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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 71 TO

FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 Introduction

By letter dated November 19, 1982(1), as supplemented November 23, 1982(2), January 27, 1983(3), and February 11, 1983(4), Arkansas Power and Light Company (AP&L or the licensee) requested amendment to the Appendix A Technical Specifications (TSs) of the Arkansas Nuclear One, Unit No. 1 (ANO-1) License No. DPR-51 which would permit power operation during Cycle 6.

The safety analysis for the previous fifth cycle of operation at ANO-1 is being used by the licensee as a reference for the proposed sixth cycle of operation. Where conditions are identical or limiting in the fifth cycle analysis, our previous evaluations (5-6) of that cycle continue to apply.

1.1 Description of the Cycle 6 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 6 will operate in a feed-and-bleed mode with the 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. 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 loaded as Batch 8 at the end of Cycle 5 (EOC 5) are mechanically interchangeable with Batches 6 and 7 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 6 using methods and limits previously reviewed and approved by the NRC.

2.2 Fuel Rod Design

All batches in the ANO-1 Cycle 6 core utilize the same B&W Mark B-4 fuel design, and the Batch 8 fuel parameters are virtually identical to the previously loaded Batch 7 fuel except for enrichment, which has been increased from 2.95 to 3.21% U^{235}.

We also note that eight of the fuel rods in Batch 8 assemblies contain fuel pellets manufactured by another vendor (General Electric) who has not previously provided pellets for B&W fuel. There is no reason to suspect that this fuel would behave much differently from the standard B&W fuel, and the amount of fuel is very small so there is no concern related to the operation of Cycle 6. However, if widescale use of such fuel is made in the future, information would be needed at that time to show that (a) important characteristics of that fuel (e.g., surface roughness, initial fuel density, resintered density, etc.) are adequately treated in the safety analysis, (b) some previous test program experience had been accumulated on the fuel and (c) some surveillance plan had been initiated to detect major anomalies. It is clear that the inclusion of these eight fuel rods in the ANO-1 Cycle 6 core is intended to satisfy item (b) of this list. We, therefore, find the proposed addition of the eight fuel rods acceptable.

The previous Batch 7 fuel includes four extended burnup lead test assemblies (LTAs), which are scheduled for a second cycle of irradiation in Cycle 6. 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(5) that the irradiation of the four LTAs in ANO-1 was acceptable.

For Cycle 6, the licensee has informed us(1) that the inspection program for the LTAs will be reduced. We believe that a substantial level of the fuel surveillance is necessary to support the irradiation of a lead prototype irradiation. The reason for this position is that surveillance of the LTAs would be required to support full-core reloads using the new design, whereas the same surveillance would not (generally) be required to assure the safety of Cycle 6. However, we agree that the LTAs are not limiting for Cycle 6 operation and will readdress the surveillance issue when a full-core reload submittal is made for the new fuel design.

2.2.1 Rod Internal Pressure

Section 4.2 of the Standard Review Plan (SRP)(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.

The licensee has stated that fuel rod internal pressure will not exceed nominal system pressure during normal operation for Cycle 6. This analysis is based on the use of the B&W TAFY code(9) rather than a newer B&W code called TACO(10). Although both of these codes are currently approved for use in safety analyses, we believe that only the newer TACO code is capable of correctly calculating fission gas release (and, therefore, rod pressure) at very high burnups. B&W has responded(11) to this concern with an analytical comparison between both codes. In this response, they have stated that the internal fuel rod pressure predicted by TACO is lower than that predicted by TAFY for fuel rod exposures of up to 42,000 MWd/tU. Although we have not examined the comparison, we note that the analyses exceed the expected exposure in ANO-1 at end-of-life Cycle 6 for all assemblies. We conclude that the rod internal pressure limits have been adequately considered for Cycle 6 operation.

2.3 Fuel Thermal Design

There are no major changes between the thermal design of the new Batch 8 fuel and previous batches reinserted in the Cycle 6 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 TAFY code (9). B&W has stated that the fuel temperature and pin pressure data used in the generic LOCA analysis(12) are conservative compared to those calculated for Cycle 6 at ANO-1.

As previously mentioned in Section 2.2.1 of this evaluation, B&W currently has two fuel performance codes, TAFY(9) and TACO(10), which could be used to calculate the LOCA initial conditions. The older code, TAFY, has been used for the Cycle 6 LOCA analysis. Recent information(13) indicates that the TAFY code predictions do not produce higher peak cladding temperatures than TACO for all Cycle 6 conditions as suggested in Reference 11. The issue involves calculated fuel rod internal gas pressures that are too low at beginning of life. The rod internal pressures are used to determine swelling and rupture behavior during LOCA. B&W has proposed(14) a method of resolving this issue which was accepted by the NRC staff(15). The method involves the use of reduced LOCA kW/ft limits at low core elevations during the first 50 effective full power days (EFPDs) of operation. The licensee has incorporated these changes into the ANO-1 TSs to support the operation of the plant at full rated power during Cycle 6. We have reviewed these changes and find them acceptable. We conclude that the initial thermal conditions for LOCA analysis have been appropriately considered for Cycle 6 operation.

2.4 Conclusion

We have reviewed the fuel system design and analysis for ANO-1 Cycle 6 operation and find the application acceptable.

3. Evaluation of the Nuclear Design

To support Cycle 6 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 5 and 6 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 6 operation. We note that the Cycle 6 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 6 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 6 TSs. The maximum stuck rod worth for Cycle 6 is less than that for the design Cycle 5 at BOC and EOC. 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.

There is one significant operational change for Cycle 6 from the reference cycle (Cycle 5). This is withdrawal of the APSRs during the last 37 EFPDs of the cycle. This results in a calculated stability index of - 0.031 per hour, which ensures the axial stability of the core.

4. 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 models and methodology used for Cycle 6 are the same as used for Cycle 5. The rod bow Departure from Nucleate Boiling Ratio (DNBR) penalty was calculated using the interim rod bow penalty evaluation procedure approved in Reference 16. The burnup used to calculate the penalty was the highest assembly burnup in Cycle 6 of 16,883 MWd/mtU.

The important thermal-hydraulic parameters are the same for both Cycles 5 and 6 as summarized in Table 1. Based on the similarities of Cycles 5 and 6, we find the operation of Cycle 6 acceptable.

5. Accident and Transient Analysis

The licensee has examined each FSAR accident analysis with respect to changes in Cycle 6 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 6. Table 1 compares the thermal-hydraulic parameters for Cycles 6 and 5. These parameters are the same for both cycles. A comparison of the key kinetic parameters from the FSAR and Cycle 5 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 densification 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 (12). That analysis used the limiting values of key parameters for all plants in the 177-fuel assembly lowered-loop category, and therefore is bounding for the ANO-1 Cycle 6 operation.

We conclude from the examination of Cycle 6 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 6.

6. Technical Specifications

As indicated in our review of Section 3 of the Cycle 6 Reload Report (1 & 2), the operating characteristics for Cycle 6 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 6 control rod position limits.

We therefore conclude the TS changes proposed by the licensee in Section 8 of the Cycle 6 Reload Report(1 & 2) including the modified changes described in References 3 and 4, are acceptable.

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.

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 an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, 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: MAR 10 1983

The following NRC personnel have contributed to this Safety Evaluation: John Voglewede, Marvin Dunenfeld, Amira Gill, Guy Vissing.

Table 1. Maximum Design Conditions, Cycles 5 and 6

	Cycle 5	Cycle 6
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
Reference design radial-local power peaking factor	1.71	1.71
Reference design axial flux shape	1.5 cosine	1.5 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.	140.2	140.2
Avg heat flux at 100% power, 10 Btu/h-ft (a)	175	175
Max heat flux at 100% power, 10 Btu/h-ft(b)	449	449
CHF correlation	BAW-2	BAW-2
Minimum DNBR		
At 112% power	2.05	2.05
At 108% power	2.18	2.18
At 100% power	2.39	2.39

(a) Heat flux was based on densified length (in the hottest core location).

(b) Based on average heat flux with reference peaking.

REFERENCES

1. William Cavanaugh III (AP&L) letter to J. F. Stolz (NRC) on "Cycle 6 Reload Report Submittal" dated November 19, 1982.

2. J. R. Marshall (AP&L) letter to J. F. Stolz (NRC) on "Cycle 6 Reload Report Submittal" dated November 23, 1982.

3. J. R. Marshall (AP&L) letter to J. F. Stolz (NRC) on "Cycle 6 Reload Submittal" dated January 27, 1983.

4. J. R. Marshall (AP&L) letter to J. F. Stolz (NRC) on "Cycle 6 Reload Submittal" dated February 11, 1983.

5. Robert W. Reid (NRC) letter to William Cavanaugh III (AP&L) on Amendment No. 52 to Facility Operating License No. DPR-51 dated March 9, 1981.

6. Guy S. Vissing (NRC) letter to William Cavanaugh III (AP&L) on Amendment No. 68 to Facility Operating License No. DPR-51 dated September 8, 1982.

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. C. D. Morgan and H. S. Kao, "TAFY-Fuel Pin Temperature and Gas Pressure Analysis", Babcock and Wilcox Company Report BAW-10044, May 1972.

10. "TACO-Fuel Pin Performance Analysis", Babcock and Wilcox Company Report BAW-10087P-A, Rev. 2, August 1977.

11. J. H. Taylor (B&W) letter to P. S. Check (NRC), dated July 18, 1978.

12. W. L. Bloomfield, et. al., "ECCS Analysis of B&W's 177-FA Lowered-Loop NSS", Babcock and Wilcox Company Report BAW-10103, Revision 1, September 1975.

13. R. O. Meyer (NRC) memorandum to L. S. Rubenstein (NRC) on "TAFY/TACO Fuel Performance Models in B&W Safety Analyses", dated June 10, 1980.

14. J. H. Taylor (B&W) letter to L. S. Rubenstein (NRC) dated September 5, 1980.

15. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) dated October 28, 1980.

16. L. S. Rubenstein (NRC) letter to J. H. Taylor (B&W) on "Evaluation of Interim Procedure for Calculating DNBR Reduction Due to Rod Bow" dated October 18, 1979.