Evaluation of Crystal River, Unit 3 Replacement Fuel Assembly 3A33

Background

Damage incurred to Crystal River, Unit 3 fuel assembly 3A33 during handling necessitated replacement with a new assembly. The replacement 3A33 assembly is made up of 40 fuel rods removed from the original assembly and 168 replacement fuel rods. On March 26, 1976, Babcock and Wilcox submitted a report (Reference 1), which described the replacement fuel assembly along with analyses performed by B&W on the potential for cladding creep collapse, fuel/clad interaction, fuel densification and fuel swelling. In response to requests by the NRC staff for additional information, this report was supplemented by a subsequent submittal on May 4, 1976 (Reference 2).

Description of Replacement Fuel Rods

The only significant differences between the 168 replacement fuel rods in replacement assembly 3A33 and the remaining fuel rods in the first cycle loading of Crystal River, Unit 3 relate to the internal spacers and the fuel pellets. Spring spacers and Zircaloy tubular spacers replaced the corrugated tube spacers and zirconia ceramic spacers in the original rods. The newer types of spacers are representative of those currently in operation in several B&W reactors and thus are not of concern. The fuel pellets, however, differ slightly in enrichment, density, and active length, as shown in Table I. Of particular interest to the Regulatory staff were the potential effects of the low-density, 90.9% theoretical density (TD), pellets on thermal and mechanical performance.

TABLE I

COMPARISON OF FUEL PARAMETERS

FUEL ASSEMBLY	NO. FUEL RODS	ENRICHMENT W/% U_{235}	%TD	STACK LENGTH, IN.	
3A33	40	1.93	92.5	144	
	168	1.98	95.35	23-1/2 (Upper Zone)	142-3/4
		1.94	90.9	95-3/4 (Central Zone)	
		1.98	95.35	23-1/2 (Lower Zone)	
Remaining Assemblies in Batch 1	208	1.93	92.5	144	

Summary of Regulatory Evaluation

Analyses of the thermal-hydraulic performance of the replacement fuel rods in replacement assembly 3A33 were performed by B&W and reported in the March 26, 1976, letter (Reference 1). A comparison of the results of these analyses to analyses of the 40 original fuel rods in assembly 3A33 and the fuel rods in the remaining 176 fuel assemblies is shown in Table II. The results of the analyses show that, except for the engineering hot channel factor, all thermal-hydraulic performance parameters for the replacement fuel rods in fuel assembly 3A33 are not more restrictive than for the fuel rods in the limiting fuel assembly in the remaining 176 fuel assemblies. The effects of the higher engineering hot channel factor for fuel assembly 3A33 were not addressed under "Fuel, Mechanical Design," but were treated under "Nuclear."

Cladding creep collapse analyses were performed by B&W in accordance with material properties and design procedures set forth in Topical Report B&W-10084P-A, entitled "Program to Determine In-Reactor Performance of B&W Fuels" (Reference 3). The evaluation was completed using the NRC - approved CROV creep ovalization analysis code described in Section 3 of the cited report. In addition other conservations were introduced, as described in reference 2. Results of the analyses indicated a collapse time > 14,000 hours, compared to the required 10,320 hours associated with the single cycle burn of assembly 3A33.

Pellet/cladding mechanical interaction (PCMI) and fuel swelling effects were addressed by B&W in their cladding strain analysis.

Of the pellet densities used in assembly 3A33, the 90.9% T.D. pellets represent the limiting PCMI case at the peak pellet burnup seen by the assembly. Accordingly, cladding strain analyses were performed on the 90.9% T.D. fuel corresponding to the worst-case specification dimensions and the as-built, 2σ dimensions. The analyses were performed in accordance with material data and design models set forth in Section 3 of Topical Report B&W-10054, Revision 2, entitled "Fuel Densification Report" (Reference 4). This represented the same approach as used in the Crystal River SAR except that additional conservations were introduced for the 3A33 analyses as listed in reference 2, p. 2. The results of the analyses indicated that the total circumferential strain resulting from PCMI for the worst-case-specification analysis and the as-built dimensions analysis were 0.80% and 0.48%, respectively, as compared to the B&W "design" value of 1.42%.

In summary, the mechanical design and thermal analysis aspects of the 168 replacement fuel rods in Crystal River, Unit 3 replacement fuel assembly 3A33 have been analyzed by B&W, using NRC-approved codes and methods and in accordance with material data and design models approved by NRC. Evaluations of the potential for cladding creep collapse, pellet/cladding mechanical interaction, fuel densification and fuel swelling were made. The results of these analyses have shown that the 3A33 fuel rods are within acceptable design limits for first cycle operation of Crystal River, Unit 3. The information provided on the results of the analyses of the 3A33 replacement fuel assembly provides an acceptable basis for demonstrating their adequacy.

References

1. J. T. Rodgers, Asst. V.P., B&W, to D. A. Butler, Chief, LWR Research #4, "Report on Replacement Assembly 3A33," March 26, 1976.

2. J. T. Rodgers to D. A. Butler, "Supplement to Report on Replacement Assembly 3A33," May 4, 1976.

3. A. F. J. Eckert, et al, "Program to Determine In-Reactor Performance of B&W Fuels," B&W-10084P-A, Nov. 1974.

4. R. A. Turner, "Fuel Densification Report," B&W-10054, Revision 2, May, 1973.

COMPARISON OF THERMAL-HYDRAULIC PARAMETERS

THERMAL-HYDRAULIC CRITERA	168 REPLACEMENT FUEL RODS IN FUEL ASSEMBLY 3A33	40 ORIGINAL FUEL RODS IN 3A33 AND FUEL RODS IN REMAINING 176 FUEL ASSEMBLIES
1. Linear Heat Rate Limit Based on Central Fuel Melting, KW/Ft.		
For Fuel Density:		
a. 95.35% TD	21.46	-
b. 90.9% TD	19.96	-
c. 92.5% TD	-	19.7
2. Average Linear Heat Rate, KW/Ft.	5.765	5.771
3. Average Fuel Temperatures (Stored Energy), F		
a. At Average Linear Heat Rate:		
(1) 95.35% TD	1285	-
(2) 90.9% TD	1327	-
(3) 92.5% TD	-	1335
b. At 18 KW/Ft.		
(1) 95.35% TD	28?0	-
(2) 90.9% TD	3066	-
(3) 92.5% TD	-	3110
4. Engineering Hot Channel Factor	1.026	1.014
5. DNBR Penalty Due to Fuel Densification,%*	1.9	2.9