50-244

ONLY COPY

original

XN-NF-79-103

R.E. GINNA NUCLEAR PLANT CYCLE 10 SAFETY ANALYSIS REPORT WITH MIXED OXIDE ASSEMBLIES

DECEMBER 1979

RICHLAND, WA 99352

EXXON NUCLEAR COMPANY, Inc.

12/20/79

8912280 240

XN-NF-79-103
Issue Date: 12/14/79

R. E. GINNA NUCLEAR PLANT CYCLE 10
SAFETY ANALYSIS REPORT
WITH MIXED OXIDE ASSEMBLIES

Drafted by: B. L. Johnson

Prepared: G. J. Busselman, Manager
Neutronics and Fuel Management

Approved: 12/14/79
G. A. Sofer, Manager
Nuclear Fuels Engineering

Concurred: 12/14/79
J. N. Morgan, Manager
Licensing and Safety Engineering

Concurred: 12/14/79
L. J. Federico, Manager
Nuclear Fuels Project

EXXON NUCLEAR COMPANY, Inc.

NUCLEAR REGULATORY COMMISSION DISCLAIMER

IMPORTANT NOTICE REGARDING CONTENTS AND USE OF THIS DOCUMENT

PLEASE READ CAREFULLY

This technical report was derived through research and development programs sponsored by Exxon Nuclear Company, Inc. It is being submitted by Exxon Nuclear to the USNRC as part of a technical contribution to facilitate safety analyses by licensees of the USNRC which utilize Exxon Nuclear-fabricated reload fuel or other technical services provided by Exxon Nuclear for light water power reactors and it is true and correct to the best of Exxon Nuclear's knowledge, information, and belief. The information contained herein may be used by the USNRC in its review of this report, and by licensees or applicants before the USNRC which are customers of Exxon Nuclear in their demonstration of compliance with the USNRC's regulations.

Without derogating from the foregoing, neither Exxon Nuclear nor any person acting on its behalf:

A. Makes any warranty, express or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this document, or that the use of any information, apparatus, method, or process disclosed in this document will not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of, any information, apparatus, method, or process disclosed in this document.

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

R. E. GINNA NUCLEAR PLANT CYCLE 10
SAFETY ANALYSIS REPORT

1.0 INTRODUCTION AND SUMMARY

The R. E. Ginna Nuclear plant will operate in Cycle 10 beginning in early 1980 with three regions of fuel supplied by Exxon Nuclear Company (ENC). The loading will consist of 32 ENC assemblies in Region 12 and 4 Westinghouse mix oxide (MOX) assemblies. The remainder of the core contains 40 once-burnt and 32 twice-burnt ENC assemblies and 13 exposed Westinghouse supplied assemblies.

The characteristics of the fuel and of the reloaded core result in conformance with existing Technical Specification limits regarding shutdown margin provisions and thermal limits. This document provides the neutronic analysis for the plant during Cycle 10 operation and the control rod ejection analysis. The ENC fuel design[1] is unchanged from the fuel design used in the Cycle 8 and 9 ENC fuel reloads. The previous Plant Transient Analysis[2] remains valid for Cycle 10. The ECCS analysis[3] is applicable to Cycle 10 operation. The consequences of the rod ejection accident for Cycle 10 are slightly less severe than those calculated for Cycles 8[4] and 9[5]. The introduction of the 4 MOX assemblies into the reactor core leads to small changes in the core average kinetic parameters resulting in minimal effects to the previous analyses performed for Cycles 8[1,2,4] and 9[1,3,5].

2.0 OPERATING HISTORY OF THE REFERENCE CYCLE

R. E. Ginna Cycle 9 has been chosen as the reference cycle with respect to Cycle 10 due to the close resemblance of the neutronic characteristics between these two cycles. The Cycle 9 operation began on April 3, 1979, and as of November 31, 1979 the core had accrued about 6,714 MWD/MT. The Cycle 9 loading included 40 fresh ENC fuel assemblies with 32 exposed ENC assemblies and 49 exposed Westinghouse assemblies.

The measured power peaking factors at hot-full-power, equilibrium xenon conditions, have remained considerably below the Technical Specification limits throughout Cycle 9. The total nuclear peaking factors, F_Q^N, and the radial nuclear pin peaking factor, $F_{\Delta H}^N$, have remained below 1.75 and 1.45, respectively. Cycle 9 operation has typically been rod free with the D control bank positioned in the range of 218 to 222 steps, 225 steps being fully withdrawn. It is anticipated that similar control bank insertions will be seen in Cycle 10.

The critical boron concentration as calculated by ENC for Cycle 9 has agreed to within about 8 ppm compared to the observed values (see Figure 2.1). Also the predicted power distributions have typically agreed to within ±3 percent of the measured values (see Figure 2.2 for typical comparison).

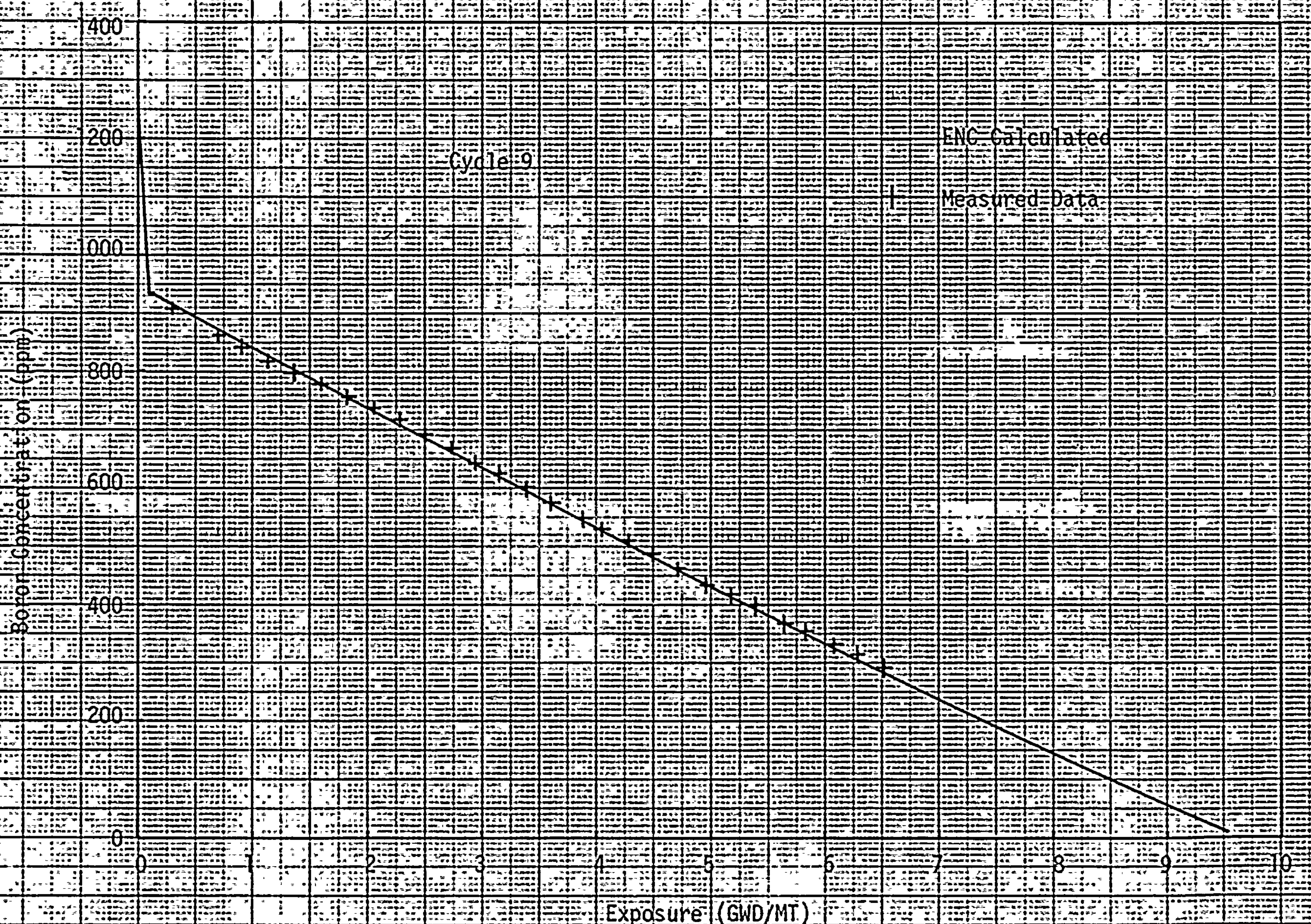

Figure 2.1 R. E. Ginna Cycle 9 Critical Boron Curve, Predicted vs. Measured

	G	F	E	D	C	B	A
7	.968 .995 -2.71	1.110 1.129 -1.68	.920 .938 -1.92	.949 .965 -1.66	1.189 1.178 .93	.996 .963 3.43	.794 .782 1.53
8	1.108 1.131 -2.03	.986 1.013 -2.67	1.029 1.048 -1.81	1.099 1.103 -.36	1.189 1.168 1.80	1.074 1.051 2.19	.663 .645 2.79
9	.918 .947 -3.06	1.030 1.051 -2.00	.996 1.006 -.99	1.203 1.191 1.01	1.079 1.055 2.27	.977 .974 .31	
10	.953 .968 -1.55	1.098 1.105 -.63	1.191 1.192 -.08	1.036 1.021 1.47	1.178 1.149 2.52	.714 .711 .42	
11	1.195 1.179 1.36	1.188 1.169 1.63	1.072 1.055 1.61	1.173 1.149 2.09	.804 .798 .75		
12	.985 .963 2.28	1.071 1.051 1.90	.977 .975 .21	.715 .712 .42			
13	.772 .782 -1.28	.651 .645 .93					

Measured Assembly Power
Calculated (XTGPWR)
$\frac{m-c}{c} \times 100$

	Calculated	Measured	%Difference
F_Q^N	1.528	1.564	2.33
$F_{\Delta H}$	1.351	1.337	-1.07
F_z	1.105	1.154	4.42

Figure 2.2 R. E. Ginna Power Distribution Comparison To Map IX-24, HFP, 5,505 MWD/MT

3.0 GENERAL DESCRIPTION

The R. E. Ginna 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. The RCC guide tubes and the instrumentation tube are made of SS-304L. Each ENC assembly contains nine zircaloy spacers with Inconel springs; eight of the spacers are located within the active fuel region. Four of the 121 assemblies contain Mixed Oxide (PuO_2 plus UO_2) bearing fuel rods. The MOX assemblies consist of three enrichment zones of PuO_2 utilizing natural UO_2 as the diluent.

The projected Cycle 10 loading pattern is shown in Figure 3.1 with the assemblies identified by their Fabrication ID's and Region ID's. The initial enrichments of the various regions are listed in Table 3.1. BOC10 exposures, based on an EOC9 exposure of 9,570 MWD/MT, along with Region ID's are shown in Figure 3.2. The core consists of 32 fresh ENC assemblies at 3.45 w/o and 4 fresh Westinghouse MOX assemblies loaded on the periphery with 72 ENC and 13 Westinghouse exposed assemblies scatter-loaded in the center portion of the core. Pertinent fuel assembly parameters for the Cycle 10 fuel are depicted in Table 3.1. The transuranic elements, including Am-241, have been accounted for up to the time of the anticipated reactor startup.

Table 3.1 R. E. Ginna Cycle 10 Fuel Assembly Design Parameters

	Region				
	9	10	11	12	MOX
Enrichment, wt% U-235	3.103	3.100	3.200	3.450	2.626*
Number of Assemblies	13	32	40	32	4
Pellet Density, % TD	95.0	94.0	94.0	94.0	95.0
Pellet-to-Clad Diametrical Gap, Mil	7.5	7.5	7.5	7.5	7.5
Fuel Stack Height, inch	141.4	142.0	142.0	142.0	141.4
Region Average Burnup at BOC10, MWD/MT	24,339	17,885	8,335	0	0
Nominal Assembly Weight, KgU	392.56	373.78	373.78	373.78	395.91**

* wt% Pu (based on assembly average)

** in Kg HM

	M	L	K	j	I	H	G	F	E	D	C	B	A
1						12	MOX	12					
2				12	12	L14	12	L31	12	12			
3			12	M09	L09	M14	K03	M39	L01	M01	12		
4		12	M02	L06	M17	L19	M28	L26	M36	L05	M12	12	
5		12	L02	M33	L21	M23	K05	M30	L24	M20	L12	12	
6	12	L32	M40	L27	M31	K13	M07	K19	M22	L18	M13	L13	12
7	MOX	12	K09	M25	K20	M08	K28	M06	K18	M27	K27	12	MOX
8	12	L15	M15	L20	M24	K26	M05	K17	M29	L25	M38	L30	12
9		12	L10	M18	L22	M32	K25	M21	L23	M35	L04	12	
10		12	M10	L07	M34	L28	M26	L17	M19	L08	M04	12	
11			12	M03	L03	M37	K14	M16	L11	M11	12		
12				12	12	L29	12	L16	12	12			
13						12	MOX	12					

Fabrication or New Fuel Region Identification

Figure 3.1 R. E. Ginna Cycle 10 Loading Pattern

G	F	E	D	C	B	A	
24,736 9	7,517 11	24,117 9	7,809 11	24,708 9	0 12	0 MOX	7
7,522 11	24,093 9	11,341 11	17,578 10	9,544 11	19,816 10	0 12	8
24,117 9	11,342 11	17,506 10	6,926 11	16,061 10	0 12		9
7,809 11	17,574 10	6,928 11	18,677 10	6,193 11	0 12		10
24,708 9	9,549 11	16,061 10	6,198 11	0 12			11
0 12	19,809 10	0 12	0 12				12
0 MOX	0 12						13

BOC10 Exposure MWD/MT

Region ID*

*See Table 3.1 for Region definitions

Figure 3.2 R. E. Ginna BOC10 Quarter Core Exposure Distribution and Region ID

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 10 core are quite similar to those of the Cycle 9 core (see Section 5.1).

The nuclear design bases for the Cycle 10 core are as follows:

1) The design shall permit operation within the Technical Specifications for the R. E. Ginna plant.
2) The length of Cycle 10 shall be determined on the basis of an assumed Cycle 9 length of 9,570 MWD/MT.
3) The Cycle 10 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 2.32 and the peak $F_{\Delta H}$ shall not exceed 1.66 (including uncertainties) in any single fuel rod through the cycle under nominal full power operation conditions.
   b) The scram worth of all rods minus the most reactive shall exceed BOC and EOC shutdown requirements.
4) The Cycle 10 core shall have a negative power coefficient.
5) The MOX assemblies shall be located in a region of the reactor core as to minimize the effects on shutdown margin provisions and thermal limits.

The neutronic design methods utilized to ensure the above requirements are consistent with those described in References 6, 7, and 8.

5.1 PHYSICS CHARACTERISTICS

The neutronic characteristics of the Cycle 10 core are compared with those of Cycle 9 and are presented in Table 5.1. The data presented in the table indicate the neutronic similarity between Cycles 9 and 10. The Cycle 10 loading pattern is applicable for Cycle 9 lengths of +700 MWD/MT and -800 MWD/MT about the nominal length of 9,570 MWD/MT.

The calculated boron letdown curve for Cycle 10 is shown in Figure 5.1. The curve indicates a BOC10, no xenon, critical boron concentration of 1,254 ppm. At 150 MWD/MT, equilibrium xenon, the critical boron concentration is 921 ppm. The Cycle 10 length is projected to be 9,500±300 MWD/MT with 7 ppm of boron at EOC.

5.1.1 Power Distribution Considerations

Representative predicted power maps for Cycle 10 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 10 loading pattern the calculated BOC nuclear power peaking factors, F_Q^N, $F_{\Delta H}^N$, and F_Z^N, are 1.745, 1.433, and 1.201, respectively. At EOC conditions the corresponding values are 1.517, 1.358, and 1.098. The Technical Specification limits relative to F_Q^N and $F_{\Delta H}^N$, with the measurement uncertainties backed out, are 2.15 and 1.60. Additionally the predicted axial F_Q^N distributions are well below the axially dependent Technical Specification limits on F_Q. The BOC F_Q^N value of 1.745 compares with the measured Cycle 9 value in Table 5.1 of 1.758.

The control of the core power distribution is accomplished by following the procedures as discussed in the report, XN-76-40, "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors", September 1976 and its addendum. The results reported in these documents demonstrate that the Power Distribution Control (PDC) procedures defined in the report will protect an axially dependent F_Q limit with a peak value of 2.30. The Technical Specification limit for R. E. Ginna has a peak of 2.32 and an axial dependence identical to that supported by the procedures. The physics characteristics of the Ginna Cycle 10 core are similar to those utilized in the PDC supporting analysis. The Ginna Technical Specification limits on F_Q can therefore be protected by operation under the PDC procedures as stated in XN-76-40.

5.1.2 Control Rod Reactivity Requirements

Detailed calculations of shutdown margins for Cycle 10 are compared with Cycle 9 data in Table 5.2. The ENC Plant Transient Simulation (PTS) Analysis indicates that the minimum required shutdown margin is 1,800 pcm based upon the steamline break accident analyzed for ENC fuel at the EOC conditions. A value of 1,900 pcm is used at EOC in the evaluation of the shutdown margin to be consistent with the Technical Specifications. The Cycle 10 analysis indicates excess shutdown margins of 1,414 pcm at the BOC and 344 pcm at the EOC. The Cycle 9 analysis[5] indicates excess shutdown margins for that cycle of 1,795 pcm at the BOC and 393 pcm at the EOC. The slightly lower Cycle 10 excess shutdown margins, when compared to the Cycle 9 values, are due to slightly lower calculated rod worths.

The control rod groups and insertion limits for Cycle 10 will remain unchanged from Cycle 9. With these limits the nominal worth of the control bank, D-bank, inserted to the insertion limits at HFP is 122 pcm at BOC and 170 pcm at EOC. The control rod shutdown requirements in Table 5.2 allow for a HFP D-bank insertion equivalent to 300 pcm for both BOC and EOC.

5.1.3 Moderator Temperature Coefficient Considerations

The reference Cycle 10 design calculations indicate that the moderator temperature coefficient is negative at all times during the cycle as shown in Table 5.1. This meets the Technical Specification requirement that the moderator temperature coefficient be negative at all times during power operation and the design criteria that the power coefficient be negative. The least negative moderator temperature coefficient occurs at BOC HZP and is -2.0 ± 2pcm/^{o}F. This compares with the BOC9 HZP value of -2.0 pcm/^{o}F.

5.2 ANALYTICAL METHODOLOGY

The methods used in the Cycle 10 core analyses are described in References 6, 7, and 8. These methods have been verified for both UO_2 and PuO_2-UO_2 lattices. In summary, the reference neutronic design analysis of the reload core was performed using the XTG (Reference 9) reactor simulator system. The input exposure data were based on quarter core depletion calculations performed from Cycle 5 to Cycle 9 using the XTG code. The BOC5 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.

Table 5.1 R. E. Ginna Neutronics Characteristics of Cycle 10 Compared with Cycle 9 Data

	Cycle 9		Cycle 10	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, Equilibrium Xenon (ppm)	961[1]	12[1] ppm	921	7
HZP, ARO No Xenon (ppm)	1,410[2]	---	1,414	--
Moderator Temperature Coefficient				
HFP, (pcm/°F)	-7.6	-30.4	-8.1	-30.4
HZP, (pcm/°F)	-2.0[2]	-21.5	-2.0	-21.6
Doppler Coefficient, (pcm/°F)	-1.25 to	-2.0[4]	-1.35	-1.84
Boron Worth, (pcm/ppm)				
HFP	-8.12	-8.72	-7.95	-8.62
HZP	-8.58	---	---	---
Total Nuclear Peaking Factor				
F_Q^N, HFP	1.758[3]	---	1.745	1.517
Delayed Neutron Fraction	.0061	.0051	.0058	.0052
Control Rod Worth of All Rods In Minus Most Reactive Rod, HZP, (pcm)	5,751[4]	5,821[4]	5,341	5,696
Excess Shutdown Margin (pcm)	1,795[4]	393[4]	1,414	344
Moderator Pressure Coefficient (pcm/psi)	----	0.35	----	0.35

(1) Extrapolated from measured data
(2) Measured Data
(3) 70% Power Map
(4) Reference 5

Table 5.2 R. E. Ginna Control Rod Shutdown Margins and Requirements for Cycle 10

	Cycle 9 **		Cycle 10	
	BOC	EOC	BOC	EOC
Control Rod Worth (HZP), pcm				
All Rods Inserted (ARI)	6,407	6,634	5,949	6,420
ARI less most reactive (N-1)	5,751	5,821	5,341	5,696
N-1 less 10% allowance [(N-1)*.9]	5,176	5,239	4,807	5,126
Reactivity Insertion, pcm				
Moderator plus Doppler	1,431	1,996	1,443	1,932
Flux Redistribution	600	600	600	600
Void	50	50	50	50
Sum of the above three	2,081	2,646	2,093	2,582
Rod Insertion Allowance	300	300	300	300
Total Requirements	2,381	2,946	2,393	2,882
Shutdown Margin (N-1)*.9 - Total Requirements	2,795	2,293	2,414	2,244
Required Shutdown Margin*	1,000	1,900	1,000	1,900
Excess Shutdown Margin	1,795	393	1,414	344

* Technical Specification 3.10
** Calculated values from Reference 5

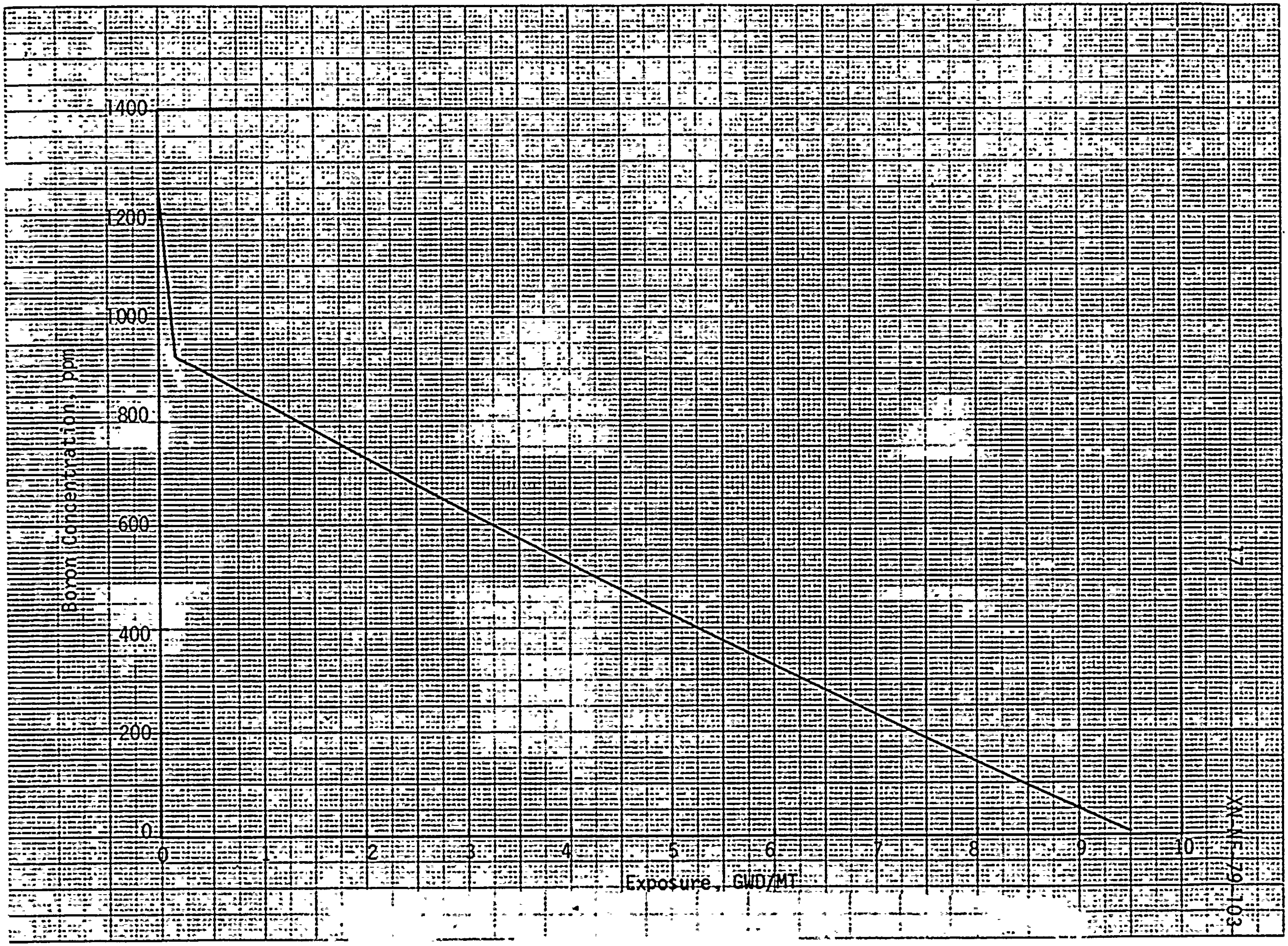

Figure 5.1 R. E. Ginna Cycle 10 ARO Critical Boron Concentration vs. Exposure

G	F	E	D	C	B	A	
.971	1.160	.987	1.206	.978	1.258	.795	7
1.160	.978	1.141	1.077	1.135	.856	.632	8
.987	1.142	1.089	1.249	1.031	.988		9
1.207	1.078	1.248	1.027	1.052	.712		10
.978	1.135	1.030	1.050	.788			11
1.258	.856	.987	.711	Assembly Power			12
.795	.631						13

F_Q^N = 1.745 (B07)

Pin $F_{\Delta H}^N$ = 1.433 (B07)

F_z = 1.201

Figure 5.2 R. E. Ginna Cycle 10 Power Distribution, HFP, 0 MWD/MT, 1,254 ppm

G	F	E	D	C	B	A	
1.017	1.161	1.022	1.177	1.000	1.192	.767	7
1.160	1.019	1.134	1.076	1.120	.887	.663	8
1.022	1.135	1.083	1.197	1.033	.984		9
1.177	1.076	1.197	1.029	1.042	.737		10
1.001	1.120	1.033	1.040	.803			11
1.192	.887	.983	.737	Assembly Power			12
.767	.663						13

F_Q^N = 1.517 (G12)

Pin $F_{\Delta H}^N$ = 1.358 (G12)

F_z = 1.098

Figure 5.3 R. E. Ginna Cycle 10 Power Distribution HFP, 9,500 MWD/MT, 7 ppm

6.0 THERMAL HYDRAULIC DESIGN

The thermal and hydraulic considerations in the Region 12 design are unchanged from those presented in Reference 4 for Region 10 fuel.

7.0 ACCIDENT AND TRANSIENT ANALYSIS

7.1 PLANT TRANSIENT AND ECCS ANALYSES FOR R. E. GINNA

The ECCS analysis provided in Reference 3 is applicable to all ENC fuel residing in the core during Cycle 10 operation.

The Plant Transient Analysis reported in XN-NF-77-40(2) for the R. E. Ginna plant was intended to cover all anticipated ranges of values for all significant fuel dependent plant parameters for Cycle 8 and for all future reloads. Table 7.1 presents a comparison of the kinetic parameters used in the Plant Transient Analysis and the parameters calculated specifically for Cycle 10. Due to the introduction of the 4 MOX assemblies the reactivity worth of the boric acid used by the HPSIS (High Pressure Safety Injection System) and the BOC delayed neutron fraction have been calculated to be outside the range reported in the XN-NF-77-40 analysis. The analysis was reviewed and it was found that the change in boric acid worth affects the small and large steamline break transients and that the delayed neutron fraction most affects the fast uncontrolled rod withdrawal transient.

The enveloping data for both steamline breaks are the EOC data and for the fast uncontrolled rod withdrawal are BOC data. The impact of the Cycle 10 parameters (see Table 7.1) have been evaluated for each of the transients. The results of the evaluation for the transients were found to be nearly equivalent to the previous results and that the figure of merit for the transients were not violated, i.e. for the small steamline break the system does not go critical, for the large steamline break the MDNBR is greater than the 1.30 limit and for the uncontrolled rod withdrawal the MDNBR margin is not altered.

7.2 ROD EJECTION ANALYSIS FOR R. E. GINNA CYCLE 10

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 damage.

The rod ejection accident analysis presented in the document XN-NF-78-53[4] is still applicable to Cycle 10 operation. The location of the 4 MOX assemblies introduces minimal effects on ejected rod worths and hot pellet peaking factors. The ejected rod worths and hot pellet peaking factors are 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 10 using XTG were two-dimensional (x-y) with appropriate axial buckling correction terms. The total peaking factors (F_Q^N) were determined as the product of the radial peaking factor (as calculated using XTG) and a conservative axial peaking factor. The pellet energy deposition resulting from an ejected rod was evaluated to be less than the results reported in References 4 and 5. The rod ejection accident was found to result in energy deposition of less than 280 cal/gm stated in Regulatory Guide 1.77 and provides a greater energy deposition margin than that determined by Reference 4. The results of the control rod ejection transient for this case are presented in Table 7.2 along with results from References 4 and 5.

7.1 R. E. Ginna Kinetic Parameters

Parameters	Reference Cycle[1] BOC	Reference Cycle[1] EOC	Cycle 10 BOC	Cycle 10 EOC
Moderator Temperature Coefficient (pcm/°F)	0.0	-35.0	-8.1	-30.4
Moderator Pressure Coefficient (pcm/psia)	+.25	+.35	+.09	+.35
Moderator Density Coefficient (pcm/gm/cm^3)	0.0	+29635.0	+6858.0	+25740.0
Doppler Coefficient (pcm/°F)	-1.25	-2.00	-1.35	-1.84
Boron Worth Coefficient (pcm/ppm)	-8.75	-8.72	-7.95	-8.62
Delayed Neutron Fraction	.0061	.0051	.0058	.0052

[1] Reference 2

Table 7.2 Ejected Rod Worth and Peaking Factors

	Cycle 8[2]		Cycle 9[3]		Cycle 10[4]	
	HFP	HZP	HFP	HZP	HFP	HZP
F_Q^N Before Ejection	2.25	2.82	2.24	2.62	2.15[1]	2.59[1]
F_Q^N After Ejection	4.36	5.30	2.96	5.59	2.84[1]	5.01[1]
Maximum Rod Worth from a Full Inserted Bank (% $\Delta\rho$)	0.470	0.640	0.362	0.553	0.280	0.435
Energy Deposition (cal/gm)	171	37				

(1) Includes a conservative estimate of F_z at HFP of 1.4 and at HZP of 1.8.

(2) Reference 4, calculated with XTRAN.

(3) Reference 5, calculated with XTGPWR.

(4) Calculated with XTGPWR.

8.0 REFERENCES

1. XN-NF-77-52, "R. E. Ginna Reload Fuel Design", November, 1977.

2. XN-NF-77-40, "Plant Transient Analysis for R. E. Ginna, Unit 1 Nuclear Power Plant", Revision 1, July, 1979.

3. XN-NF-77-58, "ECCS Analysis for the R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model", December, 1977.

4. XN-NF-77-53, "R. E. Ginna Nuclear Plant Cycle 8 Safety Analysis Report", December, 1977.

5. XN-NF-78-50, "R. E. Ginna Cycle 9 Safety Analysis Report," December, 1978.

6. F. B. Skogen, "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors", XN-75-27(A), Exxon Nuclear Company, April, 1977.

7. XN-75-27(A), Supplement 1 to Reference 6, April, 1977.

8. XN-75-27, Supplement 2 to Reference 6, December, 1977.

9. XN-CC-28, Rev. 3, "XTG: A Two Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing (PWR Version)", January, 1975.

R. E. GINNA CYCLE 10 RELOAD
SAFETY ANALYSIS REPORT
WITH MIXED OXIDE ASSEMBLIES

DISTRIBUTION

K. H. Blank
G. J. Busselman
L. J. Federico
R. L. Feuerbacher
R. G. Grummer
B. L. Johnson (2)
M. R. Killgore
T. L. Krysinski
C. E. Leach
J. N. Morgan
W. S. Nechodom
L. A. Nielsen
G. F. Owsley
J. F. Patterson
A. W. Prichard
F. B. Skogen
G. A. Sofer
A. V. Wojchouski
C. H. Wu
RG&E/L. J. Federico (80)
Document Control (10)