RELOAD SAFETY EVALUATION

BEAVER VALLEY NUCLEAR PLANT

UNIT 1 CYCLE 4

February, 1983

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0231L:6

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1.0 INTRODUCTION AND SUMMARY

1.1 INTRODUCTION

This report presents an evaluation for Beaver Valley Unit 1, Cycle 4, which demonstrates that the core reload will not adversely affect the safety of the plant. This evaluation was accomplished utilizing the methodology described in WCAP-9273, "Westinghouse Reload Safety Evaluation Methodology"[1].

Based upon the above referenced methodology, only those incidents analyzed and reported in the FSAR[2] which could potentially be affected by this fuel reload have been reviewed for the Cycle 4 design described herein. The justification for the applicability of previous results is provided.

1.2 GENERAL DESCRIPTION

The Beaver Valley Unit 1 reactor core is comprised of 157 fuel assemblies arranged in the core loading pattern configuration shown in Figure 1. During the Cycle 3/4 refueling, 52 fuel assemblies will be replaced with Region 6 fuel and one Region 1 assembly will be replaced with another Region 1 assembly. A summary of the Cycle 4 fuel inventory is given in Table 1.

A new Wet Annular Burnable Absorber (WABA) rod design may be utilized for Cycle 4. The WABA design provides significantly enhanced nuclear characteristics, when compared with the borosilicate absorber rod design. Fuel cycle benefits result from the reduced parasitic neutron absorption of Zircaloy compared to stainless steel tubes, increased water fraction in the burnable absorber cell, and a reduced boron penalty at the end of each cycle.

The materials, mechanical, thermal hydraulic, and nuclear design evaluations of the WABA rods are presented in WCAP-10021, Revision 1, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," Reference 3, which has been submitted to NRC for generic review and approval.

As in Cycle 3, this cycle will contain two Region 4 demonstration assemblies, designated in Figure 1 as 4A, of an optimized fuel assembly design. These assemblies will be loaded into the core in a manner that prevents them from becoming lead assemblies during normal operation or leading to more limiting conditions during transient conditions than analyzed for the standard fuel assemblies.

Nominal core design parameters utilized for Cycle 4 are as follows:

Core Power (MWt)	2652
System Pressure (psia)	2250
Core Inlet Temperature (°F)	542.5
Core Average Temperature (°F)	579.3
Thermal Design Flow (gpm)	265,500
Average Linear Power Density (kw/ft)	5.19

1.3 CONCLUSIONS

From the evaluation presented in this report, it is concluded that the Cycle 4 design does not cause the previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the following:

1. Cycle 3 burnup is between 9300 and 11300 MWD/MTU.

2. Cycle 4 burnup is limited to the end-of-life full power capability* plus a 1000 MWD/MTU power coastdown.

3. There is adherence to plant operating limitations given in the Technical Specifications.

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* Definition: With control rods fully withdrawn and approximately 0 to 10 ppm residual boron.

2.0 REACTOR DESIGN

2.1 MECHANICAL DESIGN

The mechanical design of the Region 6 fuel assemblies is the same as the Region 5 assemblies except that Region 6 assemblies incorporate: (1) the reconstitutable bottom nozzle design; (2) grid modifications consisting of additional guide vanes and tabs along with the corner surface condition already incorporated intc Region 5 fuel assemblies; and (3) fuel rod backfill pressure of 350 psig. Table 1 compares pertinent design parameters of the various fuel regions. The Region 6 fuel has been designed according to the fuel performance model in Reference 4. The fuel is designed to operate so that clad flattening will not occur as predicted by the Westinghouse model, Reference 5. The fuel rod internal pressure design basis, Reference 6, is satisfied for all fuel regions.

WABA rods may be used instead of the standard borosilicate glass absorber rods (see Figure 1). The WABA rod design consists of annular pellets of aluminum oxide-boron carbide (Al_2O_3-B_4C) burnable absorber material encapsulated within two concentric Zircaloy tubings. The reactor coolant flows inside the inner tubing and outside the outer tubing of the annular rod. Details of the WABA design are described in WCAP-10021, Revision 1, "Westinghouse Wet Annular Burnable Absorber Evaluation Report," Reference 3.

Westinghouse's experience with Zircaloy clad fuel, is described in WCAP-8183, "Operational Experience with Westinghouse Cores," Reference 7, which is updated annually.

2.2 NUCLEAR DESIGN

Two core designs are addressed in this report. One design utilizes all standard (borosilicate glass) burnable absorber rods and the other features Wet Annular Burnable Absorber (WABA) rods. The loading

pattern, identical for both designs as shown in Figure 1, contains 560 burnable absorber (BA) rods located in 44 BA rod assemblies. Fuel assemblies and BA assemblies are placed in identical locations for both designs, differing only in the type of burnable absorber used.

The Cycle 4 core loading is designed to meet a F_Q x P ECCS limit of $\leq$ 2.32 x K(z)* for a flux difference (ΔI) band width during normal operation conditions of $\pm 7\%\Delta I$.

Table 2 provides a summary of changes in Cycle 4 kinetic characteristics compared with the current limit based on previously submitted accident analyses. Table 2 is applicable for the Cycle 4 core using either the WABA design or the standard burnable absorber design.

Table 3A provides the control rod worths and requirements at the most limiting condition during the cycle (end-of-life) for the WABA design. Table 3B provides the same information for the standard burnable absorber design. The required shutdown margin is based on previously submitted accident analysis. The available shutdown margin for both burnable absorber designs exceeds the minimum required.

The secondary sources (with associated absorber rods) located in positions H3 and H13 in Cycle 3 will be removed. Secondary sources (without BAs) in positions J6 and G10 in Cycle 3 will be placed in H3 and H13 in Cycle 4.

2.3 THERMAL AND HYDRAULIC DESIGN

No significant variations in thermal margins will result from the Cycle 4 reload. The DNB core limits and safety analyses used for Cycle 4 are based on the conditions given in Section 1.0. The thermal hydraulic design considerations of the test assemblies are described in Reference 8.

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* K(z) - Figure 2

3.0 POWER CAPABILITY AND ACCIDENT EVALUATION

3.1 POWER CAPABILITY

The plant power capability has been evaluated considering the consequences of those incidents examined in the FSAR[2] using the previously accepted design basis. It is concluded that the core reload will not adversely affect the ability to safely operate at the design power level (Section 1) during Cycle 4. For the overpower transient, the fuel centerline temperature limit of 4700°F can be accommodated with margin in the Cycle 4 core. The time dependent densification model[9] was used for fuel temperature evaluations. The LOCA limit at rated power can be met by maintaining F_Q at or below 2.32 according to the normalized F_Q envelope (Figure 2).

3.2 ACCIDENT EVALUATION

The effects of the reload on the design basis and postulated incidents analyzed in the FSAR were examined. In all cases, it was found that the effects were accommodated within the conservatism of the initial assumptions used in the previous applicable safety analysis.

The new dropped rod methodology was instituted for this cycle as described in Reference 10. Formal notice to the NRC will be made relating: (1) the new dropped rod evaluation has been successfully applied to Cycle 4, (2) the interim restriction on rod control and insertion are no longer necessary, and (3) requesting that the NRC approve the removal of the interim operational restrictions. Plant operation should continue under the interim operating restrictions until explicit NRC approval is received.

A core reload can typically affect accident analysis input parameters in the following areas: core kinetic characteristics, control rod worths, and core peaking factors. Cycle 4 parameters in each of these three areas were examined as discussed in the following subsections to ascertain whether new accident analyses were required.

3.2.1 KINETICS PARAMETERS

Table 2 is a summary of the Cycle 4 kinetics parameters current limits. All the kinetic values fall within the bounds of the current limits; Table 2 is applicable to both Cycle 4 designs.

3.2.2 CONTROL ROD WORTHS

Changes in control rod worths may affect differential rod worths, shutdown margin, ejected rod worths, and trip reactivity. Table 2 shows that the maximum differential rod worth of two RCCA control banks moving together in their highest worth region for Cycle 4 meets the current limit. Tables 3A and 3B show that the Cycle 4 shutdown margin requirements are satisfied for both designs. Ejected rod worths for the Cycle 4 designs are also within the bounds of the current limits.

3.2.3 CORE PEAKING FACTORS

Evaluation of peaking factors for the rod out of position and dropped RCCA incidents show that DNBR is maintained above 1.3. A peaking factor evaluation for the hypothetical steamline break transient showed that the DNBR is maintained above 1.3. The peaking factors following control rod ejection are within the limits of previous analysis.

4.0 TECHNICAL SPECIFICATION CHANGES

No changes to the Beaver Valley Unit 1 Technical Specifictions are required for Cycle 4.

5.0 REFERENCES

1. Bordelon, F.M., et. al., "Westinghouse Reload Safety Evaluation Methodology", WCAP-9273, March 1978.

2. "Beaver Valley Unit No. 1 Final Safety Analysis Report", Docket Number 50-334.

3. Rahe, E. P., Westinghouse letter to C. Thomas, NRC, Number NS-EPR-2670, October 18, 1982, Subject: Westinghouse Wet Annular Burnable Absorber Evaluation Report," WCAP-10021, Revision 1, (Proprietary).

4. Miller, J.V., (Ed.), "Improved Analytical Model used in Westinghouse Fuel Rod Design Computations", WCAP-8785, October 1976.

5. George, R.A., (et. al.), "Revised Clad Flattening Model", WCAP-8381, July 1974.

6. Risher, D. H., (et. al.), "Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis," WCAP-8964, June 1977.

7. Jones, R. G., Iorii, J.A., "Operational Experience with Westinghouse Cores", WCAP-8183, Revision 11, May 1982.

8. O'Hara, T.L. (Ed.), "Optimized Fuel Assembly Demonstration Program", WCAP-9286, July 1978.

9. Hellman, J.M. (Ed.), "Fuel Densification Experimental Results and Model for Reactor Operation", WCAP-8219-A, March 1975.

10. Letter No. NS-EPR-2545, E. P. Rahe (Westinghouse) to C. H. Berlinger (NRC), January 20, 1982.

TABLE 1

BEAVER VALLEY UNIT 1 - CYCLE 4

FUEL ASSEMBLY DESIGN PARAMETERS**

Region	1	4	4A*	5	6
Enrichment (w/o U-235)+	2.106	3.199	3.203	2.999	3.25
Density(% Theoretical)+	94.80	94.38	94.38	94.34	94.5
Number of Assemblies	1	50	2	52	52
Approximate Burnup at Beginning of Cycle 4 (MWD/MTU)	13800	20700	21700	9200	0

* Optimized Fuel - Zirc grid

+ All fuel regions are as-built values except Region 6 which are nominal values.

**Applicable to either the WABA design or the standard burnable absorber design.

TABLE 2

KINETICS CHARACTERISTICS+

BEAVER VALLEY UNIT 1 - CYCLE 4

	Current Limit	Cycle 4 Changes to Current Limits
Moderator Density** Coefficient ($\Delta\rho$/gm/cc)	0 to 0.43	--
Doppler Temperature Cooefficient (pcm/^{o}F)*	-2.9 to -1.4	--
Least Negative Doppler - Only Power Coefficient, Zero to Full Power (pcm/% power)*	-6.68	--
Most Negative Doppler - Only Power Coefficient Zero to Full Power (pcm/% power)*	-19.4	--
Delayed Neutron Fraction β_{eff},(%)	0.44 to 0.75	--
Minimum Delayed Neutron Fraction Rod Ejection BOC β_{eff},(%)	0.52	--
		--
Rod Ejection EOC β_{eff},(%)	0.47	--
		--
Maximum Prompt Neutron Lifetime (μ sec)	26	--
Maximum Differential Rod Worth of Two Banks Moving Together (pcm/in.)*	100	--

*pcm = $10^{-5}\Delta\rho$

**The moderator density coefficient for the hot zero power, all rods out physics test condition may be negative at the BOC 4. The coefficient will be kept positive at that zero power by administrative controls (with appropriate D bank position and boron concentration).

+ Applicable to either the WABA design or the standard burnable absorber design.

--Indicates no change.

TABLE 3A

END-OF-CYCLE SHUTDOWN REQUIREMENTS AND MARGINS

BEAVER VALLEY UNIT 1 - CYCLE 4
WABA DESIGN

Control Rod Worth (%Δρ)	Cycle 3	Cycle 4
All Rods Inserted	8.14	8.49
All Rods Inserted Less Worst Stuck Rod	6.47	7.56
(1) Less 10%	5.82	6.80
Control Rod Requirements		
Reactivity Defects (Combined Doppler, T_{avg}, Void and Redistribution Effects)	2.86	2.98
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	3.36	3.48
Shutdown Margin [(1) - (2)] (%Δρ)	2.46	3.32
Required Shutdown Margin (%Δρ)	1.77	1.77

TABLE 3B

END-OF-CYCLE SHUTDOWN REQUIREMENTS AND MARGINS

BEAVER VALLEY UNIT 1 - CYCLE 4
STANDARD BURNABLE ABSORBER DESIGN

Control Rod Worth (%Δρ)	Cycle 3	Cycle 4
All Rods Inserted	8.14	8.59
All Rods Inserted Less Worst Stuck Rod	6.47	7.65
(1) Less 10%	5.82	6.89
Control Rod Requirements		
Reactivity Defects (Combined Doppler, T_{avg}, Void and Redistribution Effects)	2.86	3.06
Rod Insertion Allowance	0.50	0.50
(2) Total Requirements	3.36	3.56
Shutdown Margin [(1) - (2)] (%Δρ)	2.46	3.33
Required Shutdown Margin (%Δρ)	1.77	1.77

Figure 1

CORE LOADING PATTERN *

BEAVER VALLEY UNIT 1 CYCLE 4

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1							4	6	4						
2					4	6	6 (4)	4	6 (4)	6	4				
3				4	6	5 (12)	5	4 SS	5	5 (12)	6	4			
4			4	5	6 (12)	5	6 (20)	5	6 (20)	5	6 (12)	5	4		
5		4	6	6 (12)	4	4	5	5 (4)	5	4	4	6 (12)	6	4	
6		6	5 (12)	5	4	5	6 (20)	4	6 (20)	5	4	5	5 (12)	6	
7	4	6 (4)	5	6 (20)	5	6 (20)	4	5	4	6 (20)	5	6 (20)	5	6 (4)	4
8	6	4	4A	5	5 (4)	4	5	1	5	4	5 (4)	5	4A	4	6
9	4	6 (4)	5	6 (20)	5	6 (20)	4	5	4	6 (20)	5	6 (20)	5	6 (4)	4
10		6	5 (12)	5	4	5	6 (20)	4	6 (20)	5	4	5	5 (12)	6	
11		4	6	6 (12)	4	4	5	5 (4)	5	4	4	6 (12)	6	4	
12			4	5	6 (12)	5	6 (20)	5	6 (20)	5	6 (12)	5	4		
13				4	6	5 (12)	5	4 SS	5	5 (12)	6	4			
14					4	6	6 (4)	4	6 (4)	6	4				
15							4	6	4						

X - Region Number
(Y) - Number of Absorber Rods*
SS - Secondary Source Rods

*Applicable for both the WABA rod and the standard burnable absorber rod usage

FIGURE 2

K(Z) - NORMALIZED $F_Q(Z)$
AS A FUNCTION OF CORE HEIGHT

N-LOOP

BEAVER VALLEY - UNIT 1