ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
License No. DPR-51

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated August 17, 1976, as supplemented by letters dated December 20 and 22, 1976, and January 13, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.c(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

OFFICE ➤						
SURNAME ➤				8004180693		
DATE ➤						

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. , are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A portion of the Technical Specifications with the attached revised pages. The changed areas on the revised pages are identified by a marginal line.

REMOVE	ADD
17	17*
18	18
18a	18a
19	19
20	20
77	77
- -	77a
- -	77b

*There were no changes on this page. It is included as a matter of convenience in updating the Technical Specifications.

OFFICE						
SURNAME						
DATE						

Bases

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less.(1)

The decay heat removal system suction piping is designed for 300 F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat. (4) Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.(5) The pressurizer code safety valve lift set point shall be set at 2500 psig ± 1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7.

(2) FSAR, Section 4.2.5.1 and 9.5.2.3.

(3) FSAR, Section 4.2.5.4.

(4) FSAR, Section 4.3.10.4 and 4.2.4.

(5) FSAR, Section 4.3.7.

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests:

For thermal steady state system hydro tests the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core and to ASME Code Section III limits when no fuel assemblies are present provided:

a. Prior to initial criticality the reactor coolant system temperature is 100°F or greater, or

b. After initial criticality and prior to the accumulation of 1.7×10^6 thermal megawatt-days operation the reactor coolant system temperature is 215°F or greater.

3.1.2.2 Leak Tests

a. Leak tests may be conducted under the provisions of 3.1.2.1 above or

b. After initial criticality and prior to the accumulation of 1.7×10^6 thermal megawatt-days operation the system may be tested to a pressure of 1150 psig provided that the system temperature is 175°F or greater.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-1. The heatup rates shall not exceed those shown on Figure 3.1.2-1.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the left of and below the limit line in Figure 3.1.2-2. Cooldown rates shall not exceed those shown in Figure 3.1.2-2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100 F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.6 Prior to exceeding 1.7.x 10^6 thermal megawatt-days of operation, Figures 3.1.2-1 and 3.1.2-2 and Technical Specifications 3.1.2.1.b and 3.1.2.2 shall be updated for the next service period in accordance with 10 CFR 50 Appendix G, Section V.B. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with Specification 4.2.7. The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.7.

3.1.2.7 The updated proposed technical specifications referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR, Part 50, Appendix G, Section V.C.

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.(1) These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100 F per hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100 F satisfies stress levels for temperatures below the DTT.(3) The plate material and welds in the core region of the reactor vessel have been tested to verify conformity to specified requirements and a maximum NDTT value of 1(F has been determined based on Charpy V-notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40 F.

Figures 3.1.2-1 and 3.1.2-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT(4) and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4-10.(4) The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this or a similar reactor vessel.(5) The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 Mev) exposure of the reactor vessel is 3.0×10^{10} n/cm^{2}sec at 2568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation.(6) The calculated maximum values are 2.2×10^{10} n/cm^{2}sec and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load.(4) Figure 3.1.2-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1 is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days which is equivalent to 655 days at 2568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of this or a similar vessel in the core area. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitude of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1.2-1 and 3.1.2-2 are applicable to reactor core thermal ratings up to 2568 MWt.

The pressure limit line on Figure 3.1.2-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

A. A 25 psi error in measured pressure.

B. System pressure is measured in either loop.

C. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism, in lieu of portions of the operational requirements of Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50 F/hr (averaged over one hour) has been imposed below 275°F as shown on Figure 3.1.2-1.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) FSAR, Section 4.3.3

(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

IS-261 Item	Component	Exception
6.4	Bolting 2Ø	Not Applicable
6.6	Integrally Welded Valve Supports	Not Applicable

4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.

4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support, shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 Complete surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that within a 10 year period after start-up all four reactor coolant pump flywheels will be examined.

4.2.7 The reactor vessel material irradiation surveillance specimens removed from the reactor vessel in 1976 shall be installed, irradiated in and withdrawn from the Davis-Besse Unit No. 1 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2-1, Arkansas Power & Light Company shall be responsible for testing the specimens and submitting a report of test results in accordance with 10 CFR 50, Appendix H.

4.2.8 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of any of the following:

1. Failure of Davis-Besse Unit No. 1 to achieve commercial operation at 100% power by January 1, 1978, or

2. Beginning one year after attainment of commercial operation at 100% power, any time that Davis-Besse Unit No. 1 fails to maintain a cumulative reactor utilization factor of greater than 65%.

The repo[illegible] shall provide justification for continued operation of ANO-1 w[illegible] reactor vessel surveillance program conducted at Davis-B[illegible] No. 1 or the application for license amendment shall pr[illegible]ternative program for conduct of the ANO-1 reactor vess[illegible]rveillance program.

Table 4.2-1

ANO-1 CAPSULE ASSEMBLY WITHDRAWAL SCHEDULE AT DAVIS-BESSE 1

CAPSULE	INSERTION/WITHDRAWAL
ANI-E	Has been withdrawn for testing
ANI-B	Withdraw following 1st cycle at Davis-Besse 1
ANI-A	Withdraw following 3rd cycle at Davis-Besse 1
ANI-C	Withdraw following 7th Cycle at Davis-Besse 1
ANI-D	Insert in location WZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 12th cycle
ANI-F	Insert in location YZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 11th cycle

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code Inservice Inspection of Nuclear Reactor Coolant Systems, 1971, including 1972 Summer Addenda edition.

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

For the purpose of Technical Specification 4.2.8, the definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commercial operation". Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commerical operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)].