to-Clad Diametrical Gap Mil	7.5	7.5	7.5	7.5	7.5
Fuel Stack Height, inch	144.0	144.0	144.0	144.0	144.0
Batch Average Burnup at BOC 5 MWD/MT	27,208	22,827	9,592	0	0

*Batch 7B include four gadolinia bearing (1 w/o) fuel pins per assembly

	1	2	3	4	5	6	7	8	9	10	11	12	13
A						X40 7A	X1 7A	X5 7A					
B				M8 6	X32 7A	X36 7B	D8 5	X9 7B	X13 7A	M6 6			
C			X24 7A	X28 7B	C4 5	L8 6	F7 5	L6 6	C10 5	X17 7B	X21 7A		
D		H13 6	X20 7B	E5 5	L10 6	C5 5	C11 6	C9 5	L4 6	E9 5	X25 7B	H1 6	
E		X16 7A	D3 5	J12 6	D4 5	L9 6	J8 5	L5 6	D10 5	J2 6	D11 5	X29 7A	
F	X8 7A	X12 7B	H12 6	E3 5	I12 6	F6 5	M7 6	F8 5	I2 6	E11 5	H2 6	X33 7B	X37 7A
G	X4 7A	F4 5	G6 5	C3 6	F10 5	G13 6	* 4	G1 6	H4 5	K11 6	G8 5	H10 5	X2 7A
H	X39 7A	X35 7B	F12 6	I3 5	E12 6	H6 5	A7 6	H8 5	E2 6	I11 5	F2 6	X10 7B	X6 7A
I		X31 7A	J3 5	D12 6	J4 5	B9 6	D6 5	B5 6	J10 5	D2 6	J11 5	X14 7A	
J		F13 6	X27 7B	I5 5	B10 6	K5 5	K3 6	K9 5	B4 6	I9 5	X18 7B	F1 6	
K			X23 7A	X19 7B	K4 5	B8 6	H7 5	B6 6	K10 5	X26 7B	X22 7A		
L				A8 6	X15 7A	X11 7B	J6 5	X34 7B	X30 7A	A6 6			
M						X7 7A	X3 7A	X38 7A					

Cycle 4 location
Batch ID **

* Batch 4 assembly with approximate 27,000 MWD/MT Exposure
** See Table 3.1 for batch definitions

Figure 3.1 Prairie Island 1, Cycle 5, Loading Pattern

7	8	9	10	11	12	13	
27,208 4	9,154 6	21,186 5	7,112 6	25,504 5	19,849 5	0 7A	G
[illegible],137 5	25,161 5	11,328 6	24,385 5	12,848 6	0 7B	0 7A	H
21,144 5	11,324 6	22,010 5	7,976 6	21,939 5	0 7A		I
7,111 6	24,404 5	7,981 6	21,889 5	0 7B	7,696 6		J
25,491 5	12,827 6	21,984 5	0 7B	0 7A			K
19,832 5	0 7B	0 7A	7,684 6				L
0 7A	0 7A						M

BOC 5 Exposure, MWD/MT
Batch ID*

*See Table 3.1 for Batch Definitions

Figure 3.2 Prairie Island #1 BOC 5 Exposure Distribution

4.0 FUEL SYSTEM DESIGN

A description of the Exxon Nuclear supplied fuel design and design methods is contained in Reference 1. This fuel has been specifically designed to be compatible to the resident fuel supplied by Westinghouse.

5.0 NUCLEAR DESIGN

The neutronic characteristics of the projected Cycle 5 core are quite similar to those of the Cycle 4 (see Section 5.1).

The nuclear design bases for the Cycle 5 core are as follows:

1. The design shall permit operation within the Technical Specifications for Prairie Island 1 plant.
2. The length of Cycle 5 shall be determined on the basis of an assumed Cycle 4 length of 10,900 MWD/MT.
3. The Cycle 5 loading pattern shall be optimized to achieve power distributions and control rod reactivity worths according to the following constraints:
   a) The peak F_Q shall not exceed 2.21 and the peak $F_{\Delta H}$ shall not exceed 1.55 (including uncertainties) in any single fuel rod throughout the cycle under nominal full power operating conditions; and
   b) The scram worth of all rods minus the most reactive shall exceed BOC and EOC shutdown requirements.
4. The Cycle 5 core shall have a negative power coefficient.

The neutronic design methods utilized to ensure the above requirements are consistent with those described in References 5, 6, and 7.

5.1 PHYSICS CHARACTERISTICS

The neutronic characteristics of the Cycle 5 core are compared with those of Cycle 4 and are presented in Table 5.1. The data presented in the table indicates the neutronic similarity between Cycles 4 and 5. The Gadolinia Demonstration assemblies are predicted to have only a minor impact on overall core neutronic behavior. The small effect of the gadolinia is reflected in calculations for Cycle 5.

The reactivity coefficients of the Cycle 5 core are bounded by the coefficients used in the safety analysis. The predicted physics characteristics and the safety analysis for Cycle 5 are applicable for a Cycle 4 length of 10,900 $^{+600}_{-200}$ MWD/MT.

The boron letdown curve for Cycle 5 is shown in Figure 5.1. The curve indicates a BOC 5, no xenon, critical boron concentration of 1,396 ppm. At 100 MWD/MT, equilibrium xenon, the critical boron concentration is 1,060 ppm. The Cycle 5 length is projected to be 11,300 MWD/MT with 10 ppm of boron at EOC.

5.1.1 Power Distribution Considerations

Representative predicted power maps for Cycle 5 are shown in Figures 5.2 and 5.3 for BOC and EOC conditions, respectively. The power distributions were obtained from a three-dimensional model with moderator density and Doppler feedback effects incorporated. For the projected Cycle 5 loading pattern the calculated BOC nuclear power peaking factors, F_Q^N, $F_{\Delta H}^N$, and F_Z^N, are 1.680, 1.395 and 1.213, respectively. At EOC conditions the corresponding values are 1.465, 1.319 and 1.097. The Technical Specification limits relative to F_Q and $F_{\Delta H}$ are 2.21 and 1.55. The BOC F_Q^N value of 1.680 compares favorably with the Cycle 4 value of 1.78 in Table 5.1.

The control of the core power distribution can be accomplished by following the procedures as discussed in the report, XN-NF-77-57, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors Phase 2", January 1978,[12] which has been submitted to the NRC for approval. The results reported in that document provide the means for predicting the maximum $F_Q^T(z)$ distribution anticipated during operation under the PDC-II procedure taking

into account the incore measured equilibrium power distribution data. A comparison of this distribution with the Technical Specification limit curve assures that the Technical Specification limit can be projected by PDC-II procedure. Document XN-NF-77-57 describes the maximum possible variation in $F_Q^T(z)$, which can occur during operation when following the PDC-II procedure. This bounding variation in $F_Q^T(z)$ represents the maximum variation in $F_Q^T(z)$ when the axial offset is maintained within the range defined in the report ($\pm 5\%$ at full power condition).

5.1.2 Control Rod Reactivity Requirements

Detailed calculation of shutdown margins for Cycle 5 are compared with Cycle 4 data in Table 5.2. The ENC Plant Transient Simulation (PTS) Analysis indicates that at EOC the minimum required shutdown margin is 1,800 pcm based upon the steamline break accident analyzed for ENC fuel. A value of 2,000 pcm is used at EOC in the evaluation of the shutdown margin to be consistent with the Technical Specifications. The Westinghouse safety anlaysis[8] for Cycle 4 also assumed the minimum EOC shutdown requirements to be 2,000 pcm. The Cycle 5 analysis indicates excess shutdown margins of 1,539 pcm at the BOC and 598 pcm at the EOC. The Cycle 4 analysis indicated excess shutdown margins for that cycle of 530 pcm at the EOC. The Cycle 5 excess shutdown margins are seen to be similar to the Cycle 4 value.

The control rod groups and insertion limits for Cycle 5 will remain unchanged from Cycle 4. With these limits the nominal worth of the control bank, D-Bank, inserted to the insertion limits at HFP is 265 pcm at BOC and 354 pcm at EOC. The control rod shutdown requirements in Table 5.2 allow for a HFP D-bank insertion equivalent to 400 pcm for both BOC and EOC.

5.1.3 Moderator Temperature Coefficient Considerations

The moderator temperature coefficient is predicted to be +1 pcm/^{o}F at the hot zero power, all rods out, condition. A negative moderator temperature coefficient can be achieved through control rod insertion. For example with Bank D in and Bank C at 128 steps, the moderator temperature coefficient is predicted to be -1.5 pcm/^{o}F. Power operation up to 37% of rated power is permitted with this configuration. After xenon builds in, the control rods can be withdrawn and power escallation continued. The power coefficient will also be negative.

A Technical Specification change has been requested(9) which will require that the isothermal temperature coefficient be negative during power operation. The hot zero power all rods out isothermal coefficient is calculated to be -0.7 pcm/^{o}F and would meet the revised Technical Specification with no rod insertion.

5.2 ANALYTICAL METHODOLOGY

The methods used in the Cycle 5 core analyses are described in References 5, 6, and 7. In summary, the reference neutronic design analysis of the reload core was performed using the XTG (Reference 10) reactor simulator system. The input isotopics data were based on quarter core depletion calculations performed from Cycle 2 to Cycle 4 using the XTG code. The BOC 2 exposure distribution was obtained from plant data. The fuel shuffling between cycles was accounted for in the calculations.

Predicted values of F_Q, F_{xy}, and F_z were studied with the XTG reactor model. The calculational thermal-hydraulic feedback and axial exposure distribution effects on power shapes, rod worths, and cycle lifetime are explicitly included in the analysis.

5.3 GADOLINIUM DEMONSTRATION PROGRAM

The core loading for Cycle 5 will include 16 demonstration assemblies containing four gadolinia bearing (UO_2-Gd_2O_3) pins per assembly. The core loading pattern has been optimized to achieve a desirable power distribution while maximizing the benefit of the demonstration assemblies to reduce the beginning of cycle (BOC) boron concentration. The BOC worth of the gadolinia poison is predicted to be equivalent to the worth of 65 ppm soluble boron. The effect of the gadolinia poison is calculated to disappear by mid cycle.

The core locations of the gadolinia bearing assemblies are shown in Figure 5.2. The maximum assembly power is predicted to occur in assemblies other than those containing gadolinia. However, after the gadolinia has burned out, the maximum relative pin power, $F^N_{\Delta H}$, may occur in the demonstration assemblies but is predicted to be well below the Technical Specification limits. A detailed discussion of the gadolinia analysis can be found in the Appendix.

Table 5.1 Prairie Island 1 Neutronics Characteristics of Cycle 5 Compared with Cycle 4 Data

	Cycle 4		Cycle 5	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, equilibrium xenon (ppm)	1010[2]	10[2]	1060	10
HZP, ARO, No xenon (ppm)	1458[1]	--	1556	--
Moderator Temperature Coefficient				
HFP, (pcm/°F)	0	to -40[3]	-3.69	-29.0
HZP, (pcm/°F)	-1.3[2]	--	+1.00	-24.0
Doppler Coefficient, (pcm/°F)	-1.0	to -2.5[3]	-1.30	-1.63
Boron Worth, (pcm/ppm)				
HFP	--	--	-7.90	-8.89
HZP	-8.7[1]	--	--	--
Total Nuclear Peaking Factor				
F_Q^N, HFP	1.78[1]	--	1.680	1.465
Delayed Neutron Fraction	.0071	.0050[3]	.0060	.0053
Control Rod Worth of All Rods In Minus				
Most Reactive Rod, HZP, (pcm)	--	5960[3]	5543	5954
Excess Shutdown Margin (pcm)	--	530[3]	1539	598

(1) Measured data

(2) Extrapolated from measured data

(3) Reference 8

Table 5.2 Prairie Island 1 Control Rod Shutdown Margins and Requirements for Cycle 5

	Cycle 4**		Cycle 5	
	BOC	EOC	BOC	EOC
Control Rod Worth (HZP), pcm				
All rods inserted (ARI)	--	6840	6131	6602
ARI less most reactive (N-1)	--	5960	5543	5954
N-1 less 10% allowance ((N-1) * .9)	--	5360	4988	5358
Reactivity Insertion, pcm				
Moderator plus Doppler	--	---	1399	1710
Flux redistribution	--	---	600	600
Void	--	---	50	50
Sum of the above three	--	2330	2049	2360
Rod insertion allowance	--	500	400	400
Total Requirements	--	2830	2449	2760
Shutdown Margin (N-1) * .9 - Total requirements	--	2530	2539	2598
Required Shutdown Margin*	--	2000	1000	2000
Excess Shutdown Margin	--	530	1539	598

* Technical specification 3.10
** Calculated values from Reference 8

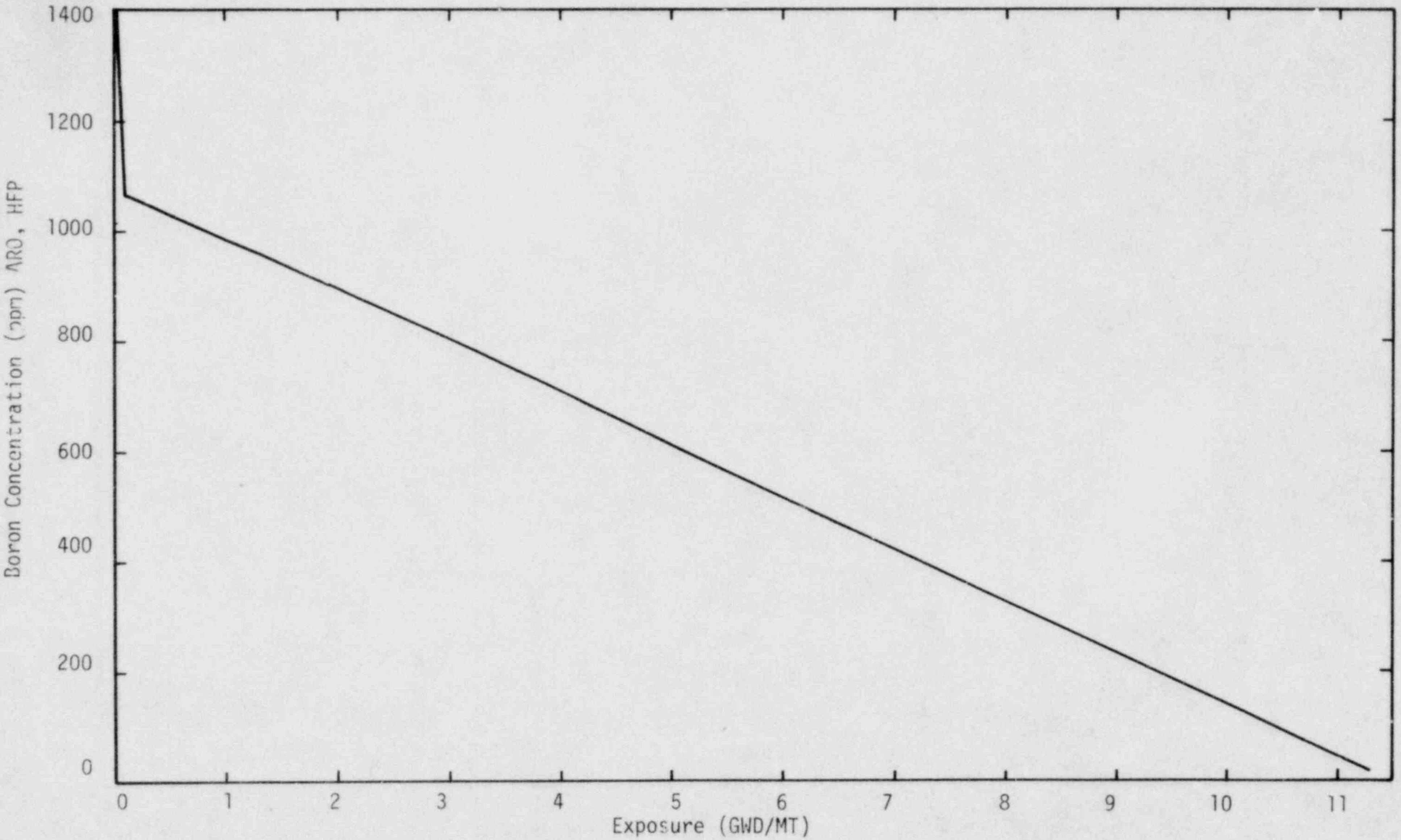

Figure 5.1 Prairie Island 1 Cycle 5 ARO, HFP, Critical Boron Vs. Exposure

7	8	9	10	11	12	13	
.887	1.173	1.059	1.219	.958	.956	.817	G
1.175	1.003	1.163	1.012	1.111	1.140 *	.670	H
1.063	1.167	1.061	1.214	.989	.980		I
1.223	1.015	1.217	1.028	1.095 *	.600		J
.961	1.114	.990	1.096 *	.773			K
.959	1.142 *	.982	.601				L
.819	.671						M

Assembly Powers

F_Q^N = 1.680 (J09)

Pin $F_{\Delta H}^N$ = 1.395 (J09)

F_Z = 1.213

*Gadolinia Bearing Assembly

Figure 5.2 Prairie Island #1 Cycle 5 Depletion
HFP, 0 MWD/MT, 1,396 ppm

	7	8	9	10	11	12	13
G	.929	1.141	1.049	1.170	.985	.987	.843
H	1.142	1.008	1.128	1.015	1.101	1.142 *	.706
I	1.050	1.130	1.047	1.165	1.002	.963	
J	1.171	1.016	1.166	1.033	1.111 *	.645	
K	.985	1.101	1.002	1.111 *	.794		
L	.988	1.142 *	.963	.645			
M	.843	.705					

Assembly Power

F_Q^N = 1.465 (H12)

Pin $F_{\Delta H}^N$ = 1.319 (J09)

F_Z = 1.097

*Gadolinia Bearing Assembly

Figure 5.3 Prairie Island #1, Cycle 5 Depletion
HFP, 11,300 MWD/MT, 10 ppm

6.0 THERMAL-HYDRAULIC DESIGN ANALYSIS

6.1 DESIGN CRITERIA

The thermal and hydraulic design performance requirements for the ENC reload fuel design are as follows:

1) The minimum departure from nucleate boiling ratio (MDNBR) will be $\geq$ 1.3 at overpower using the W-3 correlation with corrections for nonuniform axial heating and cold wall effects.
2) The fuel must be thermally and hydraulically compatible with the existing fuel and the Prairie Island 1 Reactor core throughout the life cycle of the fuel.
3) The maximum fuel temperature at overpower shall not exceed the fuel melting temperature.
4) The cladding upper temperature limits shall not exceed:

Inner surface temperature	850°F
Outer surface temperature	675°F
Average volumetric temperature	750°F

5.2 DESIGN ANALYSIS

The predicted steady state thermal-hydraulic performance of the ENC Prairie Island reload fuel satisfies all of the design criteria. The thermal-hydraulic analyses were performed at 112-percent of rated power (1650 MWt). The analyses in this section were based on a total power peaking factor F_Q of 2.32 which includes an engineering subfactor of 1.03. The results of the analysis, as well as the design conditions, are summarized in Table 6.1.

6.2.1 Hydraulic Compatibility

The hydraulic compatibility analyses of the ENC reload fuel and the existing fuel were performed at nominal reactor operating conditions and at the 112-percent overpower design conditions accounting for core inlet flow maldistribution. At nominal conditions the flow rate to each fuel type was within five percent of the core assembly average flow for a mixed core configuration, and at 112-percent overpower design conditions, the most limiting bundle flow was no less than 94-percent of the core average assembly flow.

The pressure loss coefficients used in the analyses are based on pressure drop tests performed by ENC and by Battelle-Pacific Northwest Laboratories using ENC and Westinghouse 14 x 14 test fuel assemblies. These test fuel assemblies were identical in all respects to production fuel except they were nonfueled.

The design difference between ENC and Westinghouse fuel assemblies, which have an impact on their thermal-hydraulic performance, are illustrated in the table below. The impact of the geometrical difference on the fuel's thermal-hydraulic performance is small and these geometric differences (as relevant to hydraulic performance) were present in the test fuel.

Design Factor	ENC Fuel	Westinghouse Fuel
Fuel pellet diameter (in.)	0.3565	0.3659
Clad ID (in.)	0.364	0.3734
Clad OD (in.)	0.424	0.422
Control rod OD (in.)	0.540	0.539
Bare rod flow area (in^2)	31.81	32.06
Wetted perimeter (in.)	266.9	265.7
Heated Perimeter (in.)	238.4	237.3

6.2.2 MDNBR Analysis

The MDNBR (minimum departure from nucleate boiling ratio) of Prairie Island fuel at 112-percent of rated power was calculated to be 1.97 for the ENC reload fuel and 2.05 for the Westinghouse fuel at a design total heat flux factor of 2.32 for both fuel types.

6.2.3 Fuel Temperature Analysis

The temperature analysis was performed with standard ENC calculational methods.[13] The analysis assumed the coincidence of maximum power peaking and the worst engineering tolerances that would maximize the resistance to heat transfer from the fuel rod to the coolant. The calculated maximum fuel and cladding temperatures were well below the design limits (Table 6.1) at 112-percent overpower.

6.3 EFFECT OF FUEL ROD BOW ON THERMAL HYDRAULIC PERFORMANCE

6.3.1 Rod Bow as Applied to DNBR Analysis

The calculation of the DNBR reduction as a result of rod bow considers both DNB tests with rod bow and the degree of bowing:

$$\Delta \text{MDNBR } (\%) = \sigma_{95} \times \delta_{bow}$$

where

δ_{bow} = bow to contact DNB penalty (from DNB tests)

σ_{95} = maximum anticipated fractional gap closure as a function of exposure.

The calculation of DNB rod bow to contact penalty is based on DNB tests with rod bow as referred to in the NRC's Interim Safety Evaluation Report on Effects of Fuel Rod Bowing on Thermal Margin Calculations for Light Water Reactors.

The maximum anticipated fractional gap closure is based on rod bow measurements of an ENC fuel similar to the Prairie Island design and currently being used in an operating reactor. Reference 1 presents the results of rod bow measurements taken on ENC reload fuel for the H. B. Robinson Reactor after two cycles of operation. The data base obtained from the above measurements include approximately 7000 independent measurements of rod-to-rod spacings for interior as well as peripheral rod bows. After two cycles of operation, the results indicate rod bow nowhere approaching bow to contact. Application of this data to the Prairie Island ENC reload design is in accordance with the aforementioned SER and includes a 1.2 multiplier to account for cold-to-hot variations in measured rod spacings. The maximum anticipated fractional gap closure through the fuel lifetime is:

$$\sigma_{95} = 0.4473$$

The fractional reduction in MDNBR was calculated for the three most limiting Condition II and III transients reported in XN-NF-78-35. At the time point where transient MDNBR occurred the following information was calculated.

Transient	Q" (Btu/hr-ft^2)	Pressure (psia)
Two pump coastdown	372,340	2,239
Control rod withdrawal	419,860	2,224
Loss of load	382,880	2,466

For conservatism, 60 psia was added to the pressure in calculation of δ_{bow}. The results of the analysis are given below and indicate the MDNBR values for the three most limiting transients are well above 1.3, a limiting 95/95 confidence of probability statement of DNB. Thus, no

reduction in allowable reactor peaking (F_Q = 2.32) is required as a result of a change in MDNBR due to rod bow.

Transients	MDNBR (with rod bow penalty)
Two pump coastdown	1.64
Control rod withdrawal	1.69
Loss of load	1.73

6.3.2 Effect of Rod Bow on LOCA Limits

The Prairie Island Plant Technical Specification incorporates a total nuclear peaking augmentation factor of 1.0815 in calculation of the ECCS safety limits which includes a 1.03 engineering subfactor and a 1.05 subfactor for measurement uncertainty. This factor is adequate to accommodate nuclear augmentation due to rod bow in a limiting assembly with burnup up to 28.15 x 10^3 MWD/MTU. Fuel assembly exposures in excess of this value are anticipated to be operating well below the LOCA limits due to the reduction of assembly reactivity. Therefore, no additional penalty due to rod bow needs to be applied to calculation of LOCA limits.

The combination of subfactors for the total nuclear peaking augmentation is accomplished as follows:

$$F_Q^U = 1.0 + \left(\delta_D^2 + \delta_E^2 + \delta_M^2 + \delta_B^2\right)^{\frac{1}{2}}$$

where

F_Q^U = Total nuclear peaking augmentation factor due to uncertainties

δ_D = Fractional augmentation due to densification

δ_E = Fractional augmentation due to engineering (pellet density, diameter, enrichment)

δ_M = Nuclear measurement and calculation uncertainty

δ_B = Nucl'ar augmentation due to rod bow.

The following table lists the bow augmentation factor as a function of burnup based upon ENC's measurement of rod bow of ENC-supplied reload fuel.(15)

Exposure Dependent Total Nuclear Peaking Augmentation Applied to Exxon Nuclear Fuel

Exposure Cycle	Region Average Expected EOC Burnup, 10^3 MWD/MTU	Fractional Augmentation Due to Bow	Total Nuclear Peaking Augmentation Factor
1	11.02	0.027	1.064
2	24.11	0.052	1.078
3	36.00	0.068	1.089

The total nuclear peaking augmentation factors as shown above are less than that specified in Plant Technical Specifications for Cycle 1 and Cycle 2. The region average exposure which correspond to 1.0815 augmentation was determined to be 28.5 x 10^3 KWD/MTU. It is expected that a fuel at burnup values equal or higher than this will not be the limiting fuel as discussed above. Therefore, no additional penalty due to rod bow needs to be applied to the calculation of LOCA limits.

6.4 EFFECT OF GADOLINIUM

The core loading for Cycle 5 will include 16 demonstration assemblies containing four gadolinia-bearing ($UO_2Gd_2O_3$) pins per assembly. The presence of Gd in the fuel rod will suppress the power of the fuel at beginning of cycle and tends to lessen severity of total power peaking at the

beginning of cycle. By mid-cycle (5,500 MWD/MT), the gadolinium burns out resulting in flattening the pin power and radial power. Since the current analysis considered the highest anticipated total peaking throughout the fuel lifetime, the result of the MDNBR analysis envelopes the effect of gadolinium.

TABLE 6.1

THERMAL-HYDRAULIC DESIGN CONDITIONS AND PERFORMANCE

Reactor Conditions	Design	Nominal
Core power (MWt)	1848	1650
Total reactor flow rate (Mlb/hr)	68.2	68.2
Active core flow rate (Mlb/hr)	65.1	65.1
Core coolant inlet temperature (°F)	534.5	530.5
Core pressure (psia)	2220.0	2250.0
Power Distribution		
Overall peaking (F_Q)		2.32
Radial x local		1.55
Axial		1.454
Engineering factor		1.03
Thermal-Hydraulic Performance	**W**	**ENC**
Hot assembly flow factor	1.016	0.940
MDNBR	2.05	1.97
Fuel center temperature (°F)	--	4399
Clad outer surface temperature (°F)	--	663.8
Clad inner surface temperature (°F)	--	789.9
Volumetric averaged clad temperature (°F)	--	726.5

7.0 ACCIDENT AND TRANSIENT ANALYSIS

7.1 PLANT TRANSIENT AND ECCS ANALYSES FOR PRAIRIE ISLAND 1

The plant transient and ECCS analyses are provided in References 2 and 3, respectively.

7.2 ROD EJECTION ANALYSIS FOR PRAIRIE ISLAND 1, CYCLE 5

A Control Rod Ejection Accident is defined as the mechanical failure of a control rod mechanism pressure housing, resulting in the ejection of a Rod Cluster Control Assembly (RCCA) and drive shaft. The consequence of this mechanical failure is a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage.

The rod ejection accident has been evaluated with the procedures developed in the ENC Generic Rod Ejection Analysis[4]. The ejected rod worths and hot pellet peaking factors were calculated using the XTG code. No credit was taken for the power flattening effects of Doppler or moderator feedback in the calculation of ejected rod worths or peaking factors. The calculations made for Cycle 5 using XTG were three-dimensional. The pellet energy deposition resulting from an ejected rod was evaluated explicitly for BOC and found to be 118 cal/gm at HFP and less than 20 cal/gm at HZP. The results were checked for EOC conditions by conservatively assuming that both the ejected rod worth and the power peaking were doubled. Even under these extreme conditions the energy deposition in the pellet was less than 130 cal/gm. The rod ejection accident was found to result in energy deposition of less than the 280 cal/gm stated in Regulatory Guide 1.77. The significant parameters for the analysis, along with the results are summarized in Table 7.1.

Table 7.1 Prairie Island Unit 1 Rod Ejection Accident, BOC

	HFP	HZP
F_Q^N After Ejection	1.90	3.71
Ejected Rod Worth (pcm)	213	379
Doppler Coefficient (pcm/^{o}F)	-1.30	-1.30
Delayed Neutron Fraction	.0060	.0060
Energy Deposition (cal/gm)	118	< 20

8.0 REFERENCES

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APPENDIX A

GADOLINIA DEMONSTRATION PROGRAM

A.1 INTRODUCTION

The reload for Cycle 5 of Prairie Island Unit 1 (Batch I-7) will consist of 40 assemblies with a U-235 enrichment of 3.4 w/o. Included in the reload will be sixty-four (64) fuel pins which contain 1 w/o Gd_2O_3. The gadolinia bearing pins will be uniformly distributed among sixteen (16) fuel assemblies. This core loading will demonstrate the application of a sufficient quantity of gadolinia bearing fuel rods in a pressurized water reactor reload to control a measurable quantity of core reactivity. The gadolinia demonstration program in Prairie Island will enhance the PWR data base obtained from the Palisades program and provide experience which will allow transition in the future from demonstration quantities of gadolinia to production quantities of gadolinia.

A.2 BACKGROUND

Gadolinia bearing fuel (UO_2 - Gd_2O_3) supplied by Exxon Nuclear Company (ENC) has undergone irradiation in BWR's for several years. In addition, Exxon Nuclear Company currently has test quantities of gadolinia bearing uranium fuel rods under irradiation in the Palisades Nuclear Plant.

A substantial number of Exxon Nuclear supplied BWR fuel assemblies containing gadolinia as a burnable poison have been irradiated to high burnups. The gadolinia is contained in several fuel rods in each assembly and is uniformly blended with the enriched UO_2.

Typical irradiated fuel assemblies have been examined during the reactor refueling outages. The examinations have included visual examinations, fuel rod diameter measurements, fuel rod length measurements, and gamma scan measurements. The fuel examinations performed to date, including fuel rods containing gadolinia, have revealed no abnormalities.

The gamma scan measurements have demonstrated the accuracy of the ENC calculational methods to predict the depletion of the gadolinia. A comparison of calculated and measured local power distributions for a BWR fuel assembly is shown on Figure A.1. Although the calculations have not been corrected for reactor flux tilts, the calculated powers in the gadolinia rods compare well with the measured powers.

In Palisades, there are a total of 32 rods distributed among eight assemblies with 1 w/o gadolinia in the UO_2 fuel pellets. These rods were loaded in the Palisades reactor at the start of the current operating cycle (Cycle 3). Measured power distributions show power differences in gadolinia bearing assemblies of only 1% to 2% variance from the ENC prediction. These assemblies will continue to be closely monitored and compared to ENC predictions throughout the cycle.

A.3 NEUTRONIC ANALYSIS

The neutronics calculations for the gadolinia bearing rods to be loaded in Prairie Island are based on standard Exxon Nuclear Company methods.[1,2,3] Modelling techniques developed for the gadolinia loading in Palisades are used. The UO_2 - Gd_2O_3 fuel cell cross sections are calculated with a multigroup transport theory code which includes the effect of the surrounding cells on the neutron energy spectrum. From this calculation,

transport corrected diffusion theory cross sections are developed for a discrete pin cell. These cross sections may then be input directly in a discrete mesh core model or alternately into single assembly calculations from which flux weighted cross sections are calculated for use in a nodal code.

A.3.1 Gadolinia Bearing Fuel Cell Cross Section

The gadolinia bearing fuel cell was depleted and cross sections generated with the XPIN[4] code. XPIN calculates infinite lattice parameters by multigroup transport theory. The UO_2 - Gd_2O_3 fuel cell and its surroundings are modelled with 20 spatial points; 10 in the pellet region, 1 in the cladding, 4 in the moderator region, and 5 in a homogenized ring representing the surrounding eight lattice cells. This "super cell" is depleted and the cross sections in the central cell collapsed to two groups. In Figure A.2, the infinite multiplication factor for a 3.4 w/o enriched pin cell containing 1 w/o gadolinia is compared as a function of exposure to a similar cell with no gadolinia.

Effective two group diffusion cross section are developed using a rectangular representation of the super cell. The super cell is modelled with a discrete mesh (1 mesh interval per pin pitch) diffusion theory calculation using PDQ.[5] The fast group cross sections are taken directly from the XPIN pin cell, while the thermal group absorption and fission cross sections are corrected until the diffusion theory reaction rates (fast and thermal) match those predicted with transport theory.

A.3.2 Gadolinia Bearing Fuel Assembly Calculation

Assembly calculations have been done to determine a desirable distribution of gadolinia bearing fuel rods within an assembly. A four

assembly, discrete mesh, diffusion theory PDQ representation was utilized. With this model, both the number and location of UO_2 - Gd_2O_3 pins can be studied in detail. The four assembly model also has the advantage, when compared to a single assembly, of studying the behavior of an assembly containing UO_2 - Gd_2O_3 when surrounded by various combinations of exposed fuel.

Based on the assembly calculations, and on core calculations to be discussed later, a preliminary assembly loading configuration has been determined. For the Prairie Island Unit 1 gadolinium demonstration program, it is recommended that in the gadolinium bearing fuel assemblies, four standard fuel rods be replaced with UO_2 rods containing 1 w/o gadolinia. On an assembly basis the worth of the gadolinia is predicted to be 3,630 pcm at beginning of life. At an assembly exposure of 4,000 MWD/MT, the poison worth has diminished to about 500 pcm and has become indistinquishable by an assembly exposure of 8,000 MWD/MT.

The effect of the UO_2 - Gd_2O_3 pins on assembly local peaking was studied for a variety of environments in the surrounding assemblies. Included were fresh fuel assemblies with and without UO_2 - Gd_2O_3 pins, once burned assemblies, and twice burned assemblies. From the analysis it is concluded that at the beginning of life the presence of four UO_2 - Gd_2O_3 pins will increase the local peak by about 3% when compared with an assembly containing only fresh UO_2 pins. As the gadolinium burns out the pin power distribution is expected to flatten. Thus as the average assembly power increases from poison depletion, the radial local diminishes. The pin by

pin power distribution for an assembly with and without UO_2 - Gd_2O_3 pins is shown in Figure A.3. In the illustration, an infinite array of assemblies at beginning of life has been modelled.

A.3.3 Core Analysis

The Cycle 5 reference core design has been studied with the 3-D reactor simulator XTG[6]. With this model each assembly is represented by four radial and twelve axial nodes. As a result of core design studies, the recommended UO_2 - Gd_2O_3 loading in Cycle 5 is sixteen assemblies with four pins containing 1 w/o gadolinia per assembly. With this loading the gadolinia poison is predicted to be equivalent to about 65 ppm of soluble boron at beginning of cycle. By mid cycle (5,500 MWD/MT) the gadolinia poison effect has disappeared.

The Cycle 5 fuel loading pattern has been designed such that even if the gadolinia poison worth is signficantly smaller or significantly larger than expected, a desirable core power distribution will be achieved. The effects of off-nominal gadolinia poison worths have been stuied by varying the poison worth while maintaing a fixed loading pattern. That is, the fresh, once, and twice burned fuel assembly locations were not changed in the analysis but remained as optimized for the nominal gadolinia poison worth.

Three cases were considered and the core depletion characteristics studied. The base case is the predicted core behavior if the gadolinia worth and burnout is accurately calculated. The off-nominal extremes of poison worth were bracketed by setting the beginning of cycle (BOC) poison worth to zero (only UO_2 fuel) for one case and for the other extreme, the

BOC gadolinia worth was assumed to be 40% more than predicted. These postulated core configurations were depleted and the resulting power distribution compared.

As expected, there is noticeable differences in the BOL relative power distribution. However, as shown in Figure A.4, the power shape associated with either extreme is acceptable. By a quarter of the way through the cycle (2,500 MWD/MT) the power shapes for all three cases have begun to converge (see Figure A.5). Furthermore, as illustrated in Figure A.6, the relative power distributions for all three cases become essentially the same by mid cycle (5,500 MWD/MT).

These analyses confirm that even in the event that the actual gadolinia poison worth varies from the predicted worth, the safe, full power operation of the plant will not be compromised.

APPENDIX A REFERENCE

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6. R. B. Stout, "XTG-A Two-Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing, "XN-CC-28, Rev. 3, Exxon Nuclear Company, August, 1975.

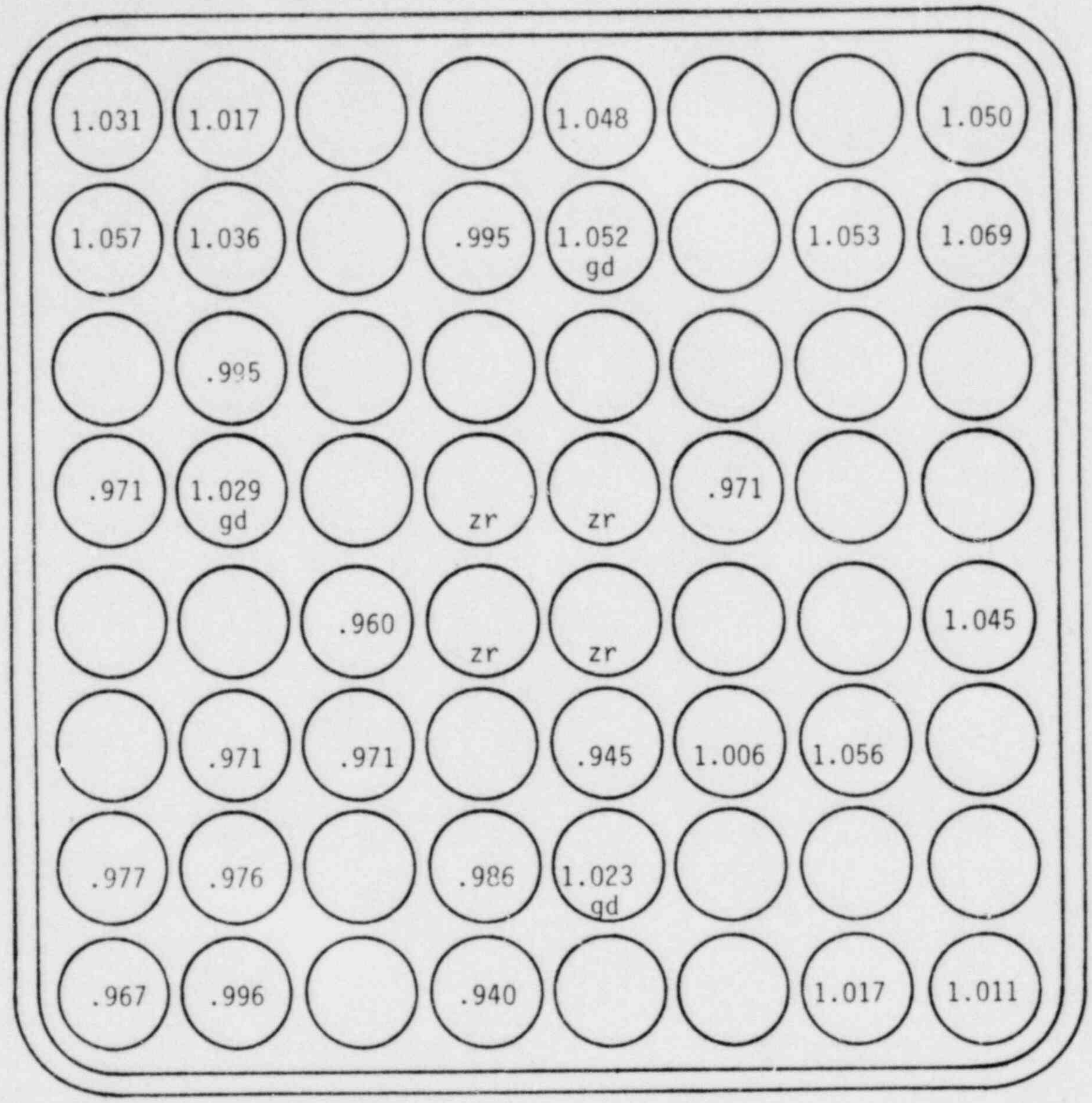

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Figure A.1 Ratio of Calculated to Measured Local Power Distribution
Oyster Creek Lead Fuel Assembly - 3,800 MWD/MTU

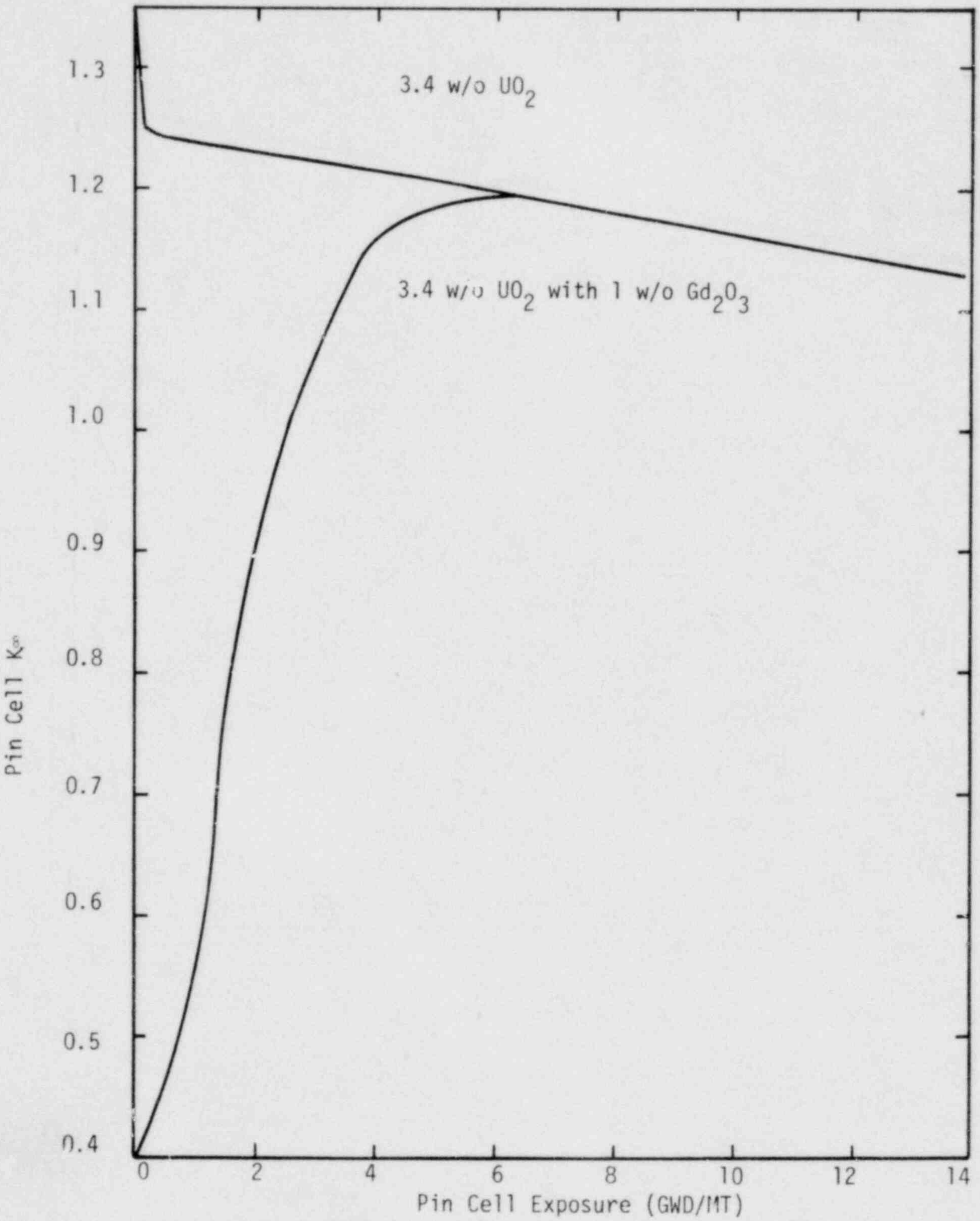

Figure A.2 Pin Cell Infinite Multiplication Factor Vs. Exposure

.94 .95	.95 .94	.96 .91	.96 .89	.96 .92	.96 .96	.96 .97	.96 .97	.96 .96	.96 .92	.96 .89	.96 .92	.95 .94	.94 .95
	.97 .97	1.00 .95	* .99 .44	1.00 .95	1.01 1.01	1.00 1.01	1.00 1.01	1.01 1.01	1.00 .95	* .99 .44	1.00 .95	.97 .97	.95 .96
		GT	1.03 .99	1.04 1.02	GT	1.03 1.05	1.03 1.05	GT	1.04 1.02	1.03 .99	GT	1.00 1.01	.96 .97
			1.05 1.06	1.06 1.08	1.06 1.08	1.02 1.05	1.02 1.05	1.06 1.08	1.07 1.08	1.05 1.06	1.03 1.05	.99 1.02	.96 .98
				GT	1.04 1.07	1.00 1.04	1.01 1.04	1.05 1.08	GT	1.07 1.10	1.04 1.07	1.00 1.03	.96 .99
					1.01 1.05	1.01 1.05	1.03 1.07	1.04 1.08	1.05 1.08	1.06 1.10	GT	1.01 1.05	.96 1.00
						1.01 1.06	IT	1.03 1.07	1.01 1.05	1.02 1.06	1.03 1.07	1.00 1.04	.96 1.00
						Without Gadolinia → With Gadolinia →	1.01 1.06	1.01 1.05	1.01 1.04	1.02 1.06	1.03 1.07	1.00 1.04	.96 1.00
								1.01 1.05	1.04 1.07	1.06 1.09	GT	1.01 1.05	.96 1.00
									GT	1.06 1.09	1.04 1.07	1.00 1.03	.96 .99
										1.05 1.06	1.03 1.05	.99 1.02	.96 .98
											GT	1.00 1.00	.96 .96
												.97 .97	.95 .96
													.94 .95

* UO_2-1 w/o Gd_2O_3

GT Guide Tube

IT Instrument Tube

Figure A.3 Relative Pin Power Prairie Island Unit 1
14 x 14 BOC, HFP

	7	8	9	10	11	12	13
G	.908 .943 .833	1.186 1.228 1.095	1.071 1.103 1.001	1.223 1.248 1.171	.964 .969 .955	.954 .941 .981	.810 .800 .830
H	1.188 1.230 1.091	1.017 1.052 .943	1.172 1.204 1.102	1.021 1.039 .982	1.111 1.111 1.109	1.128 1.090 1.211 *	.665 .653 .688
I	1.074 1.107 1.005	1.175 1.208 1.105	1.070 1.092 1.022	1.214 1.225 1.190	.989 .984 1.002	.970 .957 .997	
J	1.227 1.251 1.174	1.024 1.042 .985	1.217 1.228 1.192	1.028 1.020 1.045	1.086 1.044 1.177*	.599 .587 .623	
K	.967 .971 .957	1.113 1.114 1.112	.990 .984 1.002	1.087 1.045 1.178*	.766 .746 .811		
L	.956 .943 .983	1.130 1.091 1.213*	.971 .958 .998	.600 .588 .624			
M	.812 .802 .832	.666 .655 .689	← Reference Design - Nominal Gadolinia Worth ← Gadolinia Worth = 1.4 x Predicted ← No Gadolinia				

	F_Q^N	$F_{\Delta H}^N$
Reference Design	1.63 $\pm$ .08	1.40 $\pm$ .07
1.4 x Predicted	1.66 $\pm$ .08	1.42 $\pm$ .07
No Gadolinia	1.70 $\pm$ .09	1.40 $\pm$ .07

* Gadolinia Bearing Assemblies

Figure A.4 Assembly Relative Power Distribution
Prairie Island Unit 1 Cycle 5
BOC, HFP, ARO Equilibrium Xenon

	7	8	9	10	11	12	13	
	.875 .880 .867	1.131 1.138 1.123	1.032 1.038 1.025	1.187 1.192 1.179	.965 .967 .962	.974 .973 .976	.826 .826 .825	G
	1.133 1.139 1.124	.979 .984 .971	1.128 1.134 1.120	1.002 1.006 .996	1.109 1.110 1.107	1.172 1.162 1.186*	.683 .682 .684	H
	1.035 1.041 1.027	1.131 1.137 1.123	1.042 1.046 1.035	1.192 1.194 1.187	.999 .998 .999	.982 .981 .983		I
	1.189 1.194 1.182	1.004 1.008 .998	1.194 1.197 1.189	1.037 1.036 1.039	1.136 1.124 1.154*	.622 .620 .625		J
	.967 .969 .964	1.111 1.113 1.108	.999 .999 .999	1.136 1.125 1.155*	.793 .789 .799			K
	.975 .974 .977	1.173 1.164 1.188*	.983 .981 .984	.623 .621 .625				L
	.826 .826 .826	.684 .683 .685	← Reference Design - Nominal Gadolinia Worth ← Gadolinia Worth = 1.4 x Predicted ← No Gadolinia					M

	F_Q^N	$F_{\Delta H}^N$
Reference Design	1.58 ± .08	1.36 ± .07
1.4 x Predicted	1.58 ± .08	1.36 ± .07
No Gadolinia	1.58 ± .08	1.37 ± .07

* Gadolinia Bearing Assemblies

Figure A.5 Assembly Relative Power Distribution
Prairie Island Unit 1 Cycle 5
2,500 MWD/MT, HFP, ARO
Equilibrium Xenon

	7	8	9	10	11	12	13
G	.880 .875 .891	1.120 1.114 1.133	1.026 1.02[illegible] 1.035	1.171 1.168 1.177	.971 .971 .970	.985 .987 .980	.841 .843 .836
H	1.121 1.115 1.134	.976 .972 .987	1.116 1.112 1.125	.999 .998 1.003	1.105 1.106 1.104	1.175 1.177 1.170*	.699 .701 .695
I	1.028 1.024 1.038	1.118 1.114 1.127	1.034 1.032 1.039	1.176 1.175 1.177	1.001 1.002 .998	.981 .984 .976	
J	1.173 1.170 1.17[illegible]	1.001 .999 1.004	1.177 1.177 1.179	1.037 1.038 1.034	1.141 1.143 1.135*	.635 .637 .631	
K	.971 .972 .971	1.106 1.107 1.105	1.001 1.002 .998	1.141 1.143 1.135*	.802 .805 .797		
L	.985 .987 .981	1.175 1.178 1.170*	.98[illegible] .984 .976	.635 .637 .631			
M	.841 .844 .836	.699 .701 .695					

← Reference Design - Nominal Gadolinia Worth
← Gadolinia Worth = 1.4 x Predicted
← No Gadolinia

	F_Q^N	$F_{\Delta H}^N$
Reference Design	1.54 ± .08	1.36 ± .07
1.4 Predicted	1.55 ∓ .08	1.36 ∓ .07
No Gadolinia	1.53 ∓ .08	1.35 ∓ .07

* Gadolinia Bearing Assemblies

Figure A.6 Assembly Relative Power Distribution
Prairie Island Unit 1 Cycle 5
5,500 MWD/MT HFP, ARO
Equilibrium Xenon