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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT NO. 1

DOCKET NO. 50-313

1.0 Introduction

By letter dated December 28, 1978 (Reference 1), as supplemented by letters dated January 17, and 30, 1978, and March 3, 1978 (References 2, 3 and 4, respectively), the Arkansas Power and Light Company (AP&L or the licensee) requested an amendment to Facility Operating License No. DPR-51. The amendment would modify the Technical Specifications for Arkansas Nuclear One, Unit No. 1 (ANO-1) for Cycle 3 operation.

2.0 Evaluation

The ANO-1 reactor core consists of 177 fueled assemblies, each containing a 15x15 array of fuel rods. Each 15x15 array contains 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube.

For Cycle 3 operations all Batch 2 assemblies will be discharged from the core. Five once-burned Batch 1 fuel assemblies will be reloaded into the center of the core. Sixty (60) Batch 3 assemblies and 56 Batch 4 assemblies will be shuffled into new locations. Fifty-six (56) Batch 5 fresh assemblies will occupy the core periphery and eight interior locations. Tables 4-1 and 4-2 of Reference 1 contain summaries of fuel design parameters, dimensions and thermal analysis parameters for the fuel batches which will be burned in Cycle 3.

Reactivity control will be supplied by 61 full length Az-In-Cd control rods and soluble boron shim. In addition, eight (8) partial length axial power shaping rods (APSRs) are provided for control of the axial power distributions. Control rod interchanges or burnable poison rods are unnecessary for Cycle 3 operation.

2.1 Fuel Mechanical Design

The Batch 5 fresh fuel uses the Mark B-4 fuel assembly design which was initially used in Batch 4 during Cycle 2. The reload fuel assemblies incorporate minor changes in the spacer grid corner cells which reduce spacer grid interaction during handling. Additionally, dynamic impact testing has shown that the spacer grids have a higher seismic capability and consequently an increased safety margin over the values reported in Reference 5. The dynamic impact testing techniques are described in Reference 6.

Creep collapse time was calculated to be in excess of 30,000 effective full power hours (EFPH) which is longer than the projected three cycle exposure of 25,584 EFPH. The calculation of creep collapse time was performed using the power history of the limiting fuel assembly. As was done in Cycle 2, the CROV computer code was used to predict the collapse time (Reference 7). The licensee stated (Reference 8) that the CROV code demonstrated its ability to conservatively predict cladding collapse.

Additional conservatisms used in the CROV calculations were that no credit was taken for fission gas release; the cladding thickness used in CROV was the lower tolerance limit (LTL) of the as-built measurements; and the lowest as-fabricated pellet densities were assumed to be located in the worst case power region of the core.

The fuel clad strain analysis was performed using a number of conservative assumptions: maximum allowable fuel pellet diameter and density, lowest permitted tolerance for the cladding inner diameter, conservatively high local pellet burnup, and conservatively high heat generation rate. This insures that the 1.0% limit on cladding plastic circumferential strain is not violated.

The Batch 5 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previous successful operations with equivalent fuel, we conclude that the fuel mechanical design for Cycle 3 operations is acceptable and does not decrease the safety margin.

2.2 Fuel Thermal Design

The Batch 5 fuel produces no significant differences in fuel thermal performance relative to the other fuel remaining in the core. As was done in the Cycle 2 reload calculations, the linear heat rate (LHR) capability of ANO-1 was calculated using the TAFY computer code (Reference 9). The nominal LHR for Cycle 3 varies from a value of 5.77 for the Batch 1 fuel to 5.80 for the Batch 5 fuel. The LHR capability varies from 19.40 for Batch 3 to 20.15 for Batches 4 and 5.

The densification power spike model for Cycle 3 used the conservative combination of initial density and enrichment to calculate the spike factor. The power spike model is the same as that presented in Reference 10 with modifications to F_g and F_k. These changes reflect additional data from operating reactors which support a different approach and yield less severe penalties due to power spikes. Based on the analyses presented in Reference 1 and comparison with the allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations (Reference 11), the fuel thermal design for the ANO-1 Cycle 3 core is acceptable and does not decrease the safety margin.

2.3 Fuel Material Design

Cycle 3 fuel for ANO-1 will not have any significant material changes from previous cycles. Batch 4 started the use of a Zircaloy-4 (Zy-4) spacer material rather than Zirconium dioxide (ZrO_2) material. The use of Zy-4 spacer material is continued in Batch 5 assemblies. It was concluded in Reference 12 that the change from ZrO_2 to Zy-4 does not affect the primary coolant system chemistry. Therefore, the fuel material design for ANO-1 Cycle 3 operations is acceptable.

2.4 Nuclear Analysis

Physics parameters were calculated for the ANO-1 Cycle 3 core. There are minor differences between Cycle 3 and the Cycle 2 reference cycle physics parameters since Cycle 3 is not yet an equilibrium cycle. However, the differences in these parameters are minor.

The licensee requested a change in the ANO-1 Technical Specification regarding the correction of the hot zero power (HZP) measured moderator temperature coefficient (MTC) to compare with the 95% power Technical Specification limit (Reference 2). The proposed change would allow the use of cycle dependent parameters measured in the physics startup testing to project or extrapolate the 95% power value. The current Technical Specification requires a Technical Specification change each cycle because the cycle dependent corrections to the MTC at HZP are explicitly stated in the Technical Specification. We find that this approach will eliminate an unnecessary administrative step and is therefore acceptable.

The licensee also proposed a change in the plant Technical Specifications increasing the allowable quadrant tilt from 3.4% to 4.92%. The additional peaking allowed is a result of the statistical combination of the nuclear uncertainty factor, the hot channel factor, and the rod bow peaking penalty. We find that this Technical Specification is acceptable and does not decrease the safety margin.

The only significant proposed operational procedure change is the proposed Technical Specification change of the axial power shaping rod (APSR) position limits. The APSR position limits would provide added control of power peaking to insure that peak power limits for Loss of Coolant Accident (LOCA) conditions would not be violated.

We find that, based on the AP&L's nuclear analysis techniques and their commitment to perform acceptable physics startup testing, the ANO-1 nuclear analysis is acceptable. We also find the proposed Technical Specifications of APSR position limits and the usual regulating control rod and imbalance limits, which assure that the loss of coolant accident (LOCA) LHGR limits are not exceeded, are acceptable.

2.5 Thermal-Hydraulic Analyses

The thermal-hydraulic analyses for ANO-1 Cycle 3 were performed using previously approved methods and models per the ANO-1 Final Safety Analysis Report (FSAR). The only change in the thermal-hydraulic analysis for Cycle 3 is the removal of the densification power spike from Departure from Nucleate Boiling Ratio (DNBR) calculation, resulting in an increase in the minimum calculated steady-state DNBR from 1.84 for Cycle 2 to 1.90 for Cycle 3.

The maximum fuel rod bow, calculated using the interim NRC fuel rod bow model, is 11.2% and occurs at the end of Cycle 3. The licensee provides the requisite margin by the flux/flow trip setpoint of 1.060 and the variable low-pressure trip. We find that the thermal-hydraulic analysis for ANO-1 Cycle 3 operations is acceptable.

2.6 Accident and Transient Analysis

The generic Babcock and Wilcox (B&W) Loss of Coolant Accident (LOCA) analysis is contained in BAW-10103 (Reference 13). The analysis in BAW-10103 is generic since the limiting values of key parameters for all plants in the category I (177 FA-lowered loop) Nuclear Steam Supply System (NSSS) are used. The combination of average fuel temperature and pin pressure data, for the lifetime of the fuel, as used in the BAW-10103 LOCA limits analysis is conservative compared to those used in the Cycle 3 reload analysis. In Reference 14, B&W submitted a change to the BAW-10103 LOCA analysis because of an incorrect pressure drop assumed for the inlet nozzle region. The correction incorporates a revised reactor coolant system pressure distribution. The result is that the peak clad temperature in the revised calculation is 2060°F for the unruptured node and 1826°F for the ruptured node. This is a reduction of 86°F and 240°F, respectively, relative to the BAW-10103 results. Therefore, the analysis presented in BAW-10103 is valid for the reload cycle.

Relative to plant transients, the Cycle 3 evaluation is bounded by the FSAR, the fuel densification report (Reference 15) and previous cycle analyses.

We conclude that the LOCA analyses performed for ANO-1 meet 10 CFR 50.46 criteria and insure that the plant can be operated without undue risk to the public safety.

2.7 Physics Startup Tests

The proposed physics startup program is discussed in Reference 4. The licensee has committed to conduct physics startup tests to insure that the significant aspects of the ANO-1 Cycle 3 core would be within the acceptable criteria. These include control rod functional tests, scram times, control rod worth tests, temperature reactivity coefficient tests, and power distribution tests. The licensee has also committed to provide a report on these tests within 45 days after completion of the test program. The program has been reviewed and found acceptable.

3.0 Environmental Consideration

We have determined that the amendment does 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 the amendment involves 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 or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

4.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: March 17, 1978

References

1. Letter, Rueter (AP&L) to Davis (NRC), Docket No. 50-313, December 28, 1977, forwarding the Arkansas Nuclear One, Unit No. 1, Cycle 3 Reload Report, BAW-1471.

2. Letter, Cavanaugh (AP&L) to Davis (NRC), Docket No. 50-313, January 17, 1978.

3. Letter, Cavanaugh (AP&L) to Davis (NRC), Docket No. 50-313, January 30, 1978.

4. Letter, Williams (AP&L) to Reid (NRC), Docket No. 50-313, March 3, 1978.

5. BAW-10035, "Fuel Assembly Stress and Deflection Analysis for Loss-of-Coolant Accident and Seismic Excitation," June, 1970.

6. BAW-10133, "Mark-C Fuel Assembly Topical Report on LOCA - Seismic Analyses," October 1977.

7. BAW-1433, "Arkansas Nuclear One, Unit No. 1 - Cycle 2 Reload Report," November, 1976.

8. BAW-10084P-A, "Program to Determine In-Reactor Performance of B&W Fuels - Cladding Creep Collapse," January, 1975.

9. BAW-1004, "TAFY - Fuel Pin Temperature and Gas Pressure Analysis," May, 1972.

10. BAW-10055, Rev. 1, "Fuel Densification Report, June, 1973.

11. Standard Review Plan, Section 4.4, pp. 4.4-2 and 4.4-3.

12. Letter, Davis (NRC) to Phillips (AP&L), Safety Evaluation for Amendment No. 21 to Facility Operating License No. DPR-51, March 31, 1977.

13. BAW-10103, Rev. 1, "ECCS Analysis of B&W's 177-FA Lowered Loop NSS," September, 1975.

14. Letter, Taylor (B&W) to Baer (NRC), July 8, 1977.

15. BAW-1391, "Arkansas Nuclear One, Unit No. 1, Fuel Densification Report," June, 1973.