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SUBJECT: Informs that util elected to reanalyze Batch 18 fuel w/ currently approved version of CROV,based on recent discussions w/NRC re not completing review of CROV 9.0. Revised pages pf Oconee Unit 3 Cycle 16 reload rept encl.

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DUKE POWER

March 13, 1995

U. S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
Unit 3 Cycle 16 Reload Technical Specifications
Supplement 3

By letter dated November 22, 1994, Duke Power submitted an amendment request necessary to support operation of Unit 3 at full rated power during Cycle 16. Attachment 4 of this letter is the Oconee 3 Cycle 16 Reload Report. Section 4.2.1 of the reload report states that the cladding creep collapse analyses for the fresh Batch 18 fuel is performed with the CROV 9.0 computer code. CROV 9.0 is currently being reviewed by the Staff.

Based on recent discussions with NRC Staff, it is our understanding that the review of CROV 9.0 may not be completed in time to support operation of Oconee 3 Cycle 16. Therefore, Duke Power has elected to reanalyze the Batch 18 fuel with the currently approved version of CROV. This change requires an increase to the planned pre-pressure of the Batch 18 fuel. The affected pages of the Oconee 3 Cycle 16 Reload Report have been revised to reflect these changes and are attached to this letter.

Please contact J. E. Burchfield at (803) 885-3292 if you have any questions.

Very truly yours,

J W Hampton

J. W. Hampton

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4. FUEL SYSTEM DESIGN

4.1 Fuel Assembly Mechanical Design

The types of fuel assemblies and pertinent fuel design parameters for Oconee 3 Cycle 16 are listed in Table 4-1. The Batch 18 feed assemblies are of the Mark-B10T design. The assembly design is the same as the previous Mk-B10 fuel deployed in Oconee 3 Cycle 15, although the fuel rod design has some significant changes. Changes to the fuel rod design include a larger diameter fuel pellet, a thinner cladding wall thickness, a smaller diametral pellet to clad gap, a change in the pellet dish design from a truncated cone to a spherical dish, and the addition of axial blankets.

4.2 Fuel Rod Design

The fuel assembly design for the Mk-B10T will be identical to the Mk-B10. The differences lie within the fuel rod design. These changes have been made to yield higher uranium utilization and improve fuel reliability. The Mk-B10T will contain larger diameter pellets, a smaller pellet to clad gap, a change in the pellet dish design from a truncated cone to a spherical dish, and the addition of axial blankets at an enriched weight of 2% U235. These items, either in whole or in part, will contribute to a higher uranium utilization and improved fuel reliability. The mechanical evaluation of the Mk-B10T design is discussed in this section.

4.2.1 Cladding Collapse

The CROV computer code and the procedures described in topical report BAW-10084P-A, Rev. 2[11] were used to calculate the creep collapse time for each batch of fuel. The CROV analyses are performed either in a generic fashion or on a batch specific basis, depending on available margins. The TACO2[4] code was used to calculate internal pin pressures and clad temperatures used as input to CROV. As shown in Table 4-1, the collapse times for the most limiting batches were conservatively determined to be greater than the maximum incore residence times.

4.2.2 Cladding Stress

As described in Reference 8, Duke has performed a generic and conservative fuel rod cladding stress analysis in accordance with the guidelines set forth in Section III, Division 1 - Subsection NB, of the ASME Boiler and Pressure Vessel Code. All methods are consistent with Reference 8. Compliance with ASME Code criteria verifies the structural integrity of the cladding throughout the most limiting design conditions.

The following conservatisms exist in the generic cladding stress calculation:

- High external cladding pressure (110% of design system pressure)
- Low internal pressure (HZP - min. specified pre-pressure)
- Maximum possible radial temperature gradient through clad (fuel melt conditions)
- Conservative cladding dimensions with regard to stress

Duke's reload analysis of the Mk-B10T fuel design demonstrates that the generic analysis is bounding.

4.2.3 Cladding Strain

Duke has performed a cladding strain calculation using TACO2[4] in accordance with the approved methodology[8]. This analysis demonstrated that the uniform, circumferential strain of the cladding was within 1.0%.

Table 4-1

Fuel Design Parameters and Dimensions

	Batch No.			
	15C	16B	17	18
FA type	Mark B8	Mark B9	Mark B10	Mark B10T
No. of FAs	5	52	60	60
Fuel rod OD, in.	0.430	0.430	0.430	0.430
Fuel rod ID, in.	0.377	0.377	0.377	0.380
Flex spacers, type	Spring	Spring	Spring	Spring
Rigid spacers, type	None	None	None	None
Undensified active fuel length, in.	141.8	140.6	140.6	142.29
Fuel pellet OD (mean spec), in.	0.3686	0.3700	0.3700	0.3735
Fuel pellet initial density (mean spec), %TD	95.0	95.0	95.0	96.0
Initial fuel enrichment, wt % ^{235}U	3.45	3.55	3.60	3.36 2.00*
Est. residence time, EOC 15, Hours	42,552	32,400	21,720	10,920
Cladding collapse time, Hours	57,000	55,000	51,600	36,894

* Enrichment % for Axial Blanketed Pellets.

Table 4-2. Linear Heat Rate to Melt Analysis

	Batch No.			
	15C	16B	17	18
Nominal initial density, % TD	95.0	95.0	95.0	96.0
Nominal initial pellet diameter, in.	0.3686	0.3700	0.3700	.3735
Nominal initial clad ID, in.	0.377	0.377	0.377	0.3800
Nominal initial clad OD, in.	0.430	0.430	0.430	0.430
Average linear heat rate @ 100% of 2568 MW, kW/ft	5.74	5.79	5.79	5.73
Linear heat rate capability from 0-1000 MWD/MTU, kW/ft	20.15	20.73	20.73	21.58
Linear heat rate capability >1000 MWD/MTU, kW/ft	21.20	21.71	21.71	21.23

REFERENCES

1. Oconee Nuclear Station, Units 1, 2, and 3 Final Safety Analysis Report, Docket Nos. 50-269, 50-270, and 50-287.

2. Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003A, Duke Power Company, Charlotte, NC, July 1989.

3. Oconee Unit 3, Cycle 15 - Reload Report, DPC-RD-2023, Duke Power Company, February 1994.

4. TACO2 - Fuel Performance Analysis, BAW-10141P-A, Rev. 1, Babcock & Wilcox, June 1983.

5. Letter from A. C. Thadani (NRC) to C. H. Turk (B&WOG), Acceptance for Referencing Topical Report BAW-1915 "Bounding Analytical Assessment of NUREG-0630 Models on kW/ft Limits with the use of FLECSET", October 12, 1987.

6. Oconee 1 Fuel Densification Report, BAW-1388, Rev. 1, Babcock & Wilcox, July 1973.

7. ~~TACO3 - Fuel Performance Analysis, BAW-10162P-A, , Babcock & Wilcox, November, 1989.~~

8. Oconee Nuclear Station Reload Design Methodology II, DPC-NE-1002A, Duke Power Company, Charlotte, North Carolina, October 1985.

9. Correlation of 15x15 Geometry Zircaloy Grid Rod Bundle CHF Data with the BWC Correlation, BAW-10143, Part 2, Babcock & Wilcox, Lynchburg, Virginia, March 1980.

10. 51-1232718-00, LOCA Evaluation of Mk-B9 Axial Blankets - Oconee 2, Babcock & Wilcox, August, 1994.

11. Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse, BAW-10084P-A, Rev. 2, Babcock & Wilcox Co., Lynchburg, Virginia, October 1978.

12. Letter from H. B. Tucker (Duke Power Company) to J. F. Stolz (NRC), April 23, 1986.