UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 47 TO LICENSE NO. DPR-38

AMENDMENT NO. 47 TO LICENSE NO. DPR-47

AMENDMENT NO. 44 TO LICENSE NO. DPR-55

DUKE POWER COMPANY

OCONEE NUCLEAR STATION, UNITS 1, 2 AND 3

DOCKET NOS. 50-269, 50-270 AND 50-287

Introduction

By letter dated March 30, 1977(1) and as supplemented June 21,(2) August 23,(3) September 8 (13) and 14,(14) 1977, Duke Power Company (the licensee) requested changes to the Technical Specifications appended to the Oconee Unit 1 Operating License for operation as reloaded for Cycle 4.

Evaluation

The Oconee Unit 1 reactor core consists of 177 fuel assemblies. The reload for Cycle 4 will involve the removal of all 56 Batch 3 fuel assemblies and 4 of the Batch 4 fuel assemblies, and relocation of the residual Batch 4 and Batch 5 fuel assemblies. The removed fuel will be replaced by 56 new Batch 6 fuel assemblies and 4 Batch 2 fuel assemblies. The new assemblies will occupy the periphery of the core.

The licensee's reload analyses and Technical Specification changes submitted by letter dated March 30, 1977, were based on an originally planned 292 effective full power days (EFPD) of Oconee Unit 1 Cycle 3 operation. The licensee, however, advised us by letter dated July 27, 1977,(4) that Cycle 3 operation was being extended to 312 EFPD. As a result, the burnup distribution in the Batch 4 and 5 fuel assemblies, which are to remain in the core for Cycle 4 operation, will be different from that assumed in the original reload analysis. Based on a reanalysis of the new burnup distribution the licensee submitted revisions to the reload report and Technical Specifications.(3)

Fuel Mechanical Design

Tables 4-1 and 4-2 of Reference 5 summarize the reload core fuel assembly parameters. The Batch 6, 15 x 15 (Mark B-4), fuel assembly design and the Batch 2, 15 x 15 (Mark B-2), fuel assembly design have been previously

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reviewed and accepted by us for use in Oconee Unit 1.[6] Also, these types of assemblies are currently operating in Oconee Unit 1. The reload assemblies, therefore, do not represent any unreviewed change in mechanical design from the reference cycle.

The reload fuel assemblies (Batch 6) are the same as the residual fuel assemblies except for minor design modifications to the spacer grid corner cells, which reduce spacergrid interation during handling. Dynamic impact testing has shown this design to have a higher seismic capability than the previous design.[1] The current design and the reload design meet all requirements of the fuel assembly design and are acceptable.

These mechanical design variations have been taken into account in the various mechanical analyses. The Batch 4 fuel is generally limiting, because of its relatively low initial fuel pellet density, and previous incore exposure. The results of these analyses have shown that the mechanical design differences between Cycle 3 and Cycle 4 are of negligible effect and are acceptable.

Fuel rod cladding creep collapse analyses were performed for the fuel batches which will be present in the Cycle 4 core. The calculational methods, assumptions, and data have been previously reviewed and approved by us.[7] The CROV computer code was used to calculate the time to fuel rod cladding creep collapse. The most restrictive power profiles, to which the once-burned and new fuel assemblies may be exposed, were used in the Batch 5, Batch 2, and Batch 6 analyses. The actual reactor operating history along with the most restrictive power histories for the forthcoming cycle were used in the analysis of the Batch 4 fuel. The fuel cladding material properties are the same as those used in the CROV code. The analysis assumed no fission gas production (maximum differential pressure), lower tolerance limit on cladding thickness, and upper tolerance limit on cladding ovality. Based on the analyses performed, the fuel rod design has been shown to meet the required design life limits for fuel cladding creep collapse and is, therefore, acceptable.

From the viewpoint of cladding stress and strain, Cycle 4 operation is acceptable. The cladding stress (creep stress due to differential pressure, thermal stress due to temperature gradient and bending stress due to axial loads and restraints) will not exceed the yield stress or ultimate strength of the cladding material. The Batch 4 fuel is most limiting with respect to stress, because of its irradiation history and lower fuel pellet density. The cladding strain for Cycle 4 operation is less than the generally used 1% plastic strain acceptance criteria. The strain analysis assumed maximum specification values for fuel pellet diameter, density, and burnup, and minimum specification tolerance on fuel cladding inside diameter. These assumptions conservatively represent the cladding strain. The Batch 4 fuel will again be limiting in the Cycle 4 core based on the cladding strain. Again this is because of its irradiation history and lower fuel pellet density.

The Batch 6 and Batch 2 fuel assemblies are not new in concept and do not use different component materials. The fuel assemblies for Cycle 4 operation will not exceed any design life limits. We conclude, therefore, that the fuel mechanical design for Cycle 4 operation is acceptable.

Fuel Thermal Design

The fuel thermal design analysis was conducted using the TAFY-3 computer code.[8] This analysis established heat flux limits to fuel centerline melt. The analysis considered the effect of a power spike from fuel pellet densification.[9] Modifications to the void probability, Fg, and size distribution, F_k, have been previously reviewed and approved by us for Oconee Unit 1 fuel thermal design analysis.[15] This analysis is based on the lower tolerance limit on fuel density and assumes isotropic diametral densification shrinkage and anisotropic axial shrinkage densification. These assumptions have been approved by us.[10]

During Cycle 4 operation, the highest relative assembly power levels occur in Batch 5 fuel assemblies. The fuel temperature analysis for Cycle 4 is based on limiting beginning-of-cycle (BOC) conditions (zero burnup) and conservative peaking factors. The analysis is performed to establish linear heat generation rates to preclude central fuel melting and stored energy limits for LOCA analyses. The thermal design analysis for the Batch 5 fuel assemblies thermal design analysis is bounding, and we conclude that the fuel thermal design for Oconee Unit 1 Cycle 4 is acceptable.

Nuclear Analysis

The reactor core physics parameters for Oconee Unit 1 Cycle 4 operation were calculated using a PDQ07 computer code. Since the core has not yet reached an equilibrium cycle, there were minor differences in the physics parameters between the Cycle 3 and Cycle 4 cores. For example, EOC Doppler and moderator coefficients change by less than 2% from Cycle 3 to Cycle 4. These changes are to be expected and are not significant.

Axial Power Shaping Rod (APSR) Testing

By letter dated June 21, 1977, a program was prepared to remove one of the axial power shaping rod (APSR) assemblies for destructive examination to obtain more information on the effects of irradiation on the material properties. Since this assembly has been exposed to three cycles of irradiation, it will be replaced with an APSR assembly with an equivalent poison worth. An evaluation of the nuclear, mechanical and thermal hydraulic considerations of this program have been conducted and it is concluded that safe operation of the reactor will not be adversely affected by this program .

In view of the above and the fact that startup tests (to be conducted prior to power operation) will verify that the significant aspects of the core performance are within the assumptions of the safety analysis, we find the licensee's nuclear analysis for Cycle 4 to be acceptable.

Thermal-Hydraulic Analysis

The major acceptance criteria which are used for the thermal-hydraulic design are specified in Standard Review Plan (SRP) 4.4. These criteria establish acceptable limits on departure from nucleate boiling (DNB). The thermal-hydraulic analysis for Oconee Unit 1 Cycle 4 reload was made using previously approved models and methods. Certain aspects of the thermal-hydraulic design are new for the Cycle 4 core and are discussed below.

Reactor Coolant System Flow Rate

The reactor coolant flow rate was accurately measured during Cycle 1 operation at 108.6% of the system design flow. The licensee has proposed to take credit in the thermal-hydraulic analysis for this higher flow (as was done in the previous cycle).

The core configuration for Cycle 4 differs slightly from that of Cycle 3 in that the depleted batch 3 fuel removed at the end of Cycle 3 is the Mark B-2 fuel assembly design. Mark B-4 fuel assemblies exhibit a slightly lower resistance to flow than do the Mark B-2 assemblies, which have a revised end fitting design. This change has been considered in the Cycle 4 core flow distribution analysis. No credit has been taken for the increase in system flow that will result from the reduction in total core pressure drop.

Fuel Rod Bow

In the submittal dated March 30, 1977, the licensee summarized the method and results of the rod bow analysis. This rod bow analysis was performed with an as yet unapproved model. Therefore, the licensee was requested to provide an analysis with the NRC approved rod bow model or to show sufficient compensatory margin.

The licensee chose to show sufficient margin in order to offset the difference between models. The approved rod bow model requires a DNBR penalty of approximately 12% as compared to the unapproved rod bow model which has about a 6% DNBR penalty. The 6% difference in DNBR penalty will be accommodated by a change in the protective pump monitor trip function by tripping the reactor upon loss of one pump during four pump operation if the indicated reactor power is greater than 80% of full power.

The existing pump monitor trip function is set to trip the reactor upon loss of two pumps, and as such the analytic basis for the existing flux/flow trip setpoint is a two-pump coastdown. By adding the pump monitor setpoint trip on the indicated loss of one pump, the flux/flow trip setpoint analysis need only consider a one-pump coastdown while still providing the same protection for loss of two pumps at lower power. The proposed change provides DNBR margins of 30% for Oconee Unit 1, 34% for Oconee Unit 2, and 32% for Oconee Unit 3.

In the case of Oconee Unit 2 Cycle 3 and Oconee Unit 3 Cycle 3, the necessary DNBR margins for the flux/flow trip setpoints were demonstrated by taking credit for additional RC flow available over the thermal-hydraulic design flow. Because of the pump monitor trip on loss of one pump, these flow credits are not required.

In summary, a reactor coolant flow rate based on actual measured flow with uncertainties was used in the Oconee Unit 1 Cycle 4 thermal hydraulic analysis. The licensee has also assured us that there will be sufficient margin in the reactor protective trip function to compensate for the difference between the approved and the unapproved rod bow models. Based on our review, we find that the licensee has included appropriate conservatisms in its analysis and that the proposed Technical Specifications provide assurance that the criteria of SRP 4.4 will be met. Therefore, we conclude that the thermal hydraulic analyses as previously approved and discussed are acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by the licensee demonstrate that the Oconee FSAR analyses conservatively bound the predicted conditions of the Oconee Unit 1 Cycle 4 core and are, therefore, acceptable. Each FSAR accident analysis has been examined, with respect to changes in Cycle 4 parameters, to determine the effects of the reload and to ensure that performance is not degraded during hypothetical transients. The core thermal parameters used in the FSAR accident analysis were design operating values based on calculated values plus uncertainties. FSAR values of core thermal parameters were compared with those used in the Cycle 4 analysis. For each accident of the FSAR, a discussion and the key parameters from the FSAR and Cycle 4 was provided with the accident discussion to show that the initial conditions of the transient are bounded by the FSAR analysis. The effects of fuel densification on the FSAR accident results have been evaluated and are reported in the Oconee Unit 2 fuel densification report[11] Since Cycle 4 reload fuel assemblies contain fuel rods with theoretical density higher than those considered there, the conclusions derived in that report are valid for Oconee Unit 1 Cycle 4. Calculational techniques and methods for Cycle 4 analyses remain consistent with those used for the FSAR. No new dose calculations were performed for the reload report. The dose considerations in the FSAR are based on maximum peaking and burnup for all core cycles; therefore, the dose considerations are independent of the reload batch.

A review of the ECCS U-baffle pressure drop error has been performed and documented in reference 12. The review considered a reanalysis of the reactor coolant system pressure loss characteristics and the effects and ECCS performance. The review found the current ECCS performance analysis acceptable for all three Oconee units. Reference 12 also found that a new surveillance testing program of the reactor internals vent valves is acceptable for all three Oconee units. The review considered the impact of these changes on ECCS performance and the adequacy of the surveillance techniques.

Startup Tests

A startup program will be conducted to verify that the core performance is within the assumptions of the safety analyses and provide the necessary data for continued plant operation. The startup test program is similar to that previously approved for Cycle 3 operation.[6] Additionally, the program was discussed with the licensee for clarification of control rod worth and power distribution measurements and comparison to predicted values. These measurements and comparisons will be performed by the licensee. Within 90 days following completion of physics testing the licensee also will provide a summary of the test program results. This startup test program is acceptable.

Environmental Consideration

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: October 4, 1977

REFERENCES

1. Letter from W. O. Parker, Jr., (Duke Power Company) to B. C. Rusche (NRC) dated March 30, 1977.

2. Letter from W. O. Parker,Jr., (Duke Power Company) to E. G. Case (NRC) dated June 21, 1977, Re: Oconee Unit 1 Docket No. 50-269

3. Letter from W. O. Parker, Jr., (Duke Power Company) to B. C. Rusche, dated August 23, 1977.

4. Letter from W. O. Parker, Jr., (Duke Power Company) to E. G. Case (NRC) dated July 27, 1977.

5. "Oconee Unit 1, Cycle 4-Reload Report", BAW-1447, March 1977.

6. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 20, 20 and 17 to Facility License Nos. DPR-38, DPR-47 and DPR-55, Duke Power Company, Oconee Nuclear Station, Unit Nos. 1, 2 and 3, March 25, 1976.

7. Letter from A. Schwencer (NRC) to J. F. Mallary (B&W) dated January 29, 1975.

8. TAFY-Fuel Pin Temperature and Gas Analysis," BAW-10044, May 1972.

9. "Fuel Densification Report", BAW-10055, Revision 1, June 1973.

10. "Technical Report on Densification of Babcock & Wilcox Reactor Fuels", NRR, July 6, 1973.

11. Oconee 2 Fuel Densification Report, BAW-1395, Babcock & Wilcox, June 1973.

12. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment Nos. 45, 45 and 42 to Facility License Nos. DPR-38, DPR-47 and DPR-55, Duke Power Company, Oconee Nuclear Station Unit Nos. 1, 2 and 3, July 29, 1977.

13. Letter from W. O. Parker, Jr., (Duke Power Company) to E. G. Case (NRC) dated September 8, 1977.

14. Letter from W. O. Parker, Jr., (Duke Power Company) to E.G. Case (NRC) dated September 14, 1977.

15. Letter from S. A. Varga (NRC) to J. H. Taylor (B&W), Subject: Evaluation of BAW-10083P, Revision 1, dated May 16, 1977.