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H. B. ROBINSON UNIT 2, CYCLE 10
SAFETY ANALYSIS REPORT

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EXXON NUCLEAR COMPANY, INC.

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H. B. ROBINSON UNIT 2, CYCLE 10

SAFETY ANALYSIS REPORT

Written by: I.Z. Stone, Nuclear Engineer

Reviewed by: MRK 9 JUL 84
F. B. Skogen, Manager
PWR Neutronics

Prepared by: July 11 1984
H. E. Williamson, Manager
Neutronics and Fuel Management

Prepared by: 10 JULY 84
R. B. Stout, Manager
Licensing and Safety Engineering

Approved by: 7/11/84
G. J. Busselman, Manager
Fuel Design

Approved by: 11 JUL 84
G. A. Sofer, Manager
Fuel Engineering and Technical Services

Concurred by: for GNWard 12 July 84
J. N. Morgan, Manager
Proposals and Customer Services Engineering

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H.B. ROBINSON UNIT 2, CYCLE 10

SAFETY ANALYSIS REPORT

1.0 INTRODUCTION

The results of the safety analysis for Cycle 10 for the H.B. Robinson Unit 2 nuclear plant are presented in this report. The Cycle 10 analysis reflects plant operation at 2,300 MWt. The topics addressed herein include operating history of the reference cycle, power distribution considerations, control rod reactivity requirements, temperature coefficient considerations, and the control rod ejection analysis.

The Cycle 10 design requires the loading of sixty-five (65) fresh Exxon Nuclear Company (ENC) supplied fuel assemblies; forth-four (44) XN-7 assemblies, nine (9) XN-6 assemblies and twelve (12) Partial Length Shield Assemblies (PLSAs). The forty-four (44) fresh XN-7 fuel assemblies utilize natural uranium axial blankets (NUABs) with thirty-six (36) of the assemblies also containing gadolinia-bearing fuel pins. One of the XN-6 assemblies also contain gadolinia-bearing fuel pins. The twelve (12) PLSAs will reduce the fast neutron fluence reaching the pressure vessel wall.

2.0 SUMMARY

Cycle 10 of the H.B. Robinson Unit 2 nuclear plant is designed to operate at 2,300 MWt in Cycle 10 beginning in the fall of 1984. The characteristics of the fuel and of the reload core result in conformance with required shutdown margins and thermal limits. This document provides the neutronic analysis for the plant during Cycle 10 operation.

The ENC fuel mechanical design is presented in Reference 1. The thermal-hydraulic analyses are provided in Reference 14. The Plant Transient Analyses and the ECCS/LOCA Analyses are presented in References 15 and 16, respectively. The generic Control Rod Ejection Analysis is provided in Reference 17. These analyses are all applicable to the Cycle 10 operating conditions at 2,300 MWt.

H.B. Robinson Cycle 8 has been chosen as the reference neutronics cycle due to the close resemblance of the overall neutronic characteristics of Cycle 10 to Cycle 8.

The results contained in this report show that operation at a power level of 2,300 MWt can be safely achieved within the licensed limits.

3.0 OPERATING HISTORY OF THE REFERENCE CYCLE

H. B. Robinson Cycle 8 has been chosen as the reference neutronics cycle due to the close resemblance of the overall neutronic characteristics of Cycle 10 to Cycle 8.

The Cycle 8 plant operation at nominal core average temperature, and power (578°F, 2,300 MWt) started in October 1980 and continued until November 1981. In early December 1981, operation at a reduced core average temperature and power was initiated to improve steam generator tube performance. Cycle 8 ended with an accumulated exposure of 10,383 MWD/MT; 2,300 MWD/MT of which was achieved at reduced temperature and power.

The measured power peaking factors remained below the Technical Specification limits for Cycle 8. The total nuclear peaking factor, F^N_Q, and the radial nuclear pin peaking factor, $F^N_{\Delta H}$, remained below 1.96 and 1.49 respectively. Cycle 8 operation was typically rod free with Control Bank D positioned in the range of 200 to 220 steps; 228 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 8 has agreed well when compared to the observed values (see Figure 3.1). A power distribution calculated with the PDQ model is compared to measured values shown in Figure 3.2. The comparison is made at a Cycle 8 exposure of 7,083 MWD/MT for a core power of 96% of 2,300 MWt.

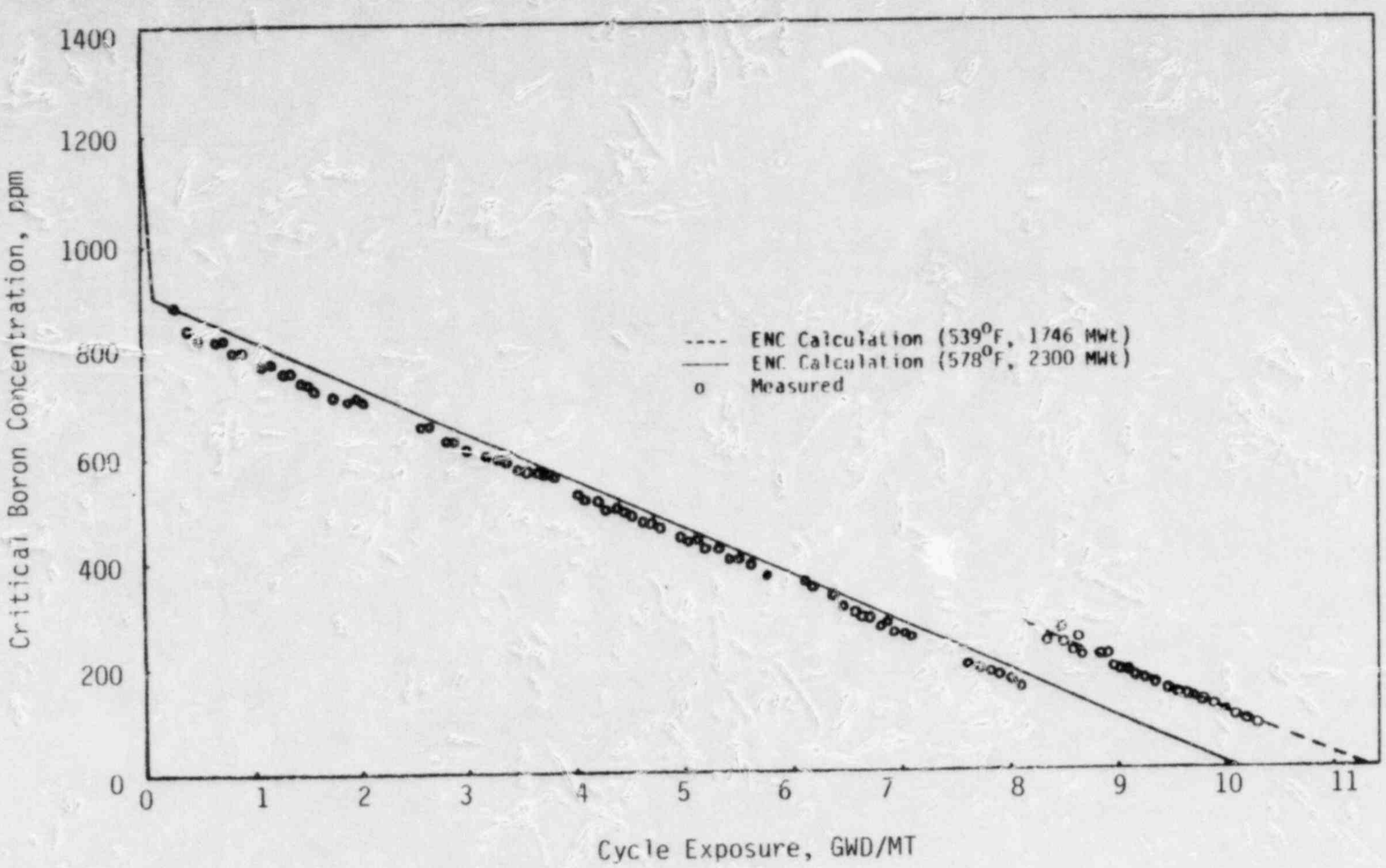

Figure 3.1 H.B. Robinson Unit 2, Critical Boron Concentration versus Exposure for Cycle 8 at Both Nominal and Reduced Temperature Conditions

	H	G	F	E	D	C	B	A
8	0.750 0.730 2.7	1.115 1.085 2.8	1.013 0.982 3.2	0.982 0.961 2.2	1.168 1.163 0.4	1.139 1.136 0.3	0.935 0.947 -1.3	0.804 0.828 -2.9
9	1.111 1.087 2.2	0.996 0.975 2.2	1.184 1.159 2.2	1.033 1.016 1.7	1.180 1.175 0.4	0.975 0.978 -0.3	1.141 1.161 -1.7	0.668 0.689 -3.0
10	1.003 0.984 1.9	1.182 1.161 1.8	1.006 0.992 1.4	1.182 1.173 0.8	1.001 1.001 0.0	1.081 1.089 -0.7	0.971 0.983 -1.2	
11	0.972 0.963 0.9	1.027 1.016 1.1	1.184 1.174 0.9	1.118 1.104 1.3	0.970 0.971 -0.1	1.127 1.138 -1.0	0.698 0.708 -1.4	
12	1.148 1.163 -1.3	1.168 1.174 -0.5	1.011 1.001 1.0	0.977 0.971 0.6	0.940 0.941 -0.1	0.731 0.738 -0.9		
13	1.099 1.136 -3.3	0.962 0.976 -1.4	1.092 1.088 0.4	1.137 1.136 0.1	0.733 0.737 -0.5			
14	0.921 0.948 -2.8	1.131 1.161 -2.6	0.974 0.982 -0.8	0.705 0.708 -0.4				
15	0.806 0.829 -2.8	0.670 0.689 -2.8						

Measured
PDQ
$\frac{M - P}{P} \times 100$

	Measured	Calculated
$F^N_{\Delta H}$	1.30	1.33
F^N_Q	1.45	1.45

Figure 3.2 H.B.Robinson Unit 2, Cycle 8 PDQ Calculated to Measured Power Distribution Comparison, 96% of 2,300 MWt 7,083 MWD/MT for D-Bank at 215 Steps

4.0 GENERAL DESCRIPTION

The H.B. Robinson reactor consists of 157 assemblies, each having a 15x15 fuel rod array. Each assembly contains 204 fuel rods, twenty RCC guide tubes, and one instrumentation tube. 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 fuel rods consist of slightly enriched UO_2 pellets inserted into zircaloy tubes.

The Cycle 10 design reflects the loading of sixty-five (65) fresh ENC supplied fuel assemblies. This core design contains the first reload of axially blanketed fuel in H. B. Robinson Unit 2, XN-7, and twelve (12) special fuel assemblies which will reduce the fast neutron fluence reaching the pressure vessel wall. The latter are denoted as Partial Length Shield Assemblies (PLSAs). This core design is the second H. B. Robinson reload fuel design utilizing 4 w/o gadolinia. Thirty-six (36) of the forty-four (44) fresh Region 13 (XN-7) fuel assemblies and one of the nine (9) fresh Region 12 (XN-6) assemblies contain gadolinia-bearing pins.

The design for Batch XN-7, Region 13, includes natural uranium axial blankets in the top and bottom six (6) inches of the active fuel region. The batch average enrichment for the blanketed assemblies is 3.08 w/o U-235. This average enrichment is achieved by using a central axial zone enrichment of 3.34 w/o in fuel pins which contain no gadolinia and 2.37 w/o in the gadolinia-bearing fuel pins. In twelve (12) Region 13 assemblies, twelve (12) fuel pins per assembly will contain 4 w/o gadolinia. In another twenty-four (24) assemblies, eight (8) pins of 4 w/o gadolinia per assembly will be utilized. In the remaining eight (8) Region 13 (XN-7) fuel assemblies no gadolinia pins will be used. In addition to the forty-four (44) blanketed assemblies loaded, nine (9) fresh XN-6 assemblies containing 2.85 w/o enrichment will be used in Cycle 10. One fresh (1) XN-6 assembly will contain twelve (12) pins of 4 w/o gadolinia and will be loaded in the center core location. An enrichment of 2.20 w/o U-235 is used in the XN-6 gadolinia-bearing fuel pins. Thus, the total number of gadolinia pins required for Cycle 10 is 348.

Twelve (12) PLSAs are being loaded on the core periphery (the "flats") as part of the program to reduce the fast neutron fluence to the pressure vessel wall. In the active fuel region, the top six (6) inches contain natural uranium, the next ninety-six (96) inches contain uranium enriched to 1.24 w/o, and the bottom forty-two (42) inches contain 304 stainless steel.

The projected Cycle 10 loading pattern is shown in Figure 4.1 with the assemblies identified by assembly fabrication ID and by their core location in the previous cycle or by fresh fuel region. BOC10 exposures, based on an EOC9 exposure of 10,637 MWD/MT, along with Region ID's, are shown in Figure 4.2. The initial enrichments of the various regions are listed in Table 4.1. Also included in Table 4.1 are the peak assembly exposures by region and fuel type.

Table 4.1 H.B. Robinson Unit 2 Cycle 10, Fuel Assembly Design Parameters

	Region										
	PLSA	11	12	12	12*	12**	12***	12***	13	13**	13***
XN Number	XN-7 PLSA	5	6	6	6	6	6	6	7	7	7
Number of Assemblies	12	48	16	8	12	8	8	1	8	24	12
Pellet Density, %TD	94.0	94.0	94.0	94.0	94.0	94.0	94.0	94.0	94.0	94.0	94.0
Pellet to Clad Diametral Gap, Mil	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
Initial Enrichment (w/o U-235)											
Upper 6 inches											
UO_2	.71	2.90	2.85	2.85	2.85	2.85	2.85	2.85	.71	.71	.71
UO_2-Gd_2O_3	----	----	----	----	2.20	2.20	2.20	2.20	----	----	----
Central 132 inches											
UO_2	1.24+	2.90	2.85	2.85	2.85	2.85	2.85	2.85	3.34	3.34	3.34
UO_2-Gd_2O_3	----	----	----	----	2.20	2.20	2.20	2.20	----	2.37	2.37
Lower 6 inches											
UO_2	304SSTL	2.90	2.85	2.85	2.85	2.85	2.85	2.85	.71	.71	.71
UO_2-Gd_2O_3	----	----	----	----	2.20	2.20	2.20	2.20	----	----	----
Average	0.86	2.90	2.85	2.85	2.84	2.82	2.81	2.81	3.12	3.09	3.02
Initial Gd_2O_3, w/o	----	----	----	----	4	4	4	4	----	4	4
Batch Average Burnup at BOC10, MWD/MT	0	21,534	8,348	0	10,899	11,043	12,646	0	0	0	0
Peak Assembly Burnup at EOC10, MWD/MT	3,705	34,705	21,003	10,465	22,820	22,975	25,349	13,936	7,909	12,099	13,756

* Fuel with 4 pins of 4 w/o gadolinia per assembly
** Fuel with 8 pins of 4 w/o gadolinia per assembly
*** Fuel with 12 pins of 4 w/o gadolinia per assembly
\+ Lower 36 inches of central 132 inches contains 304SSTL
PLSA Part Length Shield Assembly

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							PLSA N45	PLSA N49	PLSA N53						
2					13 N01	12 M18	** N21	** N25	** N29	12 M19	13 N02				
3				F13 L52	** N41	L3 L10	L2 M01	H7 L02	E2 M02	E3 L13	** N42	K13 L50			
4			C10 L31	** N37	M3 M05	J4 L04	K2 M41	H14 M49	F2 M42	G4 L09	D3 M06	** N38	N10 L28		
5		13 N05	** N33	N4 M09	F9 L33	*** N09	L4 L06	*** N17	E4 L07	*** N10	K9 L32	C4 M10	** N34	13 N06	
6		12 M22	N5 L14	M7 L16	*** N13	G10 L48	G14 M36	F5 L41	J14 M35	J10 L45	*** N14	D7 L19	C5 L15	12 M23	
7	PLSA N56	** N32	P5 M13	P6 M46	M5 L23	B9 M40	D12 M32	H5 M17	M12 M31	P9 M38	D5 L24	B6 M51	B5 M14	** N22	PLSA N46
8	PLSA N52	** N28	J8 L26	B8 M48	*** N20	L10 L18	L8 M26	+ M53	E8 M27	E10 L17	*** N18	P8 M50	G8 L27	** N26	PLSA N50
9	PLSA N48	** N24	P11 M16	P10 M45++	M11 L29	B7 M39	D4 M30	H11 M28	M4 M29	P7 M37	D11 L30	B10 M52	B11 M15	** N30	PLSA N54
10		12 M25	N11 L38	M9 L34	*** N16	G6 L08	G2 M34	F11 L11	J2 M33	J6 L05	*** N15	D9 L37	C11 L39	12 M24	
11		13 N08	** N36	N12 M12	F7 L21	*** N12	L12 L46	*** N19	E12 L47++	*** N11	K7 L20	C12 M11	** N35	13 N07	
12			C6 L25	** N40	M13 M08	J12 L44	K14 M44	H2 M47	F14 M43	G12 L49	D13 M07	** N39	N6 L22		
13				F3 L03	** N44	L13 L40	L14 M04	H9 L51	E14 M03	E13 L43	** N43	K3 L01			
14					13 N04	12 M21	** N31	** N27	** N23	12 M20	13 N03				
15							PLSA N55	PLSA N51	PLSA N47						

** 8 pins of 4 w/o gadolinia per assembly in Region 12 fuel

*** 12 pins of 4 w/o gadolinia per assembly in Region 13 fuel

PLSA Part Length Shield Assembly

\+ 12 pins of 4 w/o gadolinia per assembly in Region 12 fuel

++ This assembly contains one (1) inert zirconium rod

Number of gadolinia pins per assembly
Cycle 9 core location or region number
Assembly fabrication ID

Figure 4.1 H.B. Robinson Unit 2 Cycle 10, Reference Loading Pattern for an EOC9 Exposure of 10,637 MWD/MT

	H	G	F	E	D	C	B	A
8	0 13,936 12***	13,186 25,349 12	24,048 34,507 11	0 13,745 13***	11,179 24,140 12	21,094 32,093 11	0 12,089 13**	0 3,705 PLSA
9	13,155 25,324 12	12,121 24,292 12	11,072 22,975 12	24,067 34,698 11	10,522 22,820 12	7,909 20,549 12	0 11,792 13**	0 3,083 PLSA
10	24,045 34,510 11	11,014 22,924 12	22,612 33,404 11	0 13,602 13***	20,446 31,277 11	19,555 30,102 11	0 10,457 12	
11	13,756 13***	24,070 34,705 11	0 13,609 13***	22,634 33,492 11	8,791 20,990 12	0 11,832 13**	0 7,903 13	
12	11,636 24,086 12	10,498 22,815 12	20,406 31,256 11	8,790 21,003 12	0 11,730 13**	19,984 25,654 11		
13	21,077 32,092 11	7,902 20,557 12	19,538 30,099 11	0 11,842 13**	19,962 25,637 11			
14	0 12,099 13**	0 11,802 13**	0 10,465 12	0 7,909 13	BOC10 Exposure, MWD/MT EOC10 Exposure, MWD/MT Region ID			
15	0 3,707 PLSA	0 3,040 PLSA						

** Fuel with 8 pins of 4 w/o gadolinia per assembly
*** Fuel with 12 pins of 4 w/o gadolinia per assembly
PLSA Part Length Shield Assembly

Figure 4.2 H.B. Robinson Unit 2, BOC10 and EOC10 Quarter Core Exposure Distribution and Region ID for EOC9 = 10,637 MWD/MT

5.0 MECHANICAL DESIGN

The 44 XN-7 Reload fuel assemblies are mechanically identical to the previous reload assemblies, with the exception of the fuel rods. The 144 inch fuel column includes a six (6) inch column of natural UO_2 pellets at each end. The natural UO_2 pellets have a length-to-diameter ratio of 1.5 and a total dish volume of 0.7% as compared to 0.8 and 1.0%, respectively, for the enriched pellets. In addition, the fuel column insulator discs are no longer used.

The 12 Partial Length Shield Assemblies (PLSAs) are mechanically identical to the Reload XN-7 assemblies, with the exception of replacing the bottom 42 inches of the fuel column with inserts of type 304 stainless steel rod used for neutron shielding. A supplemental mechanical design analysis for the PLSAs is provided in Reference 3.

A description of the basic Exxon Nuclear supplied fuel design and design methods is contained in Reference 1. In addition, mechanical design analysis of the Reload XN-7 fuel and the resident XN-5 and XN-6 fuel with current methodology is contained in Reference 2. Several of the XN-5 assemblies are expected to exceed the original design burnup of 33,000 MWD/MT. The reference analysis encompasses the expected conditions, such that the relevant mechanical criteria will not be exceeded.

Due to the low power of the PLSA fuel, the design conditions are generally enveloped by previous design analyses. However, the differences in physical properties of the stainless steel relative to uranium dioxide required confirmation of certain aspects of the design.

- o The thermal and mechanical behavior of the fuel rod was evaluated for thermal expansion effects of the stainless steel. The fuel rod response was determined to be bound by the higher power fuel rods.
- o The loss in assembly holddown margin resulting from the lower weight fuel rods was determined to be within the capability of the holddown springs to prevent hydraulic liftoff under normal operating conditions.
- o The fuel assembly was determined to be no more sensitive to irradiation-induced bowing than previous design.
- o The seismic analysis was determined to be valid for the lower weight of the assemblies.

One (1) once burnt and one (1) twice burnt assembly containing one inert rod each will be loaded into the core for Cycle 10. The inert rods were designed to be mechanically compatible with the fuel assembly.

6.0 NUCLEAR CORE DESIGN

The H. B. Robinson Unit 2, Cycle 10, Region 13 reload design has been developed in accordance with the following requirements:

1. The Cycle 10 reload shall contain sixty-five (65) new fuel assemblies; nine (9) XN-6 (Region 12) assemblies, forty-four (44) XN-7 (Region 13) asemblies, and twelve (12) Part Length Shield Assemblies.
2. The length of Cycle 10 shall be maximized.
3. The rated power for Cycle 10 shall be 2,300 MWt.
4. The length of Cycle 10 shall be determined based on an actual EOC9 exposure of 10,637 MWD/MT.
5. Cycle 10 Operation is anticipated to be base loaded; however,the reload fuel shall be designed to accommodate load following operation between 50% and 100% of rated power while not precluding the current ramp and step change bases as set forth in the FSAR.
6. In accordance with plant Technical Specifications, the control rod worth requirements shall be met.
7. The loading pattern shall be designed to produce acceptable power distributions. The design F^T_Q, including uncertainties, shall be less than 2.32 at 2,300 MWt. The integrated peak to average pin power, $F_{\Delta H}$, including measurement uncertainties, shall be less than 1.65 at 2,300 MWt.

8. The loading pattern shall be designed to accommodate the PLSAs used to reduce the neutron fluence at the pressure vessel.

The neutronic design methods utilized in the analyses are consistent with those described in References 4 through 7.

6.1 PHYSICS CHARACTERISTICS

The neutronic characteristics of the Cycle 10 core are compared to those of Cycle 8 in Table 6.1. The data presented in the table indicates the neutronic similarity between Cycles 8 and 10. The reactivity coefficients of the Cycle 10 core are bounded by the coefficients used in the safety analysis. The safety analysis for Cycle 10 is based on physics characteristics representative of those expected for a Cycle 9 length of 10,637 MWD/MT.

The boron letdown curve for Cycle 10 operation at 2,300 MWt is shown in Figure 6.1. As shown, the BOC10 no xenon, hot full power (HFP) critical boron concentration is predicted to be 1,002 ppm. At 100 MWD/MT, equilibrium xenon, the critical boron concentration at HFP is 713 ppm. The Cycle 10 length is projected to be 10,820 MWD/MT (312 EFPDs) with no boron at EOC at a power level of 2,300 MWt.

6.1.1 Power Distribution Considerations

At a power level of 2,300 MWt at equilibrium xenon conditions, 100 MWD/MT, the calculated XTG peak $F_{\Delta H}$ is 1.48 including a 4% measurement uncertainty. At the same exposure the peak F^T_Q is 2.18 including a 3% engineering factor, a 5% measurement uncertainty, K(Z) consideraions, and an 11% allowance for operation with PDC-II and $\pm$5% target bands.

The peak $F_{\Delta H}$ and F_Q^T in Cycle 10 occur at a cycle exposure of 5,000 MWD/MT. The predicted value of $F_{\Delta H}$ at this exposure is 1.56 including the 4% measurement uncertainty. The peak F^T_Q, again including V(Z) and K(Z) considerations and appropriate uncertainties, is 2.18.

The quarter-core radial power distributions are presented in Figures 6.2 through 6.4 for Cycle 10 exposures of 100 MWD/MT, 5,000 MWD/MT, and (EOC) 10,820 MWD/MT, respectively.

With the normalized axial dependence factor K(Z) included, the expected peak power, F^T_Q, versus core axial elevation has been evaluated. All F^T_Q values are within Technical Specification limits. The limiting case is found to be at 5,000 MWD/MT near the core mid-plane with F^T_Q being about 6% below the allowable unit for operation with target bands of $\pm$ 5% at 2,300 MWt.

For Cycle 10 (as in Cycle 9), the total allowable power peaking factor of 2.32 at 2,300 MWt includes the local power peaking due to fuel densification and uncertainties related to engineering factor, measurement, and analysis. This power peaking limit assures that the linear power density remains below the limiting values of the fuel rod linear power density, thus meeting the LOCA and the overpower limit criteria.

6.1.2 Control Rod Reactivity Requirements

Detailed calculations of shutdown margins for Cycle 10 are compared with Cycle 8 data in Table 6.2. A value of 1,770 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 1,911 pcm at the BOC and 461 pcm at the EOC. The Cycle 8 analysis indicated excess shutdown margins for that cycle of 1,554 pcm at the BOC and 565 at the EOC. The Cycle 10 excess shutdown margins are seen to be similar to the Cycle 8 values.

The control rod groups and insertion limits for Cycle 10 will remain unchanged from Cycle 8. The control rod shutdown requirements in Table 6.2 allow for a HFP D Bank insertion equivalent to 600 pcm for BOC and 400 pcm for EOC, to bound the Cycle 10 control rod worths.

6.1.3 Isothermal Temperature Coefficient Considerations

The Cycle 10 isothermal temperature coefficients are shown in Table 6.1 for HFP and HZP conditions at both BOC and EOC. At BOC10, following a nominal EOC9 shutdown exposure of 10,637 MWd/MT, the HFP isothermal temperature coefficient is projected to be -5.1 pcm/^{o}F at a critical boron concentration of 1,002 ppm. The corresponding HZP critical boron concentration is 1,134 ppm with the isothermal temperature coefficient being -0.7 pcm/^{o}F.

The Technical Specification for the moderator temperature coefficient for Cycle 10 allows for a +5.0 pcm/^{o}F (Reference 15) at HZP conditions. With a calculated value of + 1.0 pcm/^{o}F at HZP, the Technical Specifications at this condition is expected to be met, with no control rod insertion anticipated for ascension to HFP conditions. Similarly, at HFP conditions, the calculated moderator temperature coefficient for HFP, shown in Table 6.1 for no xenon, is well below the reqirement of 0 pcm/^{o}F.

6.2 POWER DISTRIBUTION CONTROL PROCEDURES

The control of the core power distribution is accomplished by following the procedures for "Exxon Nuclear Power Distribution Control for Pressurized Water Reactors, Phase II"(8,9,10). These procedures, denoted PDC-II, have been generically approved by the NRC for application to Westinghouse type PWRs.

6.3 ANALYTICAL METHODOLOGY

The methods used in the Cycle 10 core analyses are described in References 4 through 7. In summary, the reference neutronic design analysis of the reload core was performed using the XTG(11) reactor simulator system. The fuel shuffling between cycles was accounted for in the calculations. The PDQ/HARMONY(12,13) code package will be utilized to monitor the power distribution.

Calculated values of F_Q, F_{xy}, and $F_{\Delta H}$ were studied with the twenty-four (24) axial node XTG reactor model. The thermal-hydraulic feedback and axial exposure distribution effects of power shapes, rod worths, and cycle lifetime are explicitly included in the analysis.

Table 6.1 H. B. Robinson Unit 2, Neutronics Characteristics of Cycle 10 Compared with Cycle 8 Data

	Cycle 8		Cycle 10	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, (ppm)	1,217	20	1,002	0
HZP, ARO, No Xenon (ppm)	1,305	--	1,134	--
Moderator Temp. Coefficient				
HFP, (pcm/°F)	-1.25	-29.5	-3.8	-31.3
HZP, (pcm/°F)	+3.7	--	+1.0	-21.0
Isothermal Temp. Coefficient				
HFP, (pcm°F)	-2.5	-30.9	-5.1	-32.8
HZP, (pcm°F)	+2.0	--	-0.7	-22.8
Doppler Coefficient (pcm/°F)	-1.2	-1.4	-1.3	-1.5
Power Defect (Moderator + Doppler), pcm	1,359	1,904	1,532	2,030
Boron Worth, (ppm/10^3 pcm)				
HFP	-106	-96	-107	-96
HZP	---	---	-105	-92
Promp Neutron Lifetime (μsec)	24.8	24.0	25.7	24.2
Delayed Neutron Fraction	0.0060	0.0053	0.0061	0.0053
Control Rod Worth of All Rods In Minus Most Reactive Rod, HZP, (pcm)	5,626	5,765	6,326	5,901
Excess Shutdown Margin, (pcm)	1,554	565	1,911	461

Table 6.2 H.B.Robinson Unit 2, Control Rod Shutdown Margin and Requirements for Cycle 10

	(pcm)			
	BOC 8	EOC 8	BOC 10	EOC 10
Control Rod Worth				
ARI	6,926	7,065	7,626	7,201
N-1	5,626	5,765	6,326	5,901
(N-1)* .9	5,063	5,189	5,693	5,311
Reactivity Insertion				
Power Defect (Moderator + Doppler)	1,359	1,904	1,532	2,030
Full Power D Bank Insertion	500	300	600	400
Flux Redistribution	600	600	600	600
Void	50	50	50	50
Total Requirements	2,509	2,854	2,782	3,080
Shutdown Margin				
(N-1)*.9 - Total Requirements	2,554	2,335	2,911	2,231
Required shutdown Margin	1,000	1,770	1,000	1,770
Excess Shutdown Margin	1,554	565	1,911	461

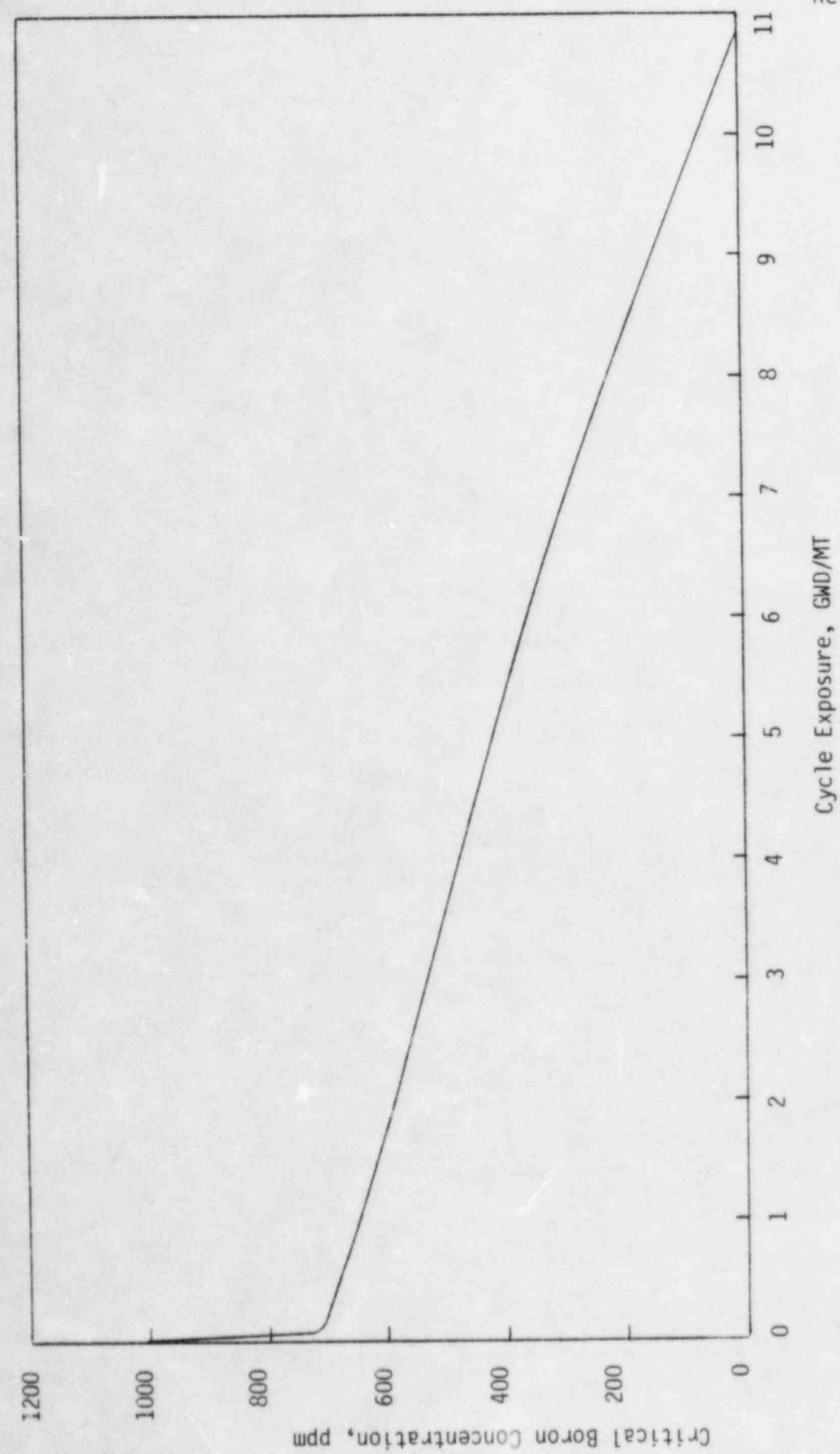

Figure 6.1 H.B. Robinson Unit 2, Cycle 10, HFP Critical Boron Concentration versus Cycle Exposure for 2,300 MWt

	H	G	F	E	D	C	B	A
8	*** + 1.265	1.244	1.029	*** 1.219	1.243	1.078	** 1.051	.283 PLSA
9	1.245	1.258	1.190	1.002	1.232	1.271	** 1.054	.236 PLSA
10	1.030	1.191	1.013	*** 1.162	1.034	1.029	1.034 +	
11	*** 1.221	1.004	*** 1.163	.991	1.158	** 1.069	.765	
12	1.248	1.235	1.037	1.160	** 1.031	.496		
13	1.082	1.275	1.031	** 1.071	.497			
14	** 1.054	** 1.056	1.036 +	.767				
15	.284 PLSA	.237 PLSA						

Relative Assembly Power

Peak Assembly = 1.28 (G13)

Peak $F^N_{\Delta H}$ = 1.43 (H8)

Peak F_{xy} = 1.52 (G9)

Peak F^N_Q = 1.82 (G14)

** 8 pins of 4 w/o gadolinia per assembly
*** 12 pins of 4 w/o gadolinia per assembly
\+ Region 12 fresh fuel loaded at BOC10
PLSA Part Length Shielding Assembly

Figure 6.2 H.B. Robinson Unit 2 Cycle 10, Assembly Power Distribution for A Cycle Exposure of 100 MWD/MT 3-D XTG analysis at 2,300 MWt

	H	G	F	E	D	C	B	A
8	*** + 1.360	1.141	.981	*** 1.323	1.163	1.031	** 1.158	.349 PLSA
9	1.141	1.135	1.114	1.005	1.150	1.182	** 1.124	.285 PLSA
10	.982	1.115	1.021	*** 1.315	1.020	.989	.982 +	
11	*** 1.325	1.005	*** 1.316	1.032	1.152	** 1.125	.744	
12	1.166	1.152	1.022	1.153	** 1.119	.537		
13	1.032	1.183	.990	** 1.126	.537			
14	** 1.159	** 1.125	.982 +	.744				
15	.350 PLSA	.285 PLSA						

Relative Assembly Power

Peak Assembly = 1.36 (H8)

Peak $F^N_{\Delta H}$ = 1.50 (H8)

Peak F_{xy} = 1.61 (H8)

Peak F^N_Q = 1.82 (H14)

** 8 pins of 4 w/o gadolinia per assembly
*** 12 pins of 4 w/o gadolinia per assembly
\+ Region 12 fresh fuel loaded at BOC10
PLSA Part Length Shielding Assembly

Figure 6.3 H.B. Robinson Unit 2 Cycle 10, Assembly Power Distribution for a Cycle Exposure of 5,000 MWD/MT, 3-D XTGPWR Analysis at 2,300 MWt

	H	G	F	E	D	C	B	A
8	*** + 1.282	1.075	.963	*** 1.314	1.124	1.025	** 1.214	.402 PLSA
9	1.075	1.073	1.076	1.002	1.114	1.148	** 1.167	.327 PLSA
10	.963	1.076	1.018	*** 1.330	1.019	.988	+ .969	
11	*** 1.314	1.002	*** 1.330	1.041	1.142	** 1.155	.746	
12	1.125	1.115	1.020	1.143	** 1.162	.579		
13	1.025	1.148	.988	** 1.155	.579			
14	** 1.213	** 1.167	+ .969	.746				
15	.401 PLSA	.327 PLSA						

Relative Assembly Power

Peak Assembly = 1.33 (F11)

Peak $F^N_{\Delta H}$ = 1.42 (E10)

Peak F_{xy} = 1.60 (E10)

Peak F^N_Q = 1.68 (E10)

** 8 pins of 4 w/o gadolinia per assembly
*** 12 pins of 4 w/o gadolinia per assembly
\+ Region 12 fresh fuel loaded at BOC10
PLSA Part Length Shielding Assembly

Figure 6.4 H.B. Robinson Unit 2 Cycle 10, Assembly Power Distribution for a Cycle Exposure of 10,820 MWD/MT, 3-D XTGPWR Analysis at 2,300 MWt

7.0 THERMAL-HYDRAULIC DESIGN

The Cycle 10 core consists of ENC supplied fuel assemblies as shown in Table 4.1 The hydraulic design of all fuel types are identical, thereby ensuring compatibility. The hydraulic design was analyzed and the results of the analysis documented in Reference 14.

The thermal hydraulic performance of the H.B. Robinson Unit 2 core has been analyzed. A bounding core assembly power distribution at the exposure which exhibited the worst radial peaking was used in the analysis. The maximum technical specification limif ($F^N_{\Delta H}$ = 1.65) assembly was assumed. A boundingly low local rod power distribution was used in the analysis of the maximum radially peaked assembly to provide low departure from nucleate boiling ratio (DNBR) results. The analysis uses a lower plenum flow maldistribution factor of 5% to the limiting fuel assembly, 4.5% core flow bypass and a low estimate of total recirculating primary coolant flow rate for 6% steam generator tube plugging with a further 3% reduction to account for flow measurement uncertainty. Therefore, the analysis of the core as constituted considered all fuel types in a bounding manner.

The results of the thermal-hydraulic performance analyses are used as bases for the analyses of anticipated operational occurences[15]. The results of the plant transient analyses show that the specified acceptable fuel design limits for each event as defined in the plant licensing bases are met. The results of the analysis are reported in Reference 15.

8.0 ACCIDENT AND TRANSIENT ANALYSIS

8.1 PLANT TRANSIENT AND ECCS ANALYSES

The plant transient[15] and ECCS[16] analyses are performed to support operation at a power level of 2,300 MWt with 6% of the tubes plugged in the replacement steam generators. Confirmation that operation with an $F_{\Delta H}^{N}$ of 1.65 and an F_Q^T of 2.32 meets the acceptance criteria for each event as defined in the licensing basis for H. B. Robinson Unit 2 is provided in References 15 and 16.

8.2 ROD EJECTION ANALYSIS

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, Reference 17. 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 resultant peaking factors. The pellet energy deposition resulting from an ejected rod was conservatively evaluated explicitly for BOC and EOC conditions. The HFP pellet energy deposited was calculated to be 165 cal/gm at BOC and 172 cal/gm at EOC. The HZP pellet energy deposition was calculated to be less than 40 cal/gm for both BOC and EOC conditions. The rod ejection accident was found to result in an energy deposition of less than the 280 cal/gm limit as stated in Regulatory Guide 1.77. The significant parameters for the analyses, along with the results, are summarized in Tables 8.1 and 8.2

8.3 RADIOLOGICAL ASSESSMENT OF POSTULATED ACCIDENTS

With the increased fuel exposures planned for H.B. Robinson Unit 2 fuel, the potential radiological consequence of the postulated accidents have been assessed to verify that the 10 CFR 100 limits continue to be satisfied. This assessment assumes that the reactor is operated at 2,300 MWt with a peak assembly exposure of 44,000 MWD/MT. The results of the assessment (Reference 17) indicate that the whole body and thyroid dose received from the postulated loss-of-coolant accident (LOCA), fuel handling accident (FHA), and other events with high burnup fuel are conservatively shown to be well below values prescribed in 10 CFR 100.

Table 8.1 H.B.Robinson Unit 2, Cycle 10 Ejected Rod Analysis, HFP

		BOC		EOC	
		Value	Contribution[a] to Energy Deposition, (cal/gm)	Value	Contribution[a] to Energy Deposition, (cal/gm)
A.	Initial Fuel Enthalpy (cal/gm)	86.5	----	102.5	----
B.	Generic Initial Fuel Enthalpy (cal/gm)	40.8	----	40.8	----
C.	Delta Initial Fuel Enthalpy (cal/gm)	45.7	45.7	61.7	61.7
D.	Maximum Control Rod Worth (pcm)	100	121	130	125
E.	Doppler Coefficient (pcm/^{o}F)	-1.3[e]	0.99[b]	-1.5 [e]	0.89[b]
F.	Delayed Neutron Fraction, β	0.0061	1.00[b]	0.0053	1.03[b]
G.	Power Peaking Factor	2.7	----	4.1	----
H.	Power Peaking Factor Used(c)	4.0	----	5.0	----
	Total		165.0[d]		172.0[d]

(a) The contribution to the total pellet energy deposition is a function of initial fuel enthalpy, maximum control rod worth, Doppler coefficient, and delayed neutron fraction. The energy deposition contribution values and factors are derived from data calculated in the "Generic Analysis of the Control Rod Ejection Transient. . ." document.

(b) These values are multiplication factors applied to (C + D).

(c) The energy deposition due to maximum control rod worth is a function of the power peaking factor.

(d) Total pellet energy deposition (cal/gm) calculated by the equation
Total (cal/gm) = (C+D)(E)(F)

(e) For this Doppler coefficient, conservative values of -1.1 and -1.4 were assumed at BOC and EOC, respectively.

Table 8.2 H.B.Robinson Unit 2, Cycle 10 Ejected Rod Analysis, HZP

	BOC		EOC	
	Value	Contribution[a] to Energy Deposition, (cal/gm)	Value	Contribution[a] to Energy Deposition, (cal/gm)
A. Initial Fuel Enthalpy (cal/gm)	16.7	----	16.7	----
B. Generic Initial Fuel Enthalpy (cal/gm)	16.7	----	16.7	----
C. Delta Initial Fuel Enthalpy (cal/gm)	0.0	0.0	0.0	0.0
D. Maximum Control Rod Worth (pcm)	600	33	600	33
E. Doppler Coefficient (pcm/^{o}F)	1.7[e]	1.04[b]	-1.8[e]	0.73[b]
F. Delayed Neutron Fraction, β	0.0061	1.00[b]	0.0053	1.13[b]
G. Power Peaking Factor	4.9	----	6.8	----
H. Power Peaking Factor Used(c)	13.0	----	13.0	----
TOTAL		34.0[d]		27.0)d)

(a) The contribution to the total pellet energy deposition is a function of initial fuel enthalpy, maximum control rod worth, Doppler coefficient, and delayed neutron fraction. The energy deposition contribution values and factors are derived from data calculated in the "Generic Analysis of the Control Rod Ejection Transient. . ." document.

(b) These values are multiplication factors applied to (C + D).

(c) The energy deposition due to maximum control rod worth is a function of the power peaking factor.

(d) Total pellet energy deposition (cal/gm) calculated by the equation
Total (cal/gm) = (C+D)(E)(F)

(e) For this Doppler coefficient, conservative values of -1.0 and -1.4 were assumed at BOC and EOC, respectively.

9.0 REFERENCES

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H. B. ROBINSON UNIT 2, CYCLE 10

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ATTACHMENT 10

XN-NF-84-72

H. B. ROBINSON UNIT 2
LARGE BREAK LOCA-ECCS ANALYSIS
WITH INCREASED ENTHALPY RISE FACTOR