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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

Introduction

Arkansas Power & Light Company (AP&L) proposed to reload Arkansas Nuclear One - Unit 1 for Reload 1 (Cycle 2) and requested amendment to Facility Operating License No. DPR-51 by letter dated December 1, 1976 and supplements thereto dated January 13, February 7, 17, 22 and 24, and March 1, 9 and 17, 1977. By filing dated July 9, 1975, as supplemented by letters dated August 8 and 22, October 15, December 13, 1975, and the December 1, 1976 reload request, AP&L submitted their emergency core cooling system (ECCS) performance reevaluation as required by Appendix K of 10 CFR Part 50 of the Commission's regulations and the Commission's Order for License Modification dated December 27, 1974.

The amendment would modify the license and Technical Specifications to allow operation of the facility with:

(1) revised core protection limits in response to the plant specific analysis for reload 1;

(2) revised limits in response to modified fuel rod bow analyses;

(3) revised limits to reflect the modified Babcock and Wilcox (B&W) Company model for nucleate boiling heat transfer correlation during blowdown,

(4) new technical specification limiting conditions for operation and surveillance requirements governing core internal vent valves; and

(5) modified operating limits based upon an evaluation of emergency core cooling system (ECCS) performance calculated in accordance with an acceptable ECCS evaluation model that conforms with the requirements of Appendix K of 10 CFR Part 50 and as required by the Commission's Order for License Modification dated December 27, 1974, with the following exception. Our analysis of the electrical

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single failure criterion is still under consideration and will be the subject of a separate review. The incorporation of the modified operating limits relating to ECCS supersede the restrictions imposed by the Commission's Order dated December 27, 1974.

During our review of the proposed technical specifications, we determined that certain changes were necessary to conform with regulatory requirements. These changes have been accepted by AP&L. That portion of the January 13, 1977 letter related to examination of primary nozzle-to-vessel welds was authorized by Amendment No. 20 issued on March 15, 1977.

Discussion and Evaluation

Fuel Reload

The ANO-1 reactor core consists of 177 fuel assemblies, each with a 15x15 array of fuel rods. The reload in preparation for cycle 2 operation consists of the removal of all 56 batch 1 fuel assemblies, the relocation of some of the partially-spent batch 2 and batch 3 fuel assemblies, and the placement of the new batch of fuel assemblies in 8 positions in the interior of the core and the remaining 48 in the periphery of the core. Tables 4-2 and 4-3 of reference 1 summarize the reload core fuel assemblies parameters.

Fuel Mechanical Design

The outside dimensions and configuration of the new Mark B-4 (Batch 4) fuel assemblies and once-burned Mark B-3 fuel assemblies are identical except that the Mark B-4 have spring-type flexible spacers and the Mark B-3 have corrugated-type flexible spacers. This new fuel rod spacer has been previously reviewed and found acceptable by the NRC staff on the basis of no significant mechanical or material change to the reactor operation[9] and has been successfully operating in similar cores for a substantial time (Reference Section 4.5 and Table 4-1 of Reference 1). The new Mark B-4 fuel assemblies, therefore, do not represent any unreviewed or untested change in mechanical design from the reference cycle and are therefore acceptable.

This mechanical design change has been taken into account in the various analyses which are discussed in the following sections. The results of these analyses have shown that this fuel design difference in the ANO-1 core is of negligible effect and that the once burned fuel assemblies, batches 2 and 3, are limiting.

Fuel rod cladding creep collapse analyses were performed for the cycle 2 core. The CROV computer code was used to calculate the time to fuel rod cladding creep collapse.[1] The calculational methods, assumptions, and data have been previously reviewed and approved by the NRC staff.[10] The analysis assumed a 2000 hour densification time which maximizes creep; no fission gas production which maximizes differential pressure; and lower tolerance limit on clad thickness and upper tolerance limit on clad ovality, both of which maximize cladding creep deformation. Also, to be conservative, the most restrictive as-built fuel density was applied to the worst power region in the core. The actual operating history along with the most restrictive future power histories to which the partially-spent fuel assemblies may be exposed were used in the analyses of Batch 2 and Batch 3 fuels. The Batch 4 fuel analysis was not specifically performed because for cycle 2 operation Batch 3 fuel has been determined to have the most restrictive power level and will therefore be most limiting. An analysis of the Batch 4 fuel will be performed for cycle 3. Based on the analyses performed, no cladding creep collapse is predicted during the life of the fuel.

From the viewpoint of cladding stress due to differential pressure, thermal stress due to fuel temperature gradients, and bending stress, neither the yield stress nor the B&W 1% total strain criterion for the cladding is predicted to be exceeded in the cycle 2 core. The cladding stress estimated for cycle 1 core will envelope the limiting stresses for the cycle 2 core because of the lower prepressurization and lower fuel pellet density of the cycle 1 Batch 1 fuel. The B&W fuel design criterion for cladding circumferential plastic strain was shown to be satisfied for ANO-1 fuel.[3] This analysis used the maximum fuel pellet diameter burnup and density, and the minimum cladding inside diameter.

The Batch 4 fuel assembly design is based upon established concepts and utilizes standard component materials. Therefore, on the bases of the analyses presented and previously successful operations with equivalent fuel the staff concludes that the fuel mechanical design for cycle 2 operation is acceptable and its application to cycle 2 operation will not endanger the health and safety of the public.

Fuel Thermal Design

The fuel thermal design analysis was conducted with the TAFY-3 computer code, as discussed in reference 2. The analysis considered the effect of a power spike from fuel pellet densification, as modeled in the "Fuel Densification Report".[3] Modifications to the "Fuel Densification Report" on the fuel pellet void probability, F_q, and fuel grain size distribution, F_k, have been previously reviewed and approved by the NRC staff.[11]

As part of our evaluation of the TAFY code, the following modifications to the code were approved for use in reference 4:

1. The code option for no restructuring of fuel has been used;

2. The calculated gap conductance was reduced by 25%.

During cycle 2 operation, the highest power levels are predicted to occur in Batch 3 fuel. The fuel temperature analysis for this fuel, as documented in reference 5, is applicable for cycle 2 and is based on limiting Beginning-of-Cycle (BOC) conditions. Based on the analyses presented in reference 1 and comparison with allowable Linear Heat Generation Rate (LHGR) for fuel centerline melt considerations,(12) the fuel thermal design for the ANO-1 cycle 2 core is acceptable and can be applied with reasonable assurance that the health and safety of the public will not be endangered.

Fuel Material Design

The fuel material design for cycle 2 operation is not significantly different from that of cycle 1 operation. The only difference is that Zircaloy-4 is used as the fuel assembly tubular spacer material in Batch 4 fuel instead of zirconium dioxide (ZrO_2), which is used in Batch 2 and Batch 3 fuel. This change does not affect the primary coolant system chemistry. This change has been reviewed and has a substantial amount of previous experience (Section 4.5 and Table 4-1 of reference 1). Therefore, the fuel material design for ANO-1 cycle 2 operation is acceptable.

Nuclear Analysis

The reactor physics parameters for ANO-1 cycle 2 core were calculated with PDQ07. Since the ANO-1 core has not yet reached an equilibrium cycle, minor differences in the physics parameters between the initial cycle and cycle 2 cores are expected but are not significant. These insignificant differences include the technical specification basis change to α_T* due to cycle dependent parameters. In view of this and the fact that the startup tests which will be conducted prior to power operation will verify that the significant aspects of the core performance, e.g., control rod drive tests, scram times, shutdown margin, criticality checks, power symmetry, and instrument calibration are within specified acceptance criteria, the staff finds AP&L's nuclear analysis for cycle 2 to be acceptable.

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*The change represents a cycle-dependent correction to the moderator temperature coefficient in going from hot zero power to 95% of rated conditions and accounts for the difference in fuel temperature.

Thermal-Hydraulic Analysis

Major acceptance criteria for the thermal-hydraulic design are specified in the NRC's Standard Review Plan Section 4.4 ("Thermal and Hydraulic Design"). These criteria establish the acceptable limits for DNBR (Departure from Nucleate Boiling Ratio). The thermal-hydraulic analyses for the ANO-1 cycle 2 reload core were made with previously approved models and methods, as stated in the ANO-1 Final Safety Analysis Report (Docket No. 50-313).

The effect of fuel rod bow was evaluated with consideration given to both the hot channel power spike due to concave bowing away from the hot rod and the effect on DNBR of flow area reduction due to convex bowing toward the hot rod. These phenomena were evaluated separately since they are mutually exclusive. In the submittal dated January 13, 1977, AP&L summarized the methods and results of the rod bow analysis. This original rod bow analysis was performed with an as yet unapproved B&W model. Therefore, AP&L was requested to provide analyses with the NRC approved rod bow model. However, by letters of March 9, 1977 and March 17, 1977, AP&L was able to show sufficient available margin in the analyses in order to offset the difference between models without reducing any margins of safety.

The effect of rod bow on DNBR must be considered for both the variable pressure-temperature setpoint, quadrant tilt specifications, and the flux-flow trip. For the variable pressure-temperature setpoint and the quadrant tilt technical specifications, removal of the densification power spike and the flow area reduction penalties, as approved in reference 6, combine to provide adequate margin for the difference between the submitted and approved model without reducing any margins of safety. In the case of the flux-flow analysis, AP&L has proposed thermal margins from comparison of test to analytical assumptions for the reactor scram time, i.e., time from breaker trip to 3/4 rod insertion level. For this analysis, ANO-1 had previously used scram times which were related to Technical Specification values. However, testing resulted in scram times that were substantially lower. Thus, by decreasing the Technical Specification value by the time interval which corresponds to difference between the submitted and approved rod bow model, and without reducing any margins of safety AP&L has shown that the thermal analysis is equivalent to that with the approved rod bow model. All other Technical Specification setpoints were established with the NRC approved model and justified on that basis.

The reactor coolant flow rate was accurately measured during cycle 1 operation and a minimum measured value of 109.7% of the system design flow was determined. AP&L has proposed to take credit in the cycle 2 thermal-hydraulic analysis for the fact that the actual system flow is greater than the design flow rate, and has also included uncertainties and conservatisms in this analysis.

In the past, a 4.6% reactor coolant flow penalty had been assumed in the thermal-hydraulic design analysis for ANO-1. This penalty is associated with the potential of a core internal vent valve being stuck open during normal operation. The core internal vent valves are incorporated into the design of the reactor internals to preclude potential vapor lock during a postulated cold-leg break Loss-of-Coolant Accident (LOCA). The NRC staff has concluded that by application of a surveillance program the vent valve flow penalty may be removed. The surveillance requirements demonstrate that the vent valves are not stuck open and that the vent valves operate freely.

AP&L's proposed surveillance program has been reviewed. The program differs from previously approved surveillance programs in that: (1) it tests on a force equivalent basis for full open position, whereas the NRC recommended program suggests a start to open and a full open pressure differential across the vent valves; and (2) the proposed force equivalent corresponds to a larger pressure differential than recommended. By letter of February 22, 1977, AP&L has shown that the force equivalent method is applicable. By letter of March 9, 1977, AP&L has also shown that not testing for the start to open case and the greater force equivalent has a negligible effect on the limiting LOCA, i.e., less than 3°F increase in the peak cladding temperature (PCT) for the limiting LOCA analysis, and PCT remains less than 2200°F. Therefore, the NRC staff concluded that AP&L has proposed a surveillance program that adequately meets the NRC staff's concerns and requirements, and the core internal vent valve penalty was properly eliminated. The ANO-1 Technical Specifications are being modified to add the new surveillance specification.

There are differences in the flow resistance between the Mark B-3 fuel assemblies of Batches 2 & 3 and the Mark B-4 fuel assemblies of Batch 4. The flow resistance for a Mark B-4 fuel assembly is slightly less than that for the Mark B-3 assemblies. These differences have been analyzed and from this analysis it was concluded that the Mark B-3 lies are limiting for the ANO-1 cycle 2 operation. This phenomenon esults in cross flow which has been calculated and demonstrated from previous operating experience to be of negligible effect.[1]

In summary, AP&L has proposed that: (1) a reactor coolant flow rate based on a conservative adjustment of the actual measured flow rather than the design flow be used; (2) the 4.6% core vent valve flow penalty be eliminated by establishment of an acceptable surveillance program; and (3) the DNBR fuel densification power spike removal, flow area reduction credit, and rod bow penalty be incorporated. Because of the analyses mentioned above, we have found the thermal-hydraulic analysis to be acceptable and the proposed Technical Specifications related to thermal hydraulic analysis also acceptable.

Accident and Transient Analysis

The accident and transient analysis provided by AP&L demonstrates that the ANO-1 FSAR analyses conservatively bound the predicted condition for cycle 2 operation except for the items discussed below.

A. Loss-of-Coolant Flow

The analysis of this transient resulted in a setpoint reduction for the flux-flow-power imbalance trip. The overall reduction in trip setpoint resulted from a combination of credits as established in reference 6 and a penalty for rod bow power spike as discussed in the thermal hydraulics section of this report. The applicable analysis has been reviewed and found acceptable by the staff and the ANO-1 Technical Specifications are being modified to reflect the reduction in trip setpoint.

B. Loss-of-Coolant Accident (LOCA) Analysis

The previously applied W-3 Critical Heat Flux (CHF) correlation was replaced with the BAW-2 correlation. Both of these have been previously approved[13] for use in the LOCA analysis. The following modifications form the basis and substance of this change: (1) An extension downward from 2000 psia to 1750 psia for the applicable pressure range based on a review of rod bundle CHF data taken in the range of interest; and (2) A reduction in DNBR from 1.32, which represents a 99% confidence level that 95% of the rods will not experience DNB, to 1.30, which represents a 95% confidence level that 95% of the hot rods will not experience DNB. This is consistent with the Standard Review Plan[12] and industry practice. A revision to B&W's ECCS evaluation model was proposed[7] and has been approved by the NRC staff.[8] This change is to use a nucleate boiling heat transfer correlation during blowdown after critical heat flux (CHF) is first predicted. By letter dated February 17, 1977, AP&L submitted the approved generic B&W analysis(7) using the revised ECCS model.

The staff has reviewed these modifications as identified above and has concluded that they are in compliance with Appendix K of 10 CFR 50 and are acceptable for use in the ANO-1 analysis. This LOCA analysis submitted for the ANO-1 reload analysis meets the criteria of 10 CFR 50.46 and is acceptable on that basis.

The ECCS analyses submitted by the licensee (letter of July 9, 1975, as supplemented by letters dated August 8, August 22, October 15, and December 31, 1975, and the AP&L reload report of December 1, 1976, with its associated supplements) and reviewed by the NRC fulfilled the requirements of the Commission's December 27, 1974 Order for Modification of License and Appendix K of 10 CFR 50. The remaining exception is the completion of the ongoing NRC review of the ECCS electrical single failure criteria in response to the NRC letter of May 7, 1976. Based on findings of the ANO-1 licensing safety evaluation report dated June 6, 1973, no single failure has yet been identified which would require further modification to the technical specifications. Completion of this ongoing review is scheduled for June 1, 1977, and will be documented subsequently. Therefore, operation in the proposed manner does not endanger the public health and safety and is in compliance with the Commission's regulations.

The change in control rod position limits specifically mentioned in the ECCS-related Notice of Proposed Issuance of Amendment to License dated July 22, 1975, has been incorporated as part of this ECCS re-evaluation.

Startup Physics Tests

The proposed startup physics test program for ANO-1 has been reviewed. The program was discussed with AP&L for clarification of the number of measurements for critical boron concentration and moderator temperature coefficient. At least two of each of these measurements will be performed and the results compared with predictions. The acceptance criterion for the control rod reactivity worth measurements is being changed to require additional measurements if the initial acceptance criterion is not met.

The proposed startup physics test program with these clarifications and additions has been reviewed and found acceptable.

Technical Specifications

The proposed Technical Specifications changes for ANO-1 cycle 2 operation include:

1. incorporation of revised core protection limits in response to analyses mentioned above.

2. incorporation of new technical specification limiting conditions for operation and surveillance requirements regarding core vent valves.

3. changes to Technical Specification Bases to reflect the modifications of 1 and 2 above, and

4. modified operating limits related to ECCS.

Some modifications to the proposed Technical Specifications were necessary to meet NRC staff requirements. The staff finds that the proposed Technical Specifications, as modified, are acceptable and consistent with the information submitted by the licensee.

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.

Conclusions

Based on our review of the items identified as (1) through (4) in the introduction to this evaluation, and the considerations discussed in this evaluation, we have concluded that (1) because the items do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in safety margin, they do not involve a significant hazards consideration. We also have concluded, based on the considerations discussed in this evaluation, that all of the activities discussed herein will be conducted in compliance with the Commission's regulations and the issuance of an amendment to the license will not be inimical to the common defense and security or to the health and safety of the public, and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

Date:

References

1. "Arkansas Nuclear One - Unit 1, Cycle 2 - Reload Report," BAW-1433 November, 1976.

2. Morgan, C. A., and Kao, H. S., "TAFY-Fuel Pin Temperature and Glass Pressure Analysis," BAW-10044, May, 1972.

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

4. "Technical Report on Densification on Babcock & Wilcox Reactor Fuel," July 6, 1973.

5. "Arkansas Nuclear One, Unit 1 - Fuel Densification Report," BAW-1391, June, 1973.

6. Letter from D. L. Ziemann (NRC) to J. D. Phillips (AP&LCo) re: ANO-1 Rod Bow Penalty and Associated Credits, dated December 30, 1976.

7. Letter from K. E. Shurke (B&W) to S. A. Varga (NRC), Re: D. F. Ross December 2, 1976 letter to Shurke on B&W ECCS Evaluation Model, dated January 24, 1977.

8. Letter to D. B. Vassallo from D. F. Ross, Re: Topical Report Evaluation BAW-10104, ECCS Evaluation Model, Revised Nucleate Boiling Lockout Model, dated February 2, 1977.

9. SER on Oconee Nuclear Station, Units 1,2,&3, dated June 30, 1976, Amendment Nos. 27, 27 and 23 for License Nos. DPR-38, DPR-47 and DPR-55.

10. Letter from A. Schwencer (NRC) to J. F. Malloy (B&W), on January 29, 1975.

11. Memorandum from R. Lobel to D. F. Ross, "Present Status of B&W Power Spike Model," July 23, 1974.

12. Standard Review Plan, Section 4.4, pg. 4.4-2 and 4.4-3.

13. Letter dated April 15, 1976 from J. Stolz (NRC) to K. E. Shurke, on BAW-10000 A Topical Reporting of May, 1976.

14. Safety Evaluation Report By The Directorate of Licensing U. S. Atomic Energy Commission In The Matter of Arkansas Power & Light Company, Arkansas Nuclear One - Unit No. 1 Nuclear Power Plant, Pope County, Arkansas, Docket No. 50-313, June 6, 1973.