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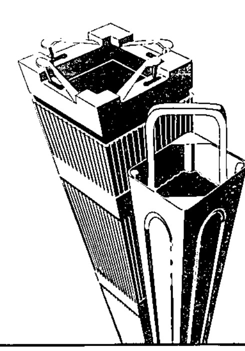
Siemens Nuclear Power Corporation
H. B. Robinson Unit 2, Cycle 15
Safety Analysis Report

February 1992



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H. B. ROBINSON UNIT 2, CYCLE 15
SAFETY ANALYSIS REPORT

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February 1992

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H. B. ROBINSON UNIT 2, CYCLE 15
SAFETY ANALYSIS REPORT

1.0 INTRODUCTION

Results of the Safety Analysis evaluation for Cycle 15 of the H. B. Robinson Unit 2 nuclear plant are presented in this report. The Cycle 15 analyses reflect plant operation at 2,300 MWt. The topics addressed herein include an operating history of the reference cycle, power distribution considerations, control rod reactivity requirements, temperature coefficient considerations, setpoint analysis, and Standard Review Plan Chapter 15 Event analysis.

The Cycle 15 design requires the loading of forty-eight (48) fresh Siemens Nuclear Power Corporation (SNP) supplied fuel assemblies. The forty-eight (48) ANF-12 Region 18 fuel assemblies utilize natural uranium axial blankets (NUAB), gadolinia-bearing fuel rods, High Thermal Performance (HTP) spacers, and Intermediate Flow Mixers (IFM).

2.0 SUMMARY

Cycle 15 of the H. B. Robinson Unit 2 nuclear plant is designed to operate at 2,300 MWt starting in June 1992. The characteristics of the fuel and of the reload core result in conformance with required shutdown margins and thermal limits. This document provides the safety analysis for the plant during Cycle 15 operation.

The Cycle 15 core consists of 12 Part Length Shielding Assemblies (PLSA), one standard (reinserted) XN-3 assembly, 48 standard ANF-10 assemblies, 48 ANF-11 High Thermal Performance assemblies, and 48 ANF-12 High Thermal Performance fuel assemblies. The SNP fuel mechanical design characteristics are discussed in Section 5.0.

The thermal-hydraulic characteristics of the Cycle 15 core design are discussed in Section 7.0 of this document. A review of the applicability to Cycle 15 of the current analysis of record, for the Standard Review Plan Chapter 15 events and reactor trip setpoint analyses, is provided in Section 8.0 of the document.

Plant transient and setpoint analyses reported and reviewed here support operation at 2,300 MWt within the licensed limits. The Cycle 15 analysis supports a peak assembly average burnup of 52,500 MWd/MTU. The analysis also supports an $F_{\Delta H}$ of 1.70 and an F_Q of 2.40 for all Chapter 15 events except for the LOCA analyses. The reference small break LOCA analysis[28] supports an $F_{\Delta H}$ of 1.65 and an F_Q of 2.32. The large break LOCA/ECCS analyses[29,41] support an $F_{\Delta H}$ of 1.75 and an F_Q of 2.50.

3.0 OPERATING HISTORY OF THE REFERENCE CYCLE

H. B. Robinson Unit 2 Cycle 14 has been chosen as the reference cycle due to the close resemblance of the overall neutronic characteristics of Cycle 15 to Cycle 14.

The measured power peaking factors have remained within the Technical Specification limits for Cycle 14. The limits for the total nuclear peaking factor, F^T_Q, and the radial nuclear pin peaking factor, $F_{\Delta H}$, were 2.32 and 1.65, respectively. Cycle 14 operation has typically been rod free with Control Bank D positioned in the range of 215 steps to ARO. It is anticipated that similar control bank insertions will be seen in Cycle 15.

The Cycle 14 XTGPWR calculated versus measured boron letdown curve is shown in Figure 3.1. Based upon experience, the deviations observed in Figure 3.1 are expected to remain within the 1000 pcm Technical Specification limit. A representative Cycle 14 PDQ calculated versus measured power distribution comparison is shown in Figure 3.2 at a Cycle 14 exposure of 6,791 MWd/MTU with Bank D inserted to 219 steps at a core power level of 99.9% of 2,300 MWt.

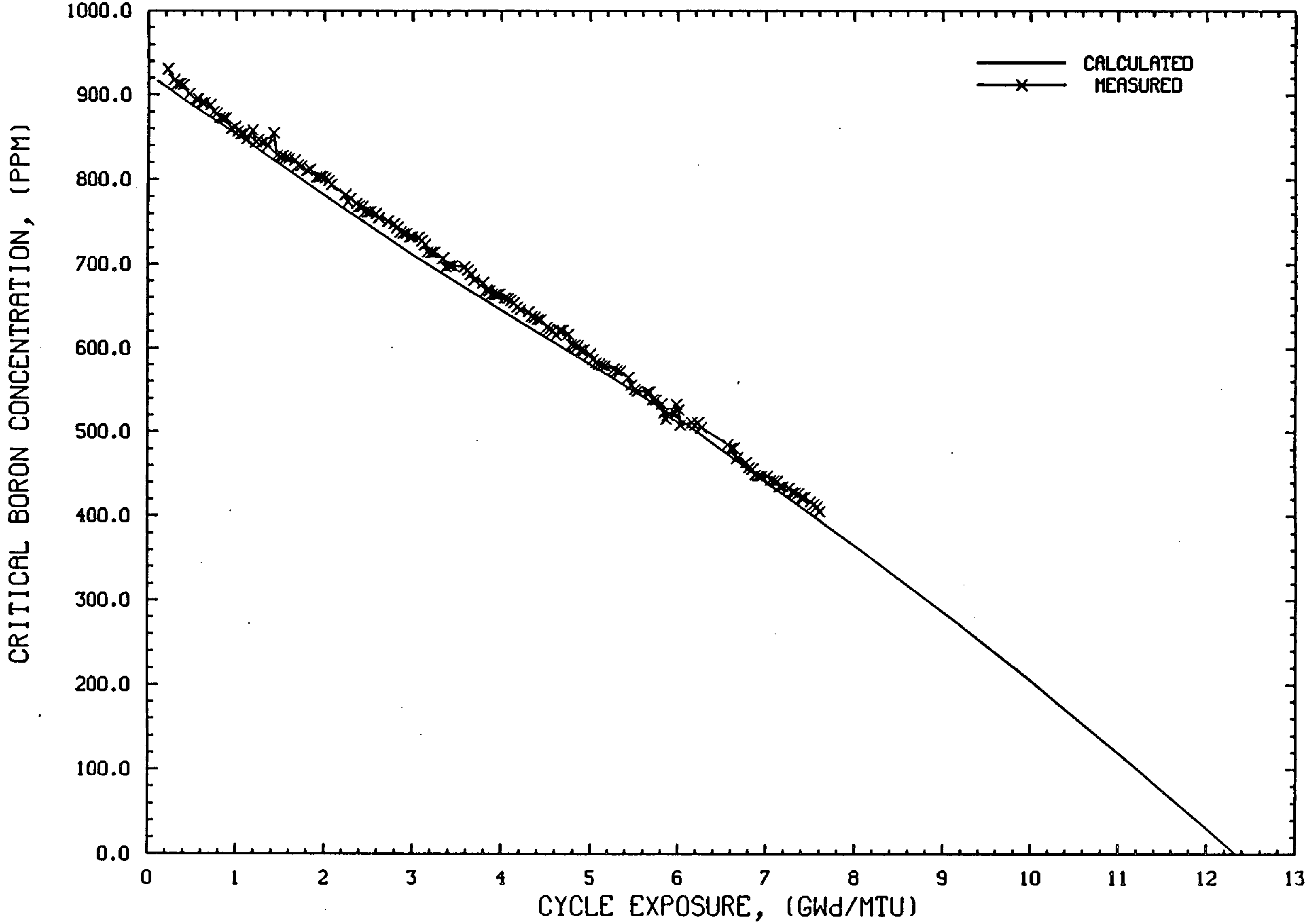

FIGURE 3.1 H. B. ROBINSON UNIT 2 CYCLE 14, CRITICAL
BORON CONCENTRATION VERSUS CYCLE EXPOSURE, HFP, 2,300 MWT

	H	G	F	E	D	C	B	A
8	0.907 0.933 2.9	1.051 1.082 2.9	1.293 1.319 2.0	1.075 1.083 0.7	0.972 0.969 -0.3	1.367 1.357 -0.7	0.818 0.814 -0.5	0.205 0.200 -2.4
9	1.052 1.076 2.3	1.088 1.114 2.4	0.912 0.927 1.6	0.955 0.962 0.7	1.149 1.144 -0.4	1.209 1.199 -0.8	1.215 1.201 -1.2	0.200 0.196 -2.0
10	1.293 1.312 1.5	0.910 0.923 1.4	0.968 0.983 1.5	1.302 1.312 0.8	1.086 1.083 -0.3	1.157 1.143 -1.2	1.160 1.143 -1.5	
11	1.075 1.080 0.5	0.953 0.958 0.5	1.301 1.312 0.8	1.046 1.053 0.7	1.226 1.229 0.2	1.207 1.205 -0.2	0.855 0.853 -0.2	
12	0.969 0.958 -1.1	1.146 1.138 -0.7	1.086 1.089 0.3	1.229 1.234 0.4	0.935 0.940 0.5	0.487 0.491 0.8		
13	1.340 1.301 -2.9	1.204 1.186 -1.5	1.156 1.149 -0.6	1.208 1.205 -0.2	0.488 0.492 0.8			
14	0.813 0.793 -2.5	1.210 1.186 -2.0	1.158 1.149 -0.8	0.855 0.855 0.0				
15	0.204 0.197 -3.4	0.199 0.195 -2.0						

PDQ
MEASURED
% DIFFERENCE

FIGURE 3.2 H. B. ROBINSON UNIT 2 CYCLE 14, RELATIVE POWER DISTRIBUTION MAP AT 99.9% HFP, 6,791 MWd/MTU, BANK D AT 219 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, 20 RCC guide tubes, and one instrumentation tube. The RCC guide tubes and the instrumentation tube are made of Zircaloy. The ANF-11 (Region 17) and ANF-12 (Region 18) fuel designs incorporate 1 bimetallic Zircaloy-4/Inconel 718 grid spacer, 6 High Thermal Performance (HTP) spacers, and 3 Intermediate Flow Mixers (IFM). The balance of the previously irradiated fuel contains seven Zircaloy spacers with Inconel springs. The fuel rods consist of slightly enriched UO_2 pellets inserted into Zircaloy tubes.

The Cycle 15 design reflects the loading of 48 fresh SNP supplied fuel assemblies. This core design contains the sixth reload of natural uranium axial blanket (NUAB) fuel in H.B. Robinson Unit 2 and the seventh successive reload containing gadolinia-bearing fuel. The ANF-12 (Region 18) fuel design reflects the use of a split batch enrichment to achieve the reference loading pattern design and cycle endurance. Within the forty-eight assembly reload design, forty assemblies contain a central zone enrichment of 4.20 w/o U-235, four assemblies contain a central zone enrichment of 4.10 w/o U-235, and the remaining four assemblies contain a central zone enrichment of 3.85 w/o U-235. The ANF-12 (Region 18) reload design contains the following distribution of fresh assembly types:

U-235 Enrichment, w/o	Number of Assemblies	Gadolinia Loading Per Assembly	U-235 Enrichment in Gadolinia Pins, w/o
4.20	8	12 Pins with 6 w/o Gd	2.90
4.20	8	6 Pins with 6 w/o Gd Asymmetrically Loaded	2.90
4.20	8	6 Pins with 4 w/o Gd Asymmetrically Loaded	2.90
4.20	8	12 Pins with 6 w/o Gd Asymmetrically Loaded	2.90
4.20	8	No Gadolinia Pins	----
4.10	4	12 Pins with 6 w/o Gd	2.70
3.85	4	12 Pins with 6 w/o Gd	2.70

The total number of gadolinia-bearing fuel pins required for Cycle 15 is 384.

One twice-burnt XN-3 (Region 9) assembly (Fabrication ID = J21) has been reinserted into the center core location (H-08) from an initial discharge at EOC7. This reinsert assembly has an initial central zone enrichment of 2.90 w/o U-235 over the total fuel pin length of 144 inches.

The design for Batch ANF-12 includes natural uranium axial blankets in the top and bottom six inches of the active fuel region of the non-gadolinia-bearing fuel pins. In the 384 gadolinia-bearing fuel pins, extended natural uranium axial blankets have been incorporated. The additional six inches of natural uranium in the top and bottom fuel pin regions have been added to the gadolinia-bearing fuel pins to flatten the axial power distribution.

The enrichment of the gadolinia-bearing fuel pins is selected to assure that the temperature in those pins never reaches the temperature of the limiting fuel pins. An enrichment of 2.90 w/o U-235 in the ANF-12 gadolinia-bearing fuel pins provides ample margin in th assemblies with a central zone enrichment of 4.20 w/o U-235. In the ANF-12 assembli containing a central zone enrichment of 3.85 w/o U-235 or 4.10 w/o U-235, an enrichment of 2.70 w/o U-235 in the gadolinia-bearing fuel pins has been used.

The projected Cycle 15 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. BOC15 and EOC15 exposures based on an EOC14 exposure of 12,360 MWd/MTU, along with Region IDs, are shown in quarter core representation in Figure 4.2. The initial enrichments in the various fuel regions are listed in Tables 4.1 through 4.3.

TABLE 4.1 H.B. ROBINSON UNIT 2, CYCLE 15 FUEL ASSEMBLY DESIGN PARAMETERS (PLSA, REINSERT ASSEMBLY, TWICE-BURNT ANF-10 FUEL)

FUEL TYPE	PLSA	PDAS	*A*	*B*	*C*	*D*	*E*	*F*
REGION	13	9	16	16	16	16	16	16
Reload Number (XN, ANF-#)	XN-7	XN-3	10	10	10	10	10	10
Number of Assemblies	12	1	8	12	4	8	8	8
Pellet Density (%TD)	94	94	94	94	94	94	94	94
Pellet to Clad Diametral Gap (Mil)	7.5	7.5	7.5	7.5	7.5	7.5	7.5	7.5
Initial Enrichment w/o U235								
Upper 6 inches								
UO2	.71	2.90	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	—	—	—	—	—	—	—	—
Central 132 inches								
UO2	1.24	2.90	3.85	3.45	3.85	3.85	3.45	3.45
UO2 + Gd2O3	—	—	—	2.40 2.70+	2.70	2.70	2.70	2.70
	(1)			(2)	(2)	(2)	(2)	(2)
Lower 6 inches								
UO2	—	2.90	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	—	—	—	—	—	—	—	—
Assembly Average Enrichment, w/o	0.86	2.90	3.59	3.17	3.52	3.52	3.21	3.20
Initial Gd2O3, w/o	—	—	—	4,6	4,6	4	4	4

PLSA Part Length Shield Assembly.
PDAS Prematurely Discharged Reinsert Assembly.
A 3.85 w/o with no gadolinia pins.
B 3.45 w/o with (8 pins @ 4 w/o Gd + 4 pins @ 6 w/o Gd)
C 3.85 w/o with (8 pins @ 4 w/o Gd + 4 pins @ 6 w/o Gd)
D 3.85 w/o with (12 pins @ 4 w/o Gd)
E 3.45 w/o with (4 pins @ 4 w/o Gd), (asymmetrically loaded)
F 3.45 w/o with (2 pins @ 4 w/o Gd), (asymmetrically loaded)
\+ 2.40 w/o U235 in 6 w/o gadolinia and 2.70 w/o U235 in 4 w/o gadolinia.
(1) Bottom 42 inches of the assembly contains 304 SSTL.
(2) Six additional inches of natural uranium axial blanket fuel is utilized in the top and bottom of gadolinia-bearing fuel pins.

TABLE 4.2 H.B. ROBINSON UNIT 2, CYCLE 15 FUEL ASSEMBLY DESIGN PARAMETERS (ONCE-BURNT ANF-11 FUEL)						
FUEL TYPE	-A-	-B-	-C-	-D-	-E-	-F-
REGION	17	17	17	17	17	17
Reload Number (ANF-#)	11	11	11	11	11	11
Number of Assemblies	4	8	8	8	8	12
Pellet Density (%TD)	94	94	94	94	94	94
Pellet to Clad Diametral Gap (Mil)	7.5	7.5	7.5	7.5	7.5	7.5
Initial Enrichment w/o U235						
Upper 6 Inches						
UO2	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	---	---	---	---	---	---
Central 132 Inches						
UO2	4.20	4.20	4.20	4.20	4.20	3.85
UO2 + Gd2O3	2.90	2.90	2.90	2.90	---	2.70
	(1)	(1)	(1)	(1)	---	(1)
Lower 6 Inches						
UO2	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	---	---	---	---	---	---
Assembly Average Enrichment	3.83	3.83	3.87	3.88	3.91	3.52
Initial Gd2O3	6	6	4	4	---	6

-A- 4.20 w/o with 12 pins of 6 w/o Gd2O3
-B- 4.20 w/o with 12 pins of 6 w/o Gd2O3 (asymmetrically loaded)
-C- 4.20 w/o with 6 pins of 4 w/o Gd2O3 (asymmetrically loaded)
-D- 4.20 w/o with 4 pins of 4 w/o Gd2O3 (asymmetrically loaded)
-E- 4.20 w/o with no gadolinia pins.
-F- 3.85 w/o with 12 pins of 6 w/o Gd2O3
(1) Six additional inches of natural uranium axial blanket fuel is utilized in the top and bottom of gadolinia-bearing fuel pins.

TABLE 4.3 H.B. ROBINSON UNIT 2, CYCLE 15 FUEL ASSEMBLY DESIGN PARAMETERS (FRESH ANF-12 FUEL)

FUEL TYPE	FA12	FB12	FC12	FD12	FE12	FF12	FG12
REGION	18	18	18	18	18	18	18
Reload Number (ANF-#)	12	12	12	12	12	12	12
Number of Assemblies	4	4	8	8	8	8	8
Pellet Density (%TD)	95	95	95	95	95	95	95
Pellet to Clad Diametral Gap (Mil)	7.0	7.0	7.0	7.0	7.0	7.0	7.0
Initial Enrichment w/o U235							
Upper 6 Inches							
UO2	.71	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	---	---	---	---	---	---	---
Central 132 Inches							
UO2	4.10	3.85	4.20	4.20	4.20	4.20	4.20
UO2 + Gd2O3	2.70	2.70	2.90	2.90	2.90	2.90	---
	(1)	(1)	(1)	(1)	(1)	(1)	---
Lower 6 Inches							
UO2	.71	.71	.71	.71	.71	.71	.71
UO2 + Gd2O3	---	---	---	---	---	---	---
Assembly Average Enrichment	3.73	3.52	3.83	3.87	3.87	3.83	3.91
Initial Gd2O3	6	6	6	6	4	6	---

FA12 4.10 w/o with 12 pins of 6 w/o Gd2O3
FB12 3.85 w/o with 12 pins of 6 w/o Gd2O3
FC12 4.20 w/o with 12 pins of 6 w/o Gd2O3
FD12 4.20 w/o with 6 pins of 6 w/o Gd2O3 (asymmetrically loaded)
FE12 4.20 w/o with 6 pins of 4 w/o Gd2O3 (asymmetrically loaded)
FF12 4.20 w/o with 12 pins of 6 w/o Gd2O3 (asymmetrically loaded)
FG12 4.20 w/o with no gadolinia
(1) Six additional inches of natural uranium axial blanket fuel is utilized in the top and bottom of gadolinia-bearing fuel pins.

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							N45 15 G	N49 15 H	N53 15 J						
2					W48 FG12	W31 FE12	W23 FD12	T10 9 H	W24 FD12	W32 FE12	W47 FG12				
3				T15 13 F	W38 FF12	U37 6 B	U43 2 J	*W08 FB12	U44 2 G	U34 6 P	W35 FF12	T16 13 K			
4			T19 10 C	U20 3 L	U28 11 P	T44 12 G	U09 5 K	T24 5 H	U10 5 F	T43 12 J	U27 11 B	U21 3 E	T20 10 N		
5		W44 FG12	W37 FF12	U30 2 E	T25 9 G	W16 FC12	T06 11 M	U03 10 H	T05 11 D	W15 FC12	T26 9 J	U29 2 L	W36 FF12	W43 FG12	
6		W25 FE12	U40 14 K	T47 9 D	W12 FC12	T31 12 D	T40 3 J	W02 FA12	T39 3 G	T32 12 M	W11 FC12	T41 9 M	U39 14 F	W29 FE12	
7	N56 9 A	W17 FD12	U42 7 P	U05 6 L	T04 4 E	T37 7 N	U23 11 C	U15 13 H	U18 11 N	T34 7 C	T03 4 L	U06 6 E	U45 7 B	W22 FD12	N46 9 R
8	N52 8 A	T09 8 G	W07 FB12	T23 8 L	U02 8 F	W01 FA12	U14 8 C	J21 PDAS	U16 8 N	W03 FA12	U04 8 K	T21 8 E	W05 FB12	T11 8 J	N50 8 R
9	N48 7 A	W18 FD12	U41 9 P	U08 10 L	T01 12 E	T38 9 N	U22 5 C	U13 3 H	U19 5 N	T33 9 C	T02 12 L	U07 10 E	U46 9 B	W21 FD12	N54 7 R
10		W26 FE12	U35 2 K	T48 7 D	W09 FC12	T30 4 D	T36 13 J	W04 FA12	T35 13 G	T29 4 M	W10 FC12	T42 7 M	U36 2 F	W30 FE12	
11		W41 FG12	W34 FF12	U31 14 E	T28 7 G	W13 FC12	T07 5 M	U01 6 H	T08 5 D	W14 FC12	T27 7 J	U32 14 L	W40 FF12	W42 FG12	
12			T18 6 C	U17 13 L	U25 5 P	T46 4 G	U12 11 K	T22 11 H	U11 11 F	T45 4 J	U26 5 B	U24 13 E	T17 6 N		
13				T14 3 F	W33 FF12	U38 10 B	U48 14 J	*W06 FB12	U47 14 G	U33 10 P	W39 FF12	T13 3 K			
14					W45 FG12	W27 FE12	W19 FD12	T12 7 H	W20 FD12	W28 FE12	W46 FG12				
15							N55 1 G	N51 1 H	N47 1 J						

: Fabrication ID/Source Assy
: Cycle 14 Location/Fr Fuel

FA12	12 pins of 6 w/o Gadolinia, 4.10 w/o CZE
FB12	12 pins of 6 w/o Gadolinia, 3.85 w/o CZE
FC12	12 pins of 6 w/o Gadolinia, 4.20 w/o CZE
FD12	6 pins of 6 w/o Gadolinia, 4.20 w/o CZE (asymmetrically loaded)
FE12	6 pins of 4 w/o Gadolinia, 4.20 w/o CZE (asymmetrically loaded)
FF12	12 pins of 6 w/o Gadolinia, 4.20 w/o CZE (asymmetrically loaded)
FG12	No Gadolinia Pins, 4.20 w/o CZE
*	Secondary Source Assembly, SN's: SS07, SS08

FIGURE 4.1 H. B. ROBINSON UNIT 2 CYCLE 15, REFERENCE LOADING PATTERN FOR AN EOC14 EXPOSURE OF 12,360 MWd/MTU

	H	G	F	E	D	C	B	A
8	9 20.590 33.220	17 16.308 31.365	18 0.000 17.473 FA12	17 16.332 31.383	16 29.810 42.335	18 0.000 17.463 FB12	16 29.292 38.668	13 20.339 23.420
9	17 16.313 31.346	17 14.766 30.067	16 27.868 39.546	16 24.897 37.934	17 16.241 30.981	17 14.444 31.184	18 0.000 15.134 FD12	13 17.962 20.999
10	18 0.000 17.430 FA12	16 27.870 39.523	16 26.242 38.720	18 0.000 17.547 FC12	16 26.783 39.584	17 14.105 30.698	18 0.000 14.994 FE12	
11	17 16.336 31.338	16 24.931 37.925	18 0.000 17.513 FC12	16 28.377 41.737	17 10.353 26.983	18 0.000 15.869 FF12	18 0.000 10.962 FG12	
12	16 29.745 42.177	17 16.243 30.906	16 26.787 39.549	17 10.360 26.957	17 14.784 27.559	16 29.479 35.117		
13	18 0.000 16.547 FB12	17 14.449 31.046	17 14.111 30.643	18 0.000 15.829 FF12	16 29.491 35.114	REGION ID BOC15 (GWd/MTU) EOC15 (GWd/MTU) FRESH FUEL TYPE (See Table 4.3)		
14	16 29.297 38.569	18 0.000 15.033 FD12	18 0.000 14.940 FE12	18 0.000 10.936 FG12				
15	13 20.344 23.396	13 17.966 20.977						

FIGURE 4.2 H. B. ROBINSON UNIT 2 CYCLE 15, BOC AND EOC EXPOSURE DISTRIBUTION FOR AN EOC14 EXPOSURE OF 12,360 MWd/MTU

5.0 MECHANICAL DESIGN

The forty-eight (48) Batch ANF-12 (Region 18) fresh reload 15 x 15 fuel assemblies each contain 204 fuel rods, 1 instrument tube, and 20 guide tubes. The fuel assembly design is the same as the previous reload in that it contains 6 all Zircaloy high thermal performance (HTP) spacers and 3 intermediate flow mixers (IFMs). The removable stainless steel upper tie plate with Inconel 718 leaf springs is unchanged as is the lower tie plate, which is a small hole debris resistant design. The fuel rod design differs from the previous reload in that the fuel pellet density and geometry are revised. The UO_2 and UO_2-Gd_2O_3 nominal pellet densities are revised upward, while the pellet/clad gap has been decreased 0.0005 inch. The UO_2 and UO_2-Gd_2O_3 pellet length is changed to be consistent with a 1.15 L/D. The length of the natural pellets used in the blanket ends of the rods remains unchanged. For UO_2 fuel rods, the 144 inch fuel column includes a 6 inch column of natural UO_2 pellets at each end. The gadolinia-bearing fuel rods contain an enriched gadolinia-bearing region of 120 inches with 12 inches of natural uranium at each end of the rod.

The ANF-12 fuel has been evaluated in accordance with the design methods contained in References 1, 2, and 43. The analytical results given in Reference 4 assure that the mechanical criteria are met.

The previously irradiated assemblies consist of 48 Batch ANF-11 High Thermal Performance assemblies, 48 Batch ANF-10 standard fuel assemblies, 1 reinserted Batch XN-3 standard fuel assembly, and 12 Part Length Shielding Assemblies (PLSA). These assemblies are described in References 3, 4, 5, and 6.

6.0 NUCLEAR CORE DESIGN

The H. B. Robinson Unit 2, Cycle 15, ANF-12 (Region 18) reload design has been developed in accordance with the following requirements:

- The Cycle 15 reload shall contain 48 fresh fuel assemblies.
- The rated power for Cycle 15 shall be 2,300 MWt.
- The length of Cycle 15 shall be maximized.
- The length of Cycle 15 shall be determined based on a projected EOC14 exposure of 12,360 MWd/MTU.
- Cycle 15 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.
- In accordance with plant Technical Specifications, the control rod worth requirements shall be met.
- The loading pattern shall be designed to produce acceptable power distributions. The design F^T_Q, including uncertainties, shall be less than or equal to 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.

The neutronics design methods utilized in the analyses are consistent with those described in References 7 through 11.

6.1 Physics Characteristics

The neutronic characteristics of the Cycle 15 core are compared to those of Cycle 14 in Table 6.1. The data presented in the table illustrates the neutronic similarity between Cycles 14 and 15. The reactivity coefficients of the Cycle 15 core are bounded by the coefficients used in the safety analysis. All results presented in this report are based upon a nominal EOC14 cycle length of 12,360 MWd/MTU. The impact of a short Cycle 14 length was explicitly considered for all neutronics calculations. Due to the additional reactivity which is carried over to Cycle 15 when Cycle 14 is shutdown short of its target end of cycle exposure, the physics parameters

determined based upon the short EOC14 cycle length are generally more limiting than those based upon an extended EOC14 cycle length. The conclusions presented in this report are valid for EOC14 exposures between 11,360 MWd/MTU and 12,860 MWd/MTU.

The boron letdown curve for Cycle 15 operation at 2,300 MWt is shown in Figure 6.1. As shown, the BOC15, no xenon, hot full power (HFP, 2,300 MWt) critical boron concentration is predicted to be 1,313 ppm. At 100 MWd/MTU, equilibrium xenon, the critical boron concentration at HFP is 1025 ppm. The Cycle 15 length is projected to be 13,395 MWd/MTU (387 EFPD) with 0 ppm boron at EOC.

6.1.1 Power Distribution Considerations

At a power level of 2,300 MWt at equilibrium xenon conditions, 100 MWd/MTU, the calculated XTGPWR peak $F_{\Delta H}$ for the cycle is 1.59 including a 4% measurement uncertainty. The peak F^T_Q for the cycle occurs at 500 MWd/MTU and is 2.24 including a 3% engineering factor, a 5% measurement uncertainty, K(z) considerations[12], and the V(z) allowance for operation with PDC-3[13] for ±5% target bands.

For the balance of the cycle, the secondary peak $F_{\Delta H}$ is calculated to occur at a cycle exposure of 11,000 MWd/MTU. The predicted value of $F_{\Delta H}$ at this exposure is 1.58 including a 4% measurement uncertainty. The corresponding F^T_Q, including the adjustments discussed above, is 2.11.

The quarter-core radial power distributions are presented in Figures 6.2 and 6.3 for Cycle 15 exposures of 100 MWd/MTU and 11,000 MWd/MTU, respectively.

6.1.2 Control Rod Reactivity Requirements

Detailed calculations of shutdown margins for Cycle 15 are provided in Table 6.2. Since the HZP all rods in (ARI) and N-1 control rod worths were calculated to be less than the corresponding HFP worths at both BOC15 and EOC15, the HZP worths are conservatively used in the shutdown margin calculations. Shutdown margin requirements of 1,000 pcm at BOC and

1,770 pcm at EOC are used in the calculation to be consistent with the Technical Specifications. The Cycle 15 analysis indicates excess shutdown margins of 1,922 pcm at BOC and 548 pcm at EOC.

Based on the nominal EOC14 shutdown exposure, the Cycle 15 refueling boron concentration of 1,950 ppm was verified to be 337 ppm in excess of the boron concentration required to ensure 6% shutdown margin at refueling conditions, with all rods inserted. The 6% shutdown margin requirement at refueling conditions is an updated value, and reflects a change in the H. B. Robinson Unit 2 Technical Specifications for Cycle 15 and beyond.

6.1.3 Moderator and Isothermal Temperature Coefficient Considerations

The Technical Specifications for Cycle 15 require that the moderator temperature coefficient (MTC) be $\leq$+5 pcm/°F below 50% power and $\leq$0 pcm/°F between 50% power and HFP. The MTC values calculated for Cycle 15 are shown in Table 6.1 for HFP, 50% power, and HZP conditions at both BOC (xenon free) and EOC (0 ppm) conditions. As shown, all Technical Specification limits are expected to be met, with no control rod insertion anticipated for ascension to HFP conditions.

The Cycle 15 isothermal temperature coefficients are shown in Table 6.1 for HFP and HZP conditions at both BOC and EOC. At BOC15 the HFP isothermal temperature coefficient is projected to be -7.0 pcm/°F at a xenon free critical boron concentration of 1,313 ppm. The corresponding HZP isothermal temperature coefficient is projected to be -1.5 pcm/°F at a critical boron concentration of 1,553 ppm.

6.2 Power Distribution Control Procedures

The control of the core power distribution is anticipated to be accomplished by following the Siemens Nuclear Power Corporation PDC-3 power distribution control procedure (Reference 13). Two cycle-specific characteristics need to be verified to ensure the H. B. Robinson plant-specific PDC-3 V(z) distributions and allowed deviations from target flux differences remain valid for Cycle 15.

1) In the Cycle 15 loading plan, all assemblies utilize natural uranium axial blankets (NUAB) with the exception of one (1) twice-burnt assembly. This is less than the eight (8) twice-burnt assemblies without NUABs assumed in the generation of the PDC-3 V(z) distributions.

2) The Cycle 15 Bank D worth at the HFP rod insertion limit of 114 steps was calculated to be 408 pcm at 500 MWd/MTU, HFP, equilibrium xenon conditions. This is less than the limit of 548 pcm assumed in the generation of the PDC-3 V(z) distributions.

As a result of the above observations, the H. B. Robinson plant-specific PDC-3 V(z) distributions and allowed deviations from target flux differences, developed in Reference 13, remain valid for the Cycle 15 loading pattern.

6.3 Analytical Methodology

The methods used in the Cycle 15 core analysis are described in References 7 through 11. In summary, the reference neutronic design analysis of the reload core was performed using the XTGPWR[14] reactor simulator. The fuel shuffling between cycles was accounted for in the calculations. The PDQ/HARMONY[15,16] code package will be utilized to monitor the power distribution.

The basic nuclear parameters and neutron cross sections were calculated with the CASMO-2E[17] code, a multigroup two-dimensional transport theory code for burnup calculations. Effective cross sections for gadolinia fuel pins were generated by MICBURN-2[18].

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

TABLE 6.1 H.B. ROBINSON UNIT 2, NEUTRONIC CHARACTERISTICS OF CYCLE 15 COMPARED WITH CYCLE 14 DATA

	Cycle 14		Cycle 15	
	BOC	EOC	BOC	EOC
Critical Boron* (ppm)				
HFP, ARO	1,232	0	1,313	0
50% Power, ARO	1,338	——	1,426	——
HZP, ARO	1,446	——	1,553	——
Moderator Temp. Coef. (pcm/°F)				
HFP	-6.7	-26.1	-5.9	-26.8
50% Power**	-3.3	-24.4	-2.0	-22.6
HZP**	-0.5	-19.9	-0.1	-20.1
Isothermal Temp. Coef. (pcm/°F)				
HFP	-7.8	-27.5	-7.0	-28.2
HZP	-1.9	-21.6	-1.5	-21.8
Pressure Coefficient ($10^{-6}\Delta\rho$/psi)				
HFP	——	+2.90	——	+2.97
HZP	+0.05	——	+0.01	——
Doppler Coefficient (pcm/°F)				
HFP	-1.2	-1.4	-1.1	-1.4
HZP	-1.5	-1.7	-1.4	-1.7
Power Defect (pcm)	1,956	2,738	1,955	2,750
Boron Worth (ppm/10^3 pcm)				
HFP	-117	-98	-122	-100
HZP	-114	-96	-119	-98
Prompt Neutron Lifetime (μsec)	21.9	22.9	21.2	22.4
Delayed Neutron Fraction	0.0065	0.0057	0.0065	0.0057
Control Rod Worth, All Rods In Minus Most Reactive Rod HZP (pcm)	6,135	6,195	6,099	6,167
Excess Shutdown Margin (pcm)	1,967	591	1,922	548

* Xenon free conditions at BOC.

** HFP, equilibrium xenon, and 0 ppm boron used at EOC.

TABLE 6.2 H.B. ROBINSON UNIT 2, CONTROL ROD SHUTDOWN MARGIN AND REQUIREMENTS FOR CYCLE 15

	BOC15 (100 MWd/MTU)		EOC15 (13,395 MWd/MTU)	
	HZP	HFP	HZP	HFP
Control Rod Worth (pcm)				
ARI	7,446	7,446	7,672	7,672
N-1	6,099	6,099	6,167	6,167
(N-1) x 0.9	5,489	5,489	5,550	5,550
Reactivity Insertion (pcm)				
Power Defect	----	1,955	----	2,750
Flux Redistribution	----	174	----	174
Bank D Insertion from PDIL	1,342*	388	1,840*	[illegible]
Void Effects	----	50	----	50
Total Requirements (pcm)	1,342	2,567	1,840	3,232
Shutdown Margin (pcm)				
(N-1) x 0.9 - (Total Requirements)	4,147	2,922	3,710	2,318
Required Shutdown Margin	1,000	1,000	1,770	1,770
Excess Shutdown Margin	3,147	1,922	1,940	548

* Banks D and C in 100 step overlap.

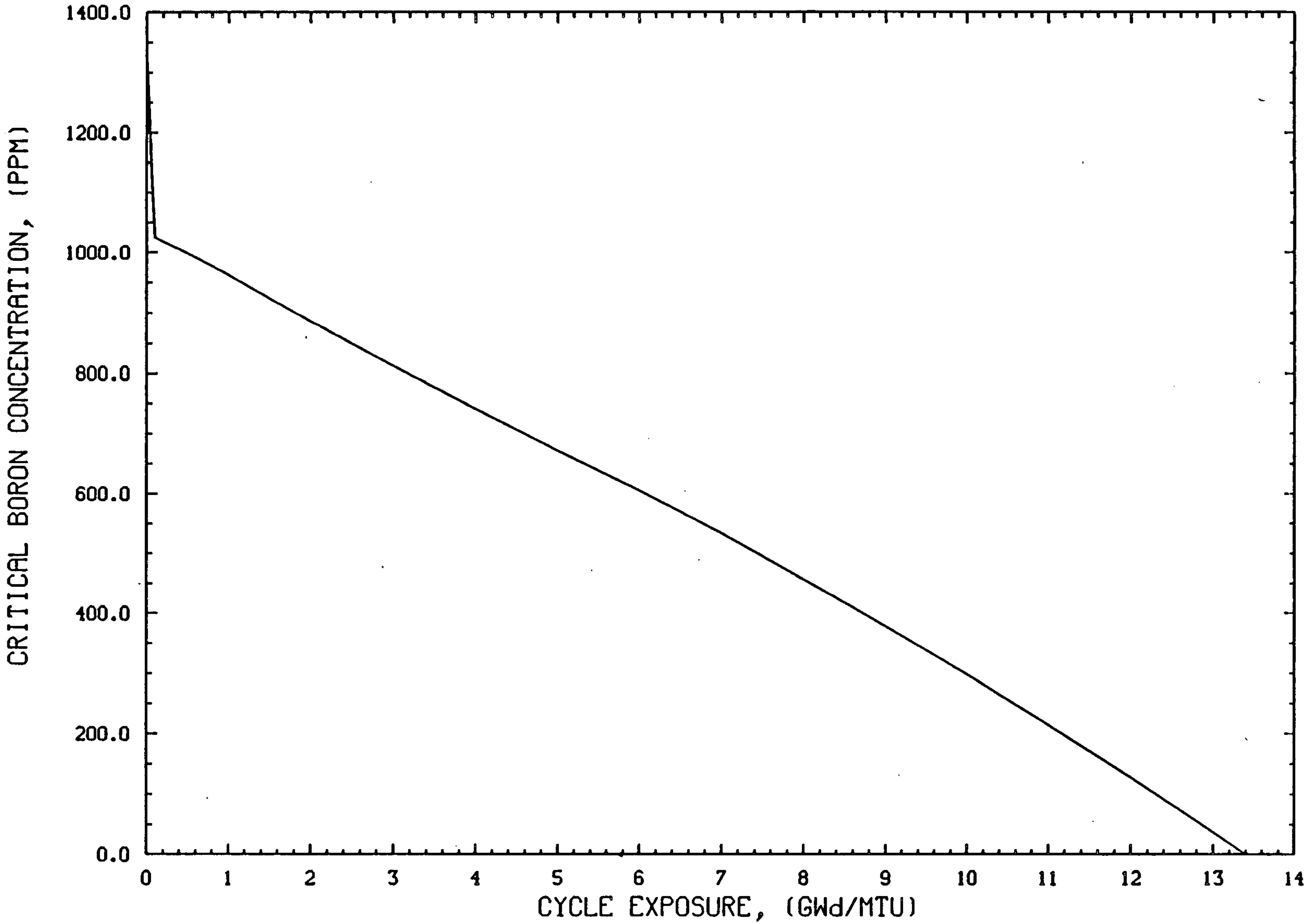

FIGURE 6.1 H. B. ROBINSON UNIT 2 CYCLE 15, CRITICAL BORON CONCENTRATION VERSUS CYCLE EXPOSURE, HFP, ARO, 2,300 MWT

	H	G	F	E	D	C	B	A
8	1.000	1.193	1.303 FA12	1.166	0.959	1.298 FB12	0.667	0.140
9	1.191	1.200	0.865	0.989	1.161	1.326	1.150 FD12	0.143
10	1.298 FA12	0.862	0.913	1.294 FC12	0.984	1.317	1.153 FE12	
11	1.158	0.983	1.290 FC12	1.017	1.327	1.188 FF12	0.856 FG12	
12	0.948	1.151	0.978	1.322	0.987	0.400		
13	1.227 FB12	1.309	1.307	1.182 FF12	0.398			
14	0.657	1.138 FD12	1.144 FE12	0.851 FG12				
15	0.138	0.142						

ASSEMBLY POWER
FRESH FUEL TYPE
(See Table 4.3)

FIGURE 6.2 H. B. ROBINSON UNIT 2 CYCLE 15, ASSEMBLY RELATIVE POWER DISTRIBUTION, 100 MWd/MTU, 2,300 MWT, XTGPWR

	H	G	F	E	D	C	B	A
8	0.933	1.105	1.380 FA12	1.123	0.943	1.373 FB12	0.743	0.187
9	1.104	1.128	0.905	0.995	1.086	1.229	1.175 FD12	0.180
10	1.379 FA12	0.905	0.975	1.401 FC12	0.969	1.217	1.139 FE12	
11	1.123	0.995	1.401 FC12	1.018	1.226	1.253 FF12	0.837 FG12	
12	0.940	1.085	0.969	1.226	0.956	0.450		
13	1.310 FB12	1.226	1.218	1.254 FF12	0.450			
14	0.739	1.174 FD12	1.141 FE12	0.838 FG12				
15	0.187	0.179						

ASSEMBLY POWER
FRESH FUEL TYPE
(See Table 4.3)

FIGURE 6.3 H. B. ROBINSON UNIT 2 CYCLE 15, ASSEMBLY RELATIVE POWER DISTRIBUTION, 11,000 MWd/MTU, 2,300 MWT, XTGPWR

7.0 THERMAL-HYDRAULIC DESIGN

The Cycle 15 core consists of 157 SNP fuel assemblies. As in Cycle 14, fuel assemblies having natural uranium axial blankets and gadolinia-bearing fuel co-reside in the core. Gadolinia-bearing fuel rods are less limiting with respect to MDNBR and centerline melt criteria than non-gadolinia-bearing fuel rods due to their lower enrichment.

There are twelve Part Length Shielding Assemblies (PLSA) loaded on the core periphery. The PLSA assemblies will not approach limiting assembly conditions in their lifetime due to their low operating power. The hydraulic impact of these twelve assemblies on the limiting assembly is negligible.

Reference 40 addressed changes in core design through Cycle 14. Cycle 15 will incorporate 48 ANF-12 High Thermal Performance (HTP) assemblies. The co-resident fuel from previous cycles consist of 12 Part Length Shielding Assemblies (PLSA), one standard (reinserted) XN-3 assembly, 48 ANF-10 Standard Mixing Vane (SMV) assemblies, and 48 ANF-11 HTP assemblies. The HTP fuel assemblies are mechanically identical to the co-resident Standard Mixing Vane (SMV) assemblies with the exception of the lower tie plate flow hole size and hole configuration (debris resistant tie plate), and the design and configuration of the HTP spacer and Intermediate Flow Mixing (IFM) grids. The HTP assemblies incorporate one SMV spacer at the assembly inlet, 6 HTP spacers, and 3 IFMs. A detailed thermal-hydraulic compatibility analysis for the mixed core configuration is documented in Reference 20. In the Reference 20 analysis, the MDNBR for the HTP assemblies is calculated using the ANFP correlation[21] and the MDNBR for the SMV assemblies is calculated using the XNB correlation[22]. A mixed core DNBR penalty[23] of 2% is applied to the 95/95 correlation limit for both the ANFP and XNB correlations.

The thermal-hydraulic performance of the Cycle 15 core under postulated transient and accident conditions is evaluated in Section 8.0 of this document.

8.0 FSAR CHAPTER 15 EVENT ANALYSIS (Transient and Accident Analysis)

The reference analyses through Cycle 14 for the Chapter 15 events are documented in References 24 through 31. The applicability of the reference analyses to Cycle 15 is reviewed in this section.

Changes in fuel cycle design, fuel mechanical design and plant operation through Cycle 14 were reviewed and evaluated in Reference 40. The analyses reported in the above mentioned references were determined to be bounding for Cycle 14. No significant changes in operating conditions have been made for Cycle 15. Rated operating conditions are summarized in Table 8.1. Significant fuel and core thermal-hydraulic design parameters employed in the reference Chapter 15 event analyses are listed in Table 8.2. The following pellet characteristics were changed between the Cycle 14 reload fuel and the Cycle 15 reload fuel:

Pellet O.D.
Diametral Gap
Pellet Length
Pellet Density.

None of the changes in fuel design parameters will affect MDNBR's. The 0.0005 inch reduction in gap width will affect the hot rod gap conductance throughout a cycle, but will not change the relative DNBRs for the various non-LOCA events. The reduction in gap width will cause a reduction in initial stored energy and a lower PCT in the large break LOCA analysis. Changes in all of the above fuel parameters have been incorporated into the large break LOCA analysis for Cycle 15. The large break LOCA analysis for Cycle 15 is documented in a separate report[41].

Cycle 15 bounding physics parameters are compared with bounding values used in the reference analyses in Table 8.3. With the exception of the BOC value of the β/ℓ, the range of reference analysis values bounds Cycle 15 values. In the Reference 32 analysis, the impact of small changes in the value of β/ℓ was evaluated. The results from the Reference 32 analysis

demonstrated that slight changes in β/ℓ had a negligible impact on the calculation results. Therefore, for Cycle 15, the increase in the BOC value of β/ℓ from 304.3 sec^{-1} to 308.5 sec^{-1} is dispositioned not to impact the reference analyses results. Therefore, the reference analyses will remain bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ.

Cycle 15 will be the second cycle to employ HTP assemblies which include 6 HTP grid spacers and 3 IFMs. The Chapter 15 events have been reviewed relative to the Cycle 15 mixed core configuration. For the Cycle 15 analysis, MDNBR is calculated using the ANFP correlation. A 2% mixed core DNBR penalty is applied to the 95/95 correlation limit for the ANFP correlation. The 95/95 safety limit for the ANFP correlation is 1.177 (including the 2% mixed core DNBR penalty). The XNB correlation will not be used to calculate MDNBR, since the twice-burnt SMV fuel assemblies are not limiting with respect to MDNBR.

Table 8.4 includes bounding peaking factors supported by the reference analyses. The reference analyses support an $F_{\Delta H}$ of 1.65 and an F_Q of 2.32. The Cycle 15 analyses reported herein support an $F_{\Delta H}$ of 1.70 and an F_Q of 2.40 for all Chapter 15 events except for the small break LOCA. The reference small break LOCA analysis[28] supports an $F_{\Delta H}$ of 1.65 and an F_Q of 2.32. The large break LOCA/ECCS analyses[29,41] support an $F_{\Delta H}$ of 1.75 and an F_Q of 2.50. Reanalysis of the MDNBR for the bounding non-LOCA DNBR events was required for Cycle 15 because of the increase in the $F_{\Delta H}$ assumed for analytical purposes.

Verification of the overtemperature ΔT trip setpoint is documented in Reference 31 and is valid for Cycle 15. The validity of the current $f(\Delta I)$ trip reset function was verified for power distributions prototypic of Cycle 15. Verification of the overpower ΔT reactor trip reset function, $f(\Delta I)$, was documented in Reference 32 and remains valid for Cycle 15.

The DNB local peaking and rod bow requirements[33] were reviewed for Cycle 15. No rod bow penalties need be applied for assembly burnups equal to or less than the design burnup of 58,000 MWd/MTU.

The Chapter 15 events are reviewed on an individual basis below. Event numbering follows that employed in Reference 34. Reanalysis of the MDNBR for the limiting DNBR events supporting the overtemperature ΔT trip was required for Cycle 15 due to the increase in $F_{\Delta H}$ used in the analyses.

15.1.1 Decrease in Feedwater Temperature

This event was determined in Reference 24 to be bounded by the Increase in Steam Flow event (15.1.3). The basis for this determination was a comparison of rate and magnitude of thermal load increase resulting from the events. This event is not affected by cycle specific changes in physics parameters and fuel design. The disposition of this event for Cycle 15 is thus unchanged from the reference analysis.

15.1.2 Increase in Feedwater Flow

Two events were considered in Reference 24: 1) at full power, one steam generator feedwater regulating valve opens to full capacity; 2) at startup, while operating on the feedwater bypass system, a feedwater flow regulating valve opens to full capacity.

Subevent 1 was determined to be bounded by the Increase in Steam Flow event (15.1.3), based on a comparison of the magnitude of the thermal load increase resulting from the two events. This event is not affected by cycle specific changes in physics parameters and fuel design. The disposition for subevent 1 is thus unchanged from the reference analysis.

Subevent 2 was determined to be bounded by the Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition event (15.4.1). The basis for the determination was a comparison of reactivity insertion rates for the two events, computed for event 15.1.2 as the product of maximum primary system cooldown and a most negative EOC moderator temperature coefficient of -35 pcm/°F. The rate of cooldown is independent of cycle physics parameters and fuel design. The EOC MTC for Cycle 15 will remain more positive than -35 pcm/°F (Tables 6.1 and 8.3). The reactivity insertion rate employed in the Reference 24 evaluation of subevent 2 is therefore bounding of Cycle 15, and the disposition of the subevent

is unchanged from the reference analysis.

15.1.3 <u>Increase in Steam Flow</u>

The reference analysis for this event is documented in Reference 25. This event is controlled by the magnitude of the steam flow, the moderator and Doppler feedbacks, and the control bank worth. The reference analysis considered a range of physics parameters which bounds values expected for Cycle 15 (Tables 6.1 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference systems analysis will remain bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR calculation was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.382.

15.1.4 <u>Inadvertent Opening of Steam Generator Relief or Safety Valve</u>

Two events were considered in Reference 24: 1) opening of a PORV or safety valve power; and 2) opening of a PORV or safety valve after trip.

Subevent 1 was determined to be bounded by the Increase in Steam Flow event (15.1.3) based on a comparison of the magnitude of the thermal load increase resulting for the two events. This event is not affected by cycle specific changes in physics parameters and fuel design. The disposition for subevent 1 is thus unchanged from the reference analysis.

Subevent 2 was determined to be bounded by the Main Steam Line Break event (15.1.5) based on a comparison of the magnitude of the steam flow for the two events. This event is not affected by cycle specific changes in physics parameters and fuel design. The event 15.1.5 analyses[26,27] demonstrate that the acceptance criteria are satisfied.

15.1.5 Steam System Piping Failures Inside and Outside Containment

This event was analyzed in Reference 26 and in Reference 27 (with removal of or reduction of boron concentration in the boron injection tank). Event results are primarily dependent upon break steam flow, moderator and Doppler feedbacks, shutdown margin, and core power distribution. Moderator feedback and shutdown margin were set to bounding values in the reference analyses. The reference analyses employed a steam flow rate approximately 20% larger than that predicted by the conservative Moody critical flow model (saturated steam), leading to a significantly conservative prediction of moderator cooldown and thus an exaggerated peak power level. Power peaking is cycle specific: the similarity of the Cycle 10 and Cycle 15 cores would indicate only small changes in power peaking between the cycles. Substantial conservatism in the predicted steam flow and peak core power in the reference analyses ensure that cycle specific variations in power peaking are bounded. The reference analyses are thus applicable to Cycle 15.

15.2.1 Steam Pressure Regulator Failure

The plant has no main steam pressure regulators. This event is thus not applicable to H.B. Robinson Unit 2.

15.2.2 Loss of External Load

Two cases were analyzed in the reference analysis[25]; one to assess the challenge to vessel pressurization limits, and the other to evaluate the challenge to the MDNBR SAFDL. The peak pressurizer pressure of 2661 psia reported in Reference 25 is bounding for Cycle 15.

The MDNBR case was reanalyzed in Reference 31 to verify the OTΔT trip function with the RTD installation which eliminated the bypass piping. The reference analyses considered ranges of cycle physics parameters which bound Cycle 15 (Tables 6.1 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference systems analysis will remain bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR case was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to

1.258.

15.2.3 Turbine Trip; 15.2.4 Loss of Vacuum; 15.2.5 Closure of Main Steam Isolation Valve

These events are similar to the Loss of External Load event (15.2.2), which was analyzed[31] in a manner to bound the outcome of events 15.2.3, 15.2.4, and 15.2.5 as well. This disposition is independent of cycle-to-cycle variations and therefore applicable to Cycle 15.

15.2.6 Loss of Non-emergency A.C. Power to the Station Auxiliaries

The event considered in Reference 24 results in turbine trip with the consequent coastdown of primary coolant pumps and trip of main feedwater pumps.

The short term phase of this event, during which a challenge to the DNB SAFDL may develop, was determined in the reference analysis to be no worse than that occurring as a result of the Loss of Forced Reactor Coolant Flow (15.3.1). The basis for the disposition is that direct reactor trip will occur as a result of the event initiator in event 15.2.6. This basis is independent of cycle, and the disposition of the short term event for Cycle 15 is unchanged from the reference analysis.

In the longer term, during which a challenge to post-trip decay heat removal capability may develop, this event was determined to be bounded by the Loss of Normal Feedwater event (15.2.7). The basis for the event disposition is again the occurrence of an anticipatory trip for event 15.2.6, and therefore, also independent of cycle. The disposition of the longer term phase of event 15.2.6 is thus also unchanged from the reference analysis.

15.2.7 Loss of Normal Feedwater

The reference analysis for this event is documented in Reference 25. Two cases were treated: one with and one without forced primary coolant flow. Controlling variables are reactor decay heat and auxiliary feedwater flow capacity. The analysis is not affected by cycle specific changes in physics parameters and fuel design. Thus, the reference analysis is applicable to Cycle 15.

15.2.8 Feedwater System Pipe Breaks

This event was determined in Reference 24 to be bounded by the Main Steamline Break event (15.1.5). The basis for the disposition is a comparison of possible break flow areas. This event is not affected by cycle specific changes in physics parameters and fuel design. Thus, the reference analysis is applicable to Cycle 15.

15.3.1 Loss of Forced Reactor Coolant Flow

Coastdown of three primary coolant pumps was analyzed in Reference 25. Two cases were considered: one to assess the challenge to vessel pressurization limits, and a second to evaluate MDNBR. The reference analysis shows the pressurization case to be clearly bounded by the Loss of External Load event (15.2.2), a result not dependent on cycle specific parameters. Therefore, the reference pressurization case remains bounding for Cycle 15.

For the MDNBR case, the reference analysis considered ranges of cycle physics parameters which bound the Cycle 15 values (Tables 6.1, 8.2 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference systems analysis remains bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.535.

15.3.2 Flow Controller Malfunction

The plant has no primary coolant flow controllers. This event is thus not applicable to H.B. Robinson Unit 2.

15.3.3 Reactor Coolant Pump Rotor Seizure

The reference analysis for this event is documented in Reference 25. Two cases were considered: one to evaluate MDNBR, and a second to assess the challenge to vessel pressurization limits. The peak pressure reached in the pressurization case is well below that calculated for the Loss of External Load event (15.2.2), a result not dependent on cycle specific parameters. Therefore, the reference pressurization case remains bounding for Cycle 15.

The reference analyses considered ranges of cycle physics parameters which bound the Cycle 15 values (Tables 6.1, 8.2 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference system analysis remains bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR case was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.146. The percentage of the core calculated to experience DNB in the reference analysis is bounding for Cycle 15. The radiological consequences for this event are bounded by the Design Basis LOCA event which assumes all fuel assemblies in the core fail.

15.3.4 Reactor Coolant Pump Shaft Break

This event was determined to be bounded by the Reactor Coolant Pump Rotor Seizure event (15.3.3) in Reference 24. The basis for this determination is a comparison of flow dec rates. This event is not affected by cycle specific changes in physics parameters and fuel desig Thus, the reference analysis is applicable to Cycle 15.

15.4.1 Uncontrolled Control Rod Assembly Withdrawal from a Subcritical or Low Power Startup Condition

The reference analysis for this event is documented in Reference 25 for two pump operation and a bounding Control Rod Assembly bank worth. The reference analysis considered ranges of cycle physics parameters which bound the values for Cycle 15 (Tables 6.1, 8.2 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference systems analysis remains bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR calculation was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.488.

15.4.2 Uncontrolled Control Rod Assembly Withdrawal at Power

The reference analysis for this event is documented in Reference 25. The analysis w performed at a rated, mid and low power initial condition, BOC and EOC conditions, and for

spectrum of possible reactivity insertion rates.

Limiting low power cases may initiate with significantly higher radial power peaking than can actually exist at the core conditions reached at reactor trip. The transient reduction in radial power peaking which occurs in these events due to Doppler and moderator feedback is sufficient to offset the initially elevated radial peaking. Consideration of radial power peaking factors in excess of the rated power design limit is thus deemed unnecessary.

The limiting case was reanalyzed in Reference 31 to verify the OTΔT trip function with the RTD installation which eliminated the bypass piping. The limiting case was determined to be a rated power case at BOC conditions.

The Reference 31 analysis considered a range of physics parameters which bounds values expected for Cycle 15 (Tables 6.1 and 8.3) except for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on the calculation results. Therefore, the reference analysis remains bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The limiting case MDNBR was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.314.

15.4.3 <u>Control Rod Misoperation</u>

This event includes the single RCCA withdrawal, the static RCCA misalignment, and the Dropped RCCA or RCCA Bank events. These events were analyzed* for Cycle 10 in Reference 25 and are addressed individually for Cycle 15 below. Bounding radial peaking augmentation factors for these events in Cycle 15 are the same or smaller than those employed in the reference analyses (Table 8.4).

15.4.3.1 <u>Withdrawal of a Single Full Length RCCA</u>

The single RCCA withdrawal event was analyzed in Reference 25 by combining a radial peaking augmentation factor calculated for the event with core thermal-hydraulic boundary conditions calculated for the most limiting case of event 15.4.2. The MDNBR for this event was

reanalyzed for an increased $F_{\Delta H}$ of 1.70 and is equal to 0.595.

As in the reference analysis, the Cycle 15 MDNBR value for this event is less than the ANFP correlation limit. The extreme radial peaking calculated for the single RCCA withdrawal is localized in the neighborhood of the withdrawn RCCA. Less than 10% of the rods in the core are calculated to experience boiling transition.

The radiological consequences evaluation[42] for the Design Basis LOCA event assumed all fuel assemblies in the core failed. The LOCA results were less than 10% of the 10 CFR 100 limits. Using the conservative assumption that boiling transition results in fuel failure, less than the full core exhibited DNB and the integrity of the primary system is maintained; therefore, the radiological results of the withdrawal of a single full length RCCA event are bounded by those of the LOCA event.

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* Safety issues raised in Reference 35 and 36 are addressed in the reference analysis.

The single RCCA withdrawal event is classified as a postulated accident. The allowable limits are: (1) radiological release less than 10% of 10 CFR 100 limits and (2) reactor vessel pressurization below 110% of the design limit. It is not anticipated that core cooling would be significantly hindered by less than 10% fuel failures. No more limiting fault is engendered by the occurrence of the event. The results of the analysis is thus in conformance with the acceptance criteria and is therefore acceptable.

15.4.3.2 Static Misalignment of a Single Full Length RCCA

The static rod misalignment event results are cycle specific only through the radial peaking augmentation factor. The bounding value for the radial peaking augmentation factor of 1.17 calculated for the reference cycle bounds Cycle 15 (Table 8.3). However, the MDNBR of this event was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.493.

15.4.3.3 Dropped Full Length RCCA or RCCA Bank

The reference analysis for the limiting dropped RCCA case is documented in Reference 31. The analysis assumed that the automatic rod control system is restricted to rod insertion, that the K1 constant in the overtemperature ΔT trip algorithm is reduced from 1.26 to 1.24 and that the moderator temperature coefficient is restricted to be non-positive above fifty percent power. The reference analysis considered a range of physics parameters which bounds values expected for the BOC value of β/ℓ. In the Reference 32 analysis, it was demonstrated that small changes in the value of β/ℓ had a negligible impact on calculation results. Therefore, the reference systems analysis remains bounding for Cycle 15 with respect to the increase in the BOC value of β/ℓ. The MDNBR calculation was reanalyzed for Cycle 15 for an increased $F_{\Delta H}$ of 1.70 and is equal to 1.401.

The radial peaking augmentation factor of 1.36 for the dropped RCCA/dropped RCCA bank event reported in the Cycle 13 safety analysis[19] is bounding for Cycle 15 (Table 8.3) with automatic rod control restricted to rod insertion. The results of this event are bounded by the results of the dropped full length RCCA. Therefore, the MDNBR analysis for this event does not require reanalysis for Cycle 15 for an increased $F_{\Delta H}$. The availability of DNBR margin for the

dropped full length RCCA ensures that the acceptance criteria for this event are met.

15.4.4 Startup of an Inactive Loop at an Incorrect Temperature

Power operation with less than three loops in service is prohibited by Technical Specifications. Analysis of this event for H.B. Robinson Unit 2 is thus unnecessary.

15.4.5 Flow Controller Malfunction

H.B. Robinson Unit 2 plant has no primary loop isolation valves nor means to control primary flow. Therefore, this event is not applicable to H.B. Robinson Unit 2.

15.4.6 Chemical and Volume Control System Malfunction that Results in a Decrease in Boron Concentration in the Reactor Coolant

This event is dependent on dilution flow rate, mixed reactor coolant volume, the BOC boron worth coefficient, and the BOC critical boron concentration. Dilution flow rate and reac coolant volume are fixed system parameters independent of cycle. This event was reanalyzed in all modes of operation for Cycle 15 to assess the impact of updated BOC critical boron concentrations. The results of the Cycle 15 analysis are reported in Table 8.5. The criteria on time available from event initiation to loss of shutdown margin, 30 minutes for refueling and 15 minutes for all other modes, are met for all modes of operation. The analysis for the refueling mode is valid for a 6% shutdown margin requirement. The 6% shutdown margin requirement at refueling conditions is an updated value, and reflects a change in the H. B. Robinson Unit 2 Technical Specifications for Cycle 15 and beyond.

An analysis has been performed to confirm the Boric Acid Storage Tank (BAST) capacity to provide 1% shutdown margin. The major features and assumptions for the Cycle 15 BAST capacity analysis are summarized below.

1. The "no-letdown" Reactor Coolant System (RCS) mixing model was used to ensure an upper limit for the calculated BAST injection time and correspondi volume. Previous evaluations compared the "no-letdown" Reactor Coolant Syste

(RCS) mixing model to the "letdown=charging" model. The "no-letdown" model was found to result in slightly larger times (greater BAST volume) to reach a shutdown boron concentration.

2. Perfect (instantaneous) mixing of the RCS and charging masses is assumed.

3. A minimum RCS boron concentration is calculated for the reactor at hot full power (initial boron concentration).

4. An uncertainty in the calculated RCS boron concentration is used to conservatively bias the calculated injection time.

   a. The calculated minimum boron concentration for full power is conservatively decreased by 100 ppm.
   b. The calculated maximum boron concentration at cold shutdown is conservatively increased by 100 ppm.

The calculated initial and end-state boron concentrations for both BOC15 and EOC15 conditions, and the corresponding calculated injection times and BAST volumes are summarized in the following table.

H.B. Robinson Unit 2 Cycle 15 BAST Capacity				
	C_o (ppm)	C_{final} (ppm)	T_{final} (minutes)	BAST Volume (gallons)
BOC15	1040.	1248.	20.38	1223.
EOC15	0.	454.	26.55	1593.

In the above table, C_o is the minimum boron concentration at HFP; C_{final} is the maximum boron concentration for 1% $\Delta k/k$ at cold shutdown; and T_{final} is the time to reach final boron concentration.

The EOC case requires the largest injection time. This is due to the larger difference between the initial and final boron concentrations, i.e. 454 ppm (EOC) vs 208 ppm (BOC). The calculated BAST volumes are well within the current 2640 gallon Technical Specification value, thus validating the capability of the BAST to provide the necessary shutdown margin.

15.4.7 <u>Inadvertent Loading and Operation of the Fuel Assembly in an Improper Position</u>

The reference analysis for this event is documented in Reference 25. Reanalysis of this event for Cycle 15 indicated that a maximum undetected $F_{\Delta H}$ of 1.89 may occur. Since this value is less than the value of 1.94 treated in the reference analysis, the reference analysis result is bounding of Cycle 15.

15.4.8 <u>Spectrum of Rod Ejection Accidents (PWR)</u>

A Control Rod Ejection Accident is defined as the mechanical failure of a control r mechanism pressure housing, resulting in the ejection of a Rod Cluster Control Assembly (RCC 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 SNP Generic Rod Ejection Analysis, Reference 37. The ejected rod worths and hot pellet peaking factors were calculated using the XTGPWR code. No credit was taken for the power flattening effects of Doppler or moderator feedback in the calculation of ejected rod worths or resultant post-transient peaking factors. The pellet energy deposition resulting from an ejected rod was conservatively evaluated explicitly for BOC and EOC conditions, at HFP and HZP. Results of the Cycle 15 calculations are presented in Tables 8.6 and 8.7. These tables may be used as cycle-specific replacements for UFSAR Tables 15.4.8-1 and 15.4.8-2 respectively. The HFP pellet energy deposition was calculated to be 192 cal/gm at BOC, and 175 cal/gm at EOC. The HZP pellet energy deposition was calculated to be less than 32 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.

15.4.9 Spectrum of Rod Ejection Accidents (BWR)

This event is not applicable to pressurized water reactors.

15.5.1 Inadvertent Operation of the ECCS that Increases Reactor Coolant Inventory

The disposition of this event is documented in Reference 24. The disposition of this event is unchanged from the reference analysis.

15.5.2 Inadvertent Operation of Chemical and Volume Control System that Increases Reactor Coolant Inventory

The consequences of unplanned additions to inventory and effect of reactivity additions due to dilution during refueling and startup are treated in the analysis of event 15.4.6. The consequences of dilutions at power are bounded by the evaluation of event 15.4.2, Uncontrolled RCCA Bank Withdrawal at Power.

The consequences of volumetric addition and effect on the pressure boundary during all operational modes have been earlier addressed in Section 15.5 of the H.B. Robinson Unit 2 FSAR[38]. The evaluation is determined by pressurizer PORV capability, and is thus independent of cycle.

15.6.1 Inadvertent Opening of a Pressurizer Pressure Relief Valve

The reference analysis for this event is documented in Reference 24. The input parameters to the analysis were reviewed and found to be non-cycle specific. The MDNBR for this event was reanalyzed for Cycle 15 for an $F_{\Delta H}$ of 1.70 and is equal to 1.407.

15.6.2 Small Break Loss-of-Coolant Accidents

The small break loss-of-coolant accident is defined as a rupture of the reactor coolant pressure boundary with a total cross-sectional area less than 1.0 ft^2 in which the normally operating charging system flow is not sufficient to sustain pressurizer level and pressure. The small break loss-of-coolant accident event was analyzed and reported in Reference 28. The analysis considered break sizes of 1.0, 1.5, and 2.0 inches. The analysis also assumed that flow

from only one high head safety injection pump is available. The 1.5 inch break size was determined to be the limiting break size. The peak cladding temperature for the limiting break size was calculated to be 2095.6°F[39]. The analysis demonstrated the H.B. Robinson Unit 2 plant to be in compliance with the requirements of 10CFR50.46. The results of the analysis are not affected by cycle specific changes in physics parameters and fuel design. Therefore, the reference analysis is applicable to Cycle 15 with an $F_{\Delta H}$ of 1.65 and an F_Q limit of 2.32.

15.6.3 Radiological Consequences of Steam Generator Tube Failure

The reference analysis for this event is documented in Reference 25. Radioactive releases depend on the Technical Specification limits on primary coolant activity and the amount of coolant released. Cycle 15 primary coolant activity limits are unchanged from the reference analysis. Primary coolant release is dependent on primary coolant thermodynamic state and tube rupture area. No changes have occurred between the reference analysis and Cycle 15 which would change these controlling parameters. The reference analysis is thus applicable to Cycle 15.

15.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment

This event is not applicable to pressurized water reactors.

15.6.5 Loss-of-Coolant Accidents

Major loss-of-coolant accidents are defined as a rupture 1.0 ft^2 or larger of the Reactor Primary Coolant System (RCS) piping, including the double-ended rupture of the largest pipe in the RCS or of any line connected to that system up to the first closed valve. The large break LOCA analysis for H. B. Robinson Unit 2 is documented in References 29 and 41. The Reference 41 analysis included fuel design changes for Cycle 15. The Reference 41 analysis also included the effect of reduced LPSI flow to the RCS due to (1) a degraded pump head-flow curve, (2) changes in LPSI line losses, and (3) an increased LPSI pump recirculation flow rate. The analysis verified the loss of a diesel generator to be the limiting single failure. The limiting break size was determined in Reference 29 to be the 0.8 double-ended-cold-leg-guillotine break. The peak cladding temperature was calculated for three different axial power shapes. The maximum peak

cladding temperature was calculated to be 2146°F for the cosine axial power shape. The analysis demonstrated the H.B. Robinson Unit 2 plant to be in compliance with the requirements of 10CFR50.46. The analysis supports an F_Q of 2.50, an $F_{\Delta H}$ of 1.75, a 6% steam generator tube plugging level, and a maximum bundle average exposure of 52,500 MWd/MTU.

An evaluation of the Reference 29 analysis was performed for once-burnt and twice-burnt fuel relative to the reduced LPSI flow to the RCS mentioned above. It was determined that the reduced LPSI flow has a negligible effect on the PCTs reported in Reference 29. In addition, the once-burnt fuel has a significantly lower initial fuel stored energy than that assumed in the analysis since the fuel has completed a full cycle. Therefore, the LOCA results reported in Reference 29 are bounding and conservative for the once and twice-burnt fuel for Cycle 15. The Reference 29 analysis supports an F_Q of 2.32 and an $F_{\Delta H}$ of 1.65 for the once and twice-burnt fuel assemblies in Cycle 15.

The radioactivity released to the environment for the large break LOCA event (Reference 42) was shown to be well below the limits of 10CFR100.

An analysis has been performed to confirm that the post-LOCA containment sump boron concentration is sufficiently high to preclude core criticality. The features and assumptions for the Cycle 15 post-LOCA criticality analysis are summarized below.

1. The volume of the Boron Injection Tank and the piping external to the RCS have been conservatively neglected. The mass of water in these volumes is less than 1% of the post-LOCA containment sump inventory.

2. A minimum pre-LOCA RCS boron concentration is assumed for mixing with the other water sources in the containment sump. The calculated minimum pre-LOCA RCS boron concentration is conservatively reduced by 100 ppm to ensure a lower bound sump concentration.

3. A maximum post-LOCA critical boron concentration is calculated assuming ambient temperature and pressure and all rods out. The calculated critical boron concentration is conservatively increased by 100 ppm to ensure an upper bound boron concentration.

The post-LOCA sump boron concentration and post-LOCA maximum boron concentration are compared below.

Minimum containment sump boron concentration	=	1778 ppm
Maximum post-LOCA critical boron concentration + 100	=	1655 ppm

Since the minimum sump boron concentration is greater than the maximum post-LOCA critical boron concentration, criticality after the LOCA is not possible.

15.7.1, 15.7.2 (Events Deleted from the Standard Review Plan)

15.7.3 <u>Postulated Radioactive Releases due to Liquid-Containing Tank Failure</u>

This event has been reviewed and the results documented in the H.B. Robinson Unit 2 FSAR. This event is not affected by cycle specific changes. Thus, the reference analysis bounds Cycle 15.

15.7.4 <u>Radiological Consequences of Fuel Handling Accidents</u>

This event is characterized by the release of fission products from a single limiting fuel assembly during handling in the spent fuel pool. Fission product inventories are determined by exposure and power level. The reference analysis[30] employed a maximum design burnup of 44,000 MWd/MTU, a power level of 2,300 MWt, and a radial peaking factor of 1.65.

This event was reanalyzed for Cycle 15 to support a maximum design burnup of 52,500

MWd/MTU, a power level of 2300 MWt, and a radial peaking factor of 1.75. The Cycle 15 event results are reported in Reference 42. The analysis demonstrated that the whole body and thyroid doses received from the postulated loss-of-coolant accident, fuel handling accident, and other events with high burnup fuel were be well below the values prescribed by the Standard Review Plan.

15.7.5 Spent Fuel Cask Drop Accidents

The results of this accident are unchanged from those documented in the H.B. Robinson Unit 2 FSAR. The results of the analysis therefore bound Cycle 15.

TABLE 8.1 PLANT RATED OPERATING CONDITIONS

Core Thermal Power	2,300 MWt
Core Coolant Inlet Temperature	546.2°F
Core Coolant Average Temperature	575.4°F
Vessel Coolant Flow*	$100.3 * 10^6$ lb/hr
Steam Generator Pressure (Dome)	800 psia
Steam Flow	$10.1 * 10^6$ lb/hr
Feedwater Temperature	441.5°F

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* Coolant Flow reflects 6% steam generator tube plugging for rebuilt steam generator.

TABLE 8.2 CORE AND FUEL DESIGN PARAMETERS
USED IN REFERENCE CHAPTER 15 EVENT ANALYSES THROUGH CYCLE 14

Number of fuel assemblies of all types in core	157
Number of Part Length Shielding Assemblies	12
Fuel assembly pitch	8.466 in.
Fuel assembly design type	15x15
Fuel rods per assembly	204
Guide tubes per assembly	20
Instrument tubes per assembly	1
Fuel rod pitch	.563 in.
Fuel rod O.D.	.424 in.
Guide and instrument tube O.D. (above dashpot)	.544 in.
Active fuel length	144 in.
Fuel rod length	152 in.
Number of spacers	7
Maximum spacer span length	26.19 in.
Number of Intermediate Flow Mixers*	3
Maximum span length between spacers and IFMs*	13.32 in.

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* HTP assemblies only.

TABLE 8.3 COMPARISON OF BOUNDING PHYSICS PARAMETERS AND AUGMENTATION FACTORS FOR REFERENCE ANALYSES AND CYCLE 15

		Reference Analyses		Cycle 15	
		BOC	EOC	BOC	EOC
1.	Moderator Temperature Coefficient, 10^{-5} $\Delta\rho$/°F	+5.0*	-35.0	+5.0*	-35.0
2.	Moderator Pressure Coefficient, 10^{-6} $\Delta\rho$/psi	0.0	+3.9	0.0	+3.9
3.	Doppler Coefficient, 10^{-5} $\Delta\rho$/°F	-1.0	-1.7	-1.0	-1.7
4.	Delayed Neutron Fraction, β	0.0070	0.0045	0.0070	0.0045
5.	Rod Worth, .9 (N-1), $\Delta\rho$	0.036	0.036	0.036	0.036
6.	β/ℓ**, sec^{-1}	304.3	204.5	308.5	204.5
7.	U-238 Atoms Consumed per Total Atoms Fissioned	0.50	0.76	0.50	0.76
8.	$F_{\Delta H}$ Augmentation Used in Single RCCA Withdrawal Event	1.27		1.27	
9.	$F_{\Delta H}$ Augmentation Used in Static Misalignment Event	1.17		1.17	
10.	$F_{\Delta H}$ Augmentation Used in Rod/Bank Drop Analysis (Manual Rod Control)				
	Large Bank Worth	1.36		1.36	
	Small Bank Worth	1.10		1.10	

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* An MTC of 0.0 was used in the dropped RCCA event analysis. The Technical Specification for the MTC is $\leq$0.0 above 50% power.

** ℓ is the effective neutron lifetime.

TABLE 8.4 TECHNICAL SPECIFICATION PEAKING FACTORS SUPPORTED BY REFERENCE ANALYSES

Nuclear Enthalpy Rise Factor, $F_{\Delta H}$,* (100% rated power)	1.65
Total Heat Flux Peaking Factor*	2.32
Fraction of Power Deposited in Fuel	0.974

* Value supported by SBLOCA/ECCS event analysis, (Reference 28).

TABLE 8.5 H.B. ROBINSON UNIT 2 CYCLE 15 RESULTS OF THE ANALYSES OF CVCS MALFUNCTION

Reactor Conditions	Control Rod Bank Configuration	Initial Boron Concentration (ppm)	Final Boron Concentration (ppm)	Time to Loss of Shutdown Margin (min.)
Refueling	ARI	1,950	1,099	59.7
Cold Shutdown				
Case 1 (1 pump)	N-1	1,344	1,255	21.4
Case 2 (Closed RCS)	N-1	1,344	1,255	17.9
Hot Shutdown	N-1	1,317	1,225	18.4
Startup (HZP)	N-1	1,057	941	22.8
Power Operation			Bounded by analysis in Sections 15.4.1 and 15.4.2	

TABLE 8.6 H. B. ROBINSON UNIT 2, RESULTS OF THE CYCLE 15 EJECTED CONTROL ROD ANALYSIS, HFP

	BOC15		EOC15	
	VALUE	CONTRIBUTION TO ENERGY DEPOSITION (cal/gm)	VALUE	CONTRIBUTION TO ENERGY DEPOSITION (cal/gm)
Initial Fuel Enthalpy (cal/gm)	89.6	----	83.0	----
Generic Initial Fuel Enthalpy (cal/gm)	40.8	----	40.8	----
Delta Initial Fuel Enthalpy (cal/gm)	48.8	48.8	42.2	42.2
Maximum Control Rod Worth (pcm)	491.6	143.6	365.4	133.8
Doppler Coefficient (pcm/°F)	-1.14	0.98	-1.14	0.98
Delayed Neutron Fraction, β	0.0057	1.02	0.0057	1.02
Power Peaking Factor Used	4.00		4.00	
Total Fuel Enthalpy (cal/gm)		191.6		175.2

TABLE 8.7 H. B. ROBINSON UNIT 2, RESULTS OF THE CYCLE 15 EJECTED CONTROL ROD ANALYSIS, HZP

	BOC15		EOC15	
	VALUE	CONTRIBUTION TO ENERGY DEPOSITION (cal/gm)	VALUE	CONTRIBUTION TO ENERGY DEPOSITION (cal/gm)
Initial Fuel Enthalpy (cal/gm)	18.6	----	18.6	----
Generic Initial Fuel Enthalpy (cal/gm)	16.7	----	16.7	----
Delta Initial Fuel Enthalpy (cal/gm)	1.9	1.9	1.9	1.9
Maximum Control Rod Worth (pcm)	200.6	24.3	292.1	30.2
Doppler Coefficient (pcm/°F)	-1.14	0.91	-1.14	0.91
Delayed Neutron Fraction, β	0.0057	1.06	0.0057	1.06
Power Peaking Factor Used	6.00		9.00	
Total Fuel Enthalpy (cal/gm)		25.5		31.3

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SAFETY ANALYSIS REPORT

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