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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 92 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated September 26, 1984 (Ref. 1), supplemented by letter dated October 31, 1984 (Ref. 2), Arkansas Power and Light Company (AP&L or the licensee) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed changes would modify the TSs to permit operation for the seventh cycle. The safety analyses performed and the resulting modifications to the plant TSs are described in the Cycle 7 Reload Report (Ref. 3). Additional supporting information was provided by letter dated December 6, 1984 (Ref. 4).

The safety analysis for the previous sixth cycle of operation at ANO-1 is being used by the licensee as a reference for the proposed seventh cycle of operation. Where conditions are identical or limiting in the sixth cycle analysis, our previous evaluation (Ref. 5) of that cycle continues to apply.

1.1 Description of the Cycle 7 Core

The ANO-1 core consists of 177 fuel assemblies, each of which is a 15x15 array containing 208 fuel rods, 16 control rod guide tubes, and one incore instrument guide tube. Cycle 7 will operate in a feed-and-bleed mode with core reactivity control supplied mainly by soluble boron in the reactor coolant and supplemented by 61 full length control rod assemblies composed of silver-indium-cadmium alloy clad in stainless steel. In addition to the full length control rods, eight axial power shaping rods (APSRs) are provided for additional control of the axial power distribution. The licensed core full power level is 2568 MWt.

2.0 EVALUATION OF THE FUEL SYSTEM DESIGN

2.1 Fuel Assembly Mechanical Design

The 72 Babcock and Wilcox (B&W) Mark B-4 15x15 fuel assemblies to be loaded as Batch 9 fuel for Cycle 7 operation are mechanically interchangeable with Batch 8 fuel assemblies previously loaded at ANO-1. The cladding stress, strain and collapse analyses are bounded by conditions previously analyzed for ANO-1 or were analyzed specifically for Cycle 7 using methods and limits previously reviewed and approved by the NRC.

2.2 Fuel Rod Design

All batches in the ANO-1 Cycle 7 core utilize the same B&W Mark B-4 fuel design, and the Batch 9 fuel parameters are virtually identical to the previously loaded Batch 8 fuel except for enrichment, which has been increased from 3.21 to 3.30 wt/% U.

There has been a change in the pellet design for Batch 9 fuel rods. The fuel pellet length/diameter ratio has been decreased from 1.63 to 1.18. The licensee claims this change will not affect fuel performance, and at high burnups it is expected to decrease local cladding strains.

Four fuel assemblies in the highest burnup Batch 7B are extended burnup lead test assemblies (LTAs), which are scheduled for a third cycle of irradiation in Cycle 7. These assemblies, which are described in Reference 7, are similar in design to the standard Mark B-4 fuel assemblies except for changes to the fuel rod and fuel assembly structure to extend their burnup capability. We previously concluded (Ref. 6) that the irradiation of the four LTAs in ANO-1 was acceptable.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (Ref. 8) addresses a number of acceptance criteria used to establish the design bases and evaluation of the fuel system. Among those which may affect the operation of the fuel rod is the internal pressure limit Our current criterion (SRP 4.2, Section II.A.1(f)) is that fuel rod internal gas pressure should remain below normal system pressure during normal operation unless otherwise justified.

AP&L has stated that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 7. This analysis is based on the use of the approved B&W TACO2 code (Ref. 9). We conclude that the rod internal pressure limits have been adequately considered for Cycle 7 operation.

2.3 Fuel Thermal Design

There are no major changes between the thermal design of the new Batch 9 fuel and previous batches reinserted in the Cycle 7 core. The licensee presented results of the thermal design evaluation of the Cycle 7 core. These are based upon analyses performed with the TACO2 code. The Cycle 7 core protection limits were based on a linear heat rate to centerline fuel melt of 20.5 kW/ft. The results of the thermal design evaluation show no difference between Batch 9 fuel and the Batch 7 and 8 fuel already approved for use in the core. We have reviewed the fuel design parameters for normal operation and find them acceptable.

2.3.1 Loss of Coolant Accident (LOCA) Initial Conditions

In addition to the steady state conditions, the average fuel temperature as a function of linear heat rate and lifetime pin pressure data used in the LOCA analysis (Section 7.2 of the reload submittal) are also calculated with the TACO2 code (Ref. 9). The reload report stated that the fuel temperature and pin pressure data used in the generic LOCA analysis (Ref. 10) are conservative compared to those calculated for Cycle 7 at ANO-1.

2.4 Conclusion

We have reviewed the fuel system design and analysis for ANO-1 Cycle 7 operation and find the application acceptable.

3.0 EVALUATION OF THE NUCLEAR DESIGN

To support Cycle 7 operation of ANO-1, the licensee has provided analyses using analytical techniques and design bases established in reports that have been approved by the NRC staff. The licensee has provided a comparison of the core physics parameters for Cycles 6 and 7 as calculated with these techniques. There are slight differences in these parameters. This is to be expected since the core has not yet reached an equilibrium cycle. All of the accidents analyzed in the Final Safety Analysis Report (FSAR) were reviewed for Cycle 7 operation. We note that the Cycle 7 characteristics were conservative compared to those analyzed for previous cycles and no new accident analyses were performed.

We find the predicted characteristics acceptable because they use approved techniques, the validity of which has been reinforced through a number of cycles of predictions for this and other reactors. As a result of our review of the characteristics compared to previous cycles, we agree with the licensee's conclusions regarding Cycle 7 accident analysis.

The licensee's calculations took into account ejected rod worths and their adherence to accident analysis criteria in development of rod position limits for Cycle 7 Technical Specifications. The licensee has provided predictions of rod worths and control requirements demonstrating adequate shutdown margin throughout the cycle. Startup tests of control rod worth will provide a verification of the accuracy of these predictions.

Withdrawal of the APSRs is planned near the end of Cycle 7, as in Cycle 6. This results in a calculated stability index of -0.052 per hour, which ensures the axial stability of the core.

Core design changes for Cycle 7 are the transition to a very low leakage design and the use of "short-stack" lumped burnable poison rods. For this transition cycle, 12 twice-burned assemblies are located on the core periphery to reduce fluence levels on the reactor vessel. The analytic techniques used by B&W to predict physics parameters adequately account for the effects of such changes in the process of performing a reload analysis.

The lumped burnable poison used in Cycle 7 has a 9 inch shorter stack than that used in the standard Mark B design, i.e., 117 versus 126 inches of Al_2O_3- B_4C. The top 9 inches of the poison stack are replaced by a Zircaloy tubular spacer. This design produces a lower axial peak at the beginning of the cycle and increases operational flexibility. We reviewed the effects of this design and its impact on the calculation of the Technical Specification changes at a meeting with the licensee and B&W on November 27, 1984. The calculations conservatively account for such changes and, therefore, the short stack burnup poison design is acceptable.

4.0 EVALUATION OF THE THERMAL-HYDRAULIC DESIGN

The objective of the thermal-hydraulic review is to confirm that the design of the reload core has been accomplished using acceptable methods and that acceptable safety margin is available from conditions which would lead to fuel damage during normal operation and anticipated transients.

The thermal-hydraulic analysis for Cycle 7 was performed with a 1.71 design radial - local ($F\Delta_H^N$) power peak with a 1.65 symmetric chopped cosine design axial flux shape. This is in comparison with the 1.71 radial - local and 1.5 axial flux shape used in Cycle 6. The changed shape results in an allowable increase in the total peak for Cycle 7 to 2.83 from the Cycle 6 value of 2.57. The selection of the Cycle 7 peaking was made to increase flexibility in the determination of operating limits (i.e., rod insertion limits), and is appropriately accounted for in the safety analysis.

The thermal-hydraulic models and methodology used for Cycle 7 are the same as used for Cycle 6, except for the implementation of crossflow modeling with the LYNX1, LYNX2, and LYNXT computer codes (References 11-13, respectively). The crossflow modeling is described in Reference 14 submitted as part of the Cycle 7 reload package. LYNX1 and LYNX2 are approved codes. Our review of LYNXT is not yet complete, but it has progressed sufficiently to allow its use for this application. We reviewed the crossflow modeling described in Reference 13, and find it acceptable for Cycle 7.

Departure from Nucleate Boiling (DNB) margin improvement gained with crossflow modeling would support an increase of the flux/flow reactor trip setpoint up to 1.08 for Cycle 7. The licensee, however, has elected to use a value of 1.07 for this setpoint. This, and the other Technical Specification changes for Cycle 7 have been conservatively selected to permit the potential application of these limits to future cycles without the need for additional Technical Specification changes. Since the changes have been chosen conservatively, this approach is acceptable.

The important thermal-hydraulic parameters are the same for both Cycles 6 and 7 as summarized in Table 1. Based on the similarities of Cycles 6 and 7, we find the operation of Cycle 7 acceptable.

Table 1. Maximum Design Conditions, Cycles 6 and 7

	Cycle 6	Cycle 7
Design power level, MWt	2568	2568
System pressure, psia	2200	2200
Reactor coolant flow, % design	106.5	106.5
Vessel inlet/outlet coolant temp at 100% power, F	555.6/602.4	555.6/602.4
DNBR modeling	Closed-channel	Crossflow
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.5 cosine	1.65 cosine
Hot channel factors		
Enthalpy rise	1.011	1.011
Heat flux	1.014	1.014
Flow area	0.98	0.98
Active fuel length, in.	140.7[a]	141.8
Avg. heat flux at 100% power, 10^3 Btu/h-ft^2	175[a]	174
Max heat flux at 100% power, 10^3 Btu/h-ft^2	450[a]	492
CHF correlation	B&W-2	B&W-2
Minimum DNBR		
At 112% power	2.05	2.08
At 100% power	2.39	2.43

[a] Based on densified length.

5.0 ACCIDENT AND TRANSIENT ANALYSIS

The licensee has examined each FSAR accident analysis with respect to changes in Cycle 7 parameters to determine their effect on the plant thermal performance during the analyzed accidents and transients. The key parameters having the greatest effect on the outcome of a transient or accident are the core thermal parameters, thermal-hydraulic parameters, and physics and kinetics parameters. Fuel thermal analysis values are listed in Table 4-2 of Reference 3 for all fuel batches in Cycle 7. Table 1 compares the thermal-hydraulic parameters for Cycles 6 and 7. These parameters are the same for both cycles. A comparison of the key kinetic parameters from the FSAR and Cycle 7 is provided in Table 7-2 of Reference 3. These comparisons indicate no significant changes or changes in the conservative direction. The effects of fuel densifictaion on the FSAR accident analyses have also been evaluated.

A generic LOCA analysis for the B&W 177-fuel assembly, lowered loop nuclear steam supply system (NSSS) has been performed using the final acceptance criteria emergency core cooling system (ECCS) evaluation model (Reference 10). That analysis used the limiting values of key parameters for all plants in the 177-FA lowered-loop category, and therefore is bounding for the ANO-1 Cycle 7 operation.

We conclude from the examination of Cycle 7 core thermal and kinetic properties, with respect to acceptable previous cycle values and with respect to the FSAR values, that this core reload will not adversely affect the ANO-1 plant's ability to operate safely during Cycle 7.

6.0 TECHNICAL SPECIFICATIONS

As indicated in our review of Section 3, the operating characteristics for Cycle 7 were calculated with well-established, approved methods. In addition, we agreed with the licensee's evaluation of control rod worths and their role in the establishment of Cycle 7 control rod position limits. The proposed Technical Specifications are the result of the cycle-specific analyses for power peaking, control rod worths, and quadrant tilt allowance. We discussed the Specification of the flux/flow reactor trip setpoint in Section 4.

With the above modification, we therefore conclude that the Technical Specification changes proposed by the licensee in Reference 1 and repeated in Section 8 of the Cycle 7 Reload Report (Ref. 3) are acceptable.

At our request, in Reference 2 the licensee withdrew credit for use of the FLECSET heat transfer correlation in the LOCA analysis contained in the original submittal. This affected only the linear heat rate limits. Figure 3.5.2-4, "LOCA Limited Maximum Allowable Linear Heat Rate," as revised in Reference 2, is the correct figure to use. The licensee proposed to delete this figure, but also provided the corrected figure if we did not agree to the deletion. Since this figure defines the "maximum peaking factor allowed by the Technical Specifications" mentioned in Part 50, Appendix K, we do not approve of its deletion.

7.0 STARTUP TESTING

We reviewed the startup testing program for ANO-1 presented in Reference 3. We find that this program will acceptably verify the cycle design and provide data required by the Technical Specifications.

8.0 EVALUATION FINDINGS

We have reviewed the fuels, physics, thermal-hydraulic and transient information presented in the ANO-1 reload report. We find the proposed reload and the associated modified Technical Specifications acceptable.

9.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. We have determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

10.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) 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: December 20, 1984

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REFERENCES

1. John M. Griffin (AP&L) letter tc J. F. Stolz (NRC) on "Cycle 7 Reload Report and Proposed Technical Specification Change Request" dated September 26, 1984.

2. John M. Griffin (AP&L) letter to J. F. Stolz (NRC) on "Revision to ANO-1 Cycle 7 Reload Report" dated October 31, 1984.

3. "Arkansas Nuclear One, Unit 1, Cycle 7 Reload Report," Babcock and Wilcox Company Report BAW-1840 dated August 1984 (transmitted with Reference 1 above).

4. J. Ted Enos (AP&L) letter to John F. Stolz (NRC) on ANO-1 Technical Specification Revision - Cycle 7 Reload, dated December 6, 1984.

5. John F. Stolz (NRC) letter to John M. Griffin (AP&L) on Amendment No. 71, Cycle 6 Reload, dated March 10, 1983.

6. Robert W. Reid (NRC) letter to William Cavanaugh III (AP&L) on Amendment No. 52, Cycle 5, dated March 9, 1981.

7. D. C. Trimble (AP&L) letter to R. W. Reid (NRC) dated November 6, 1980 and transmitting BAW-1626.

8. Standard Review Plan, Section 4.2 (Rev. 1), "Fuel System Design," U. S. Nuclear Regulatory Commission Report NUREG-0800 (formerly NUREG-75/087, July 1981.

9. Y. Hsii, et al., "TACO2-Fuel Pin Performance Analysis," Babcock and Wilcox Company Report BAW-10141P-A, Rev. 1, June 1983.

10. W. L. Bloomfield, et. al., "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS," BAW-10103, Revision 1, September 1975.

11. B. R. Hao and J. M. Alcorn, "LYNX1: Reactor Fuel Assembly Thermal-Hydraulic Analysis Code," BAW-10129, October 1976.

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14. R. L. Harne and J. H. Jones, "Thermal-Hydraulic Crossflow Applications," BAW-1829, April 1984.