Amendment No. 28

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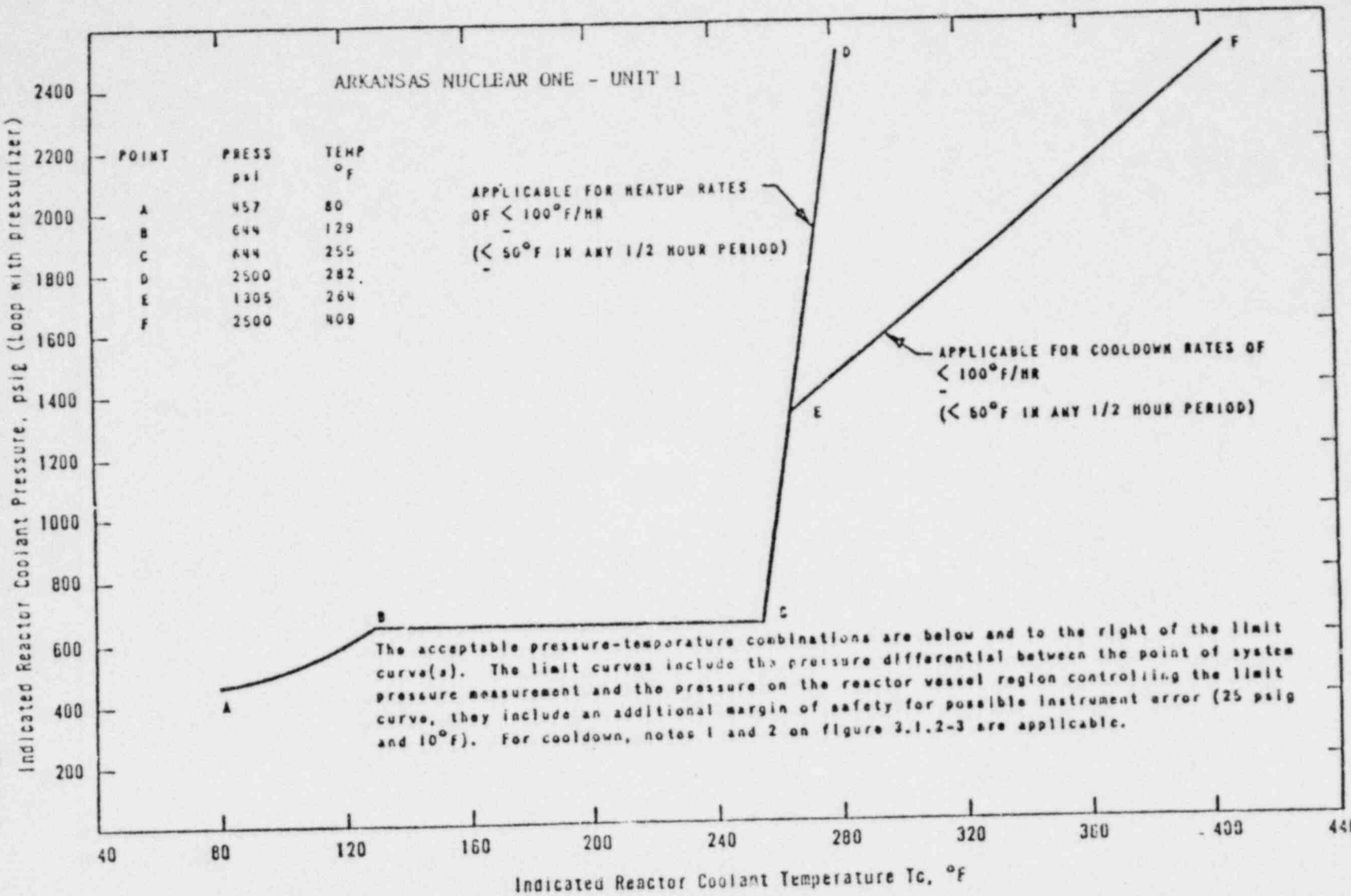
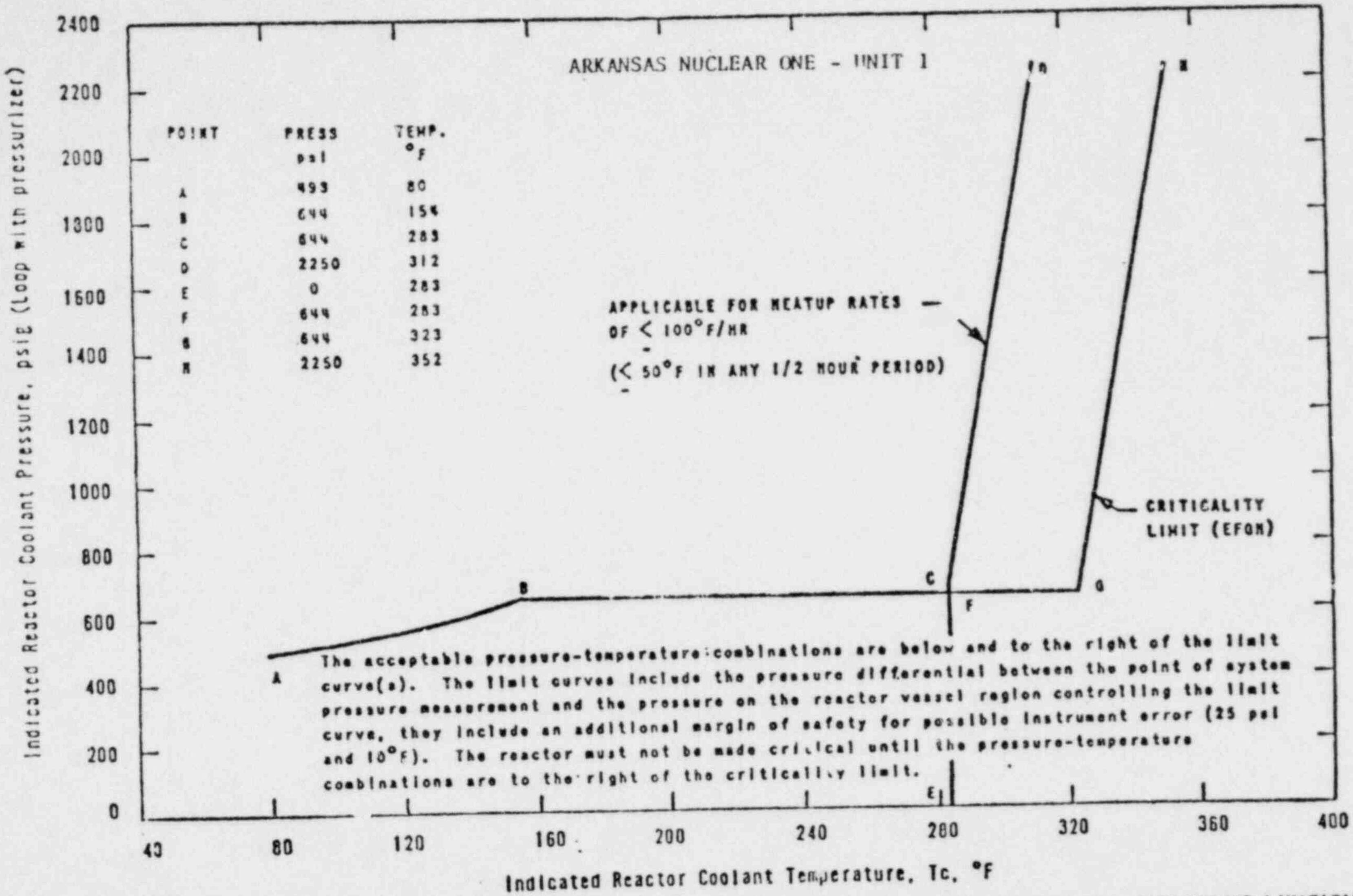


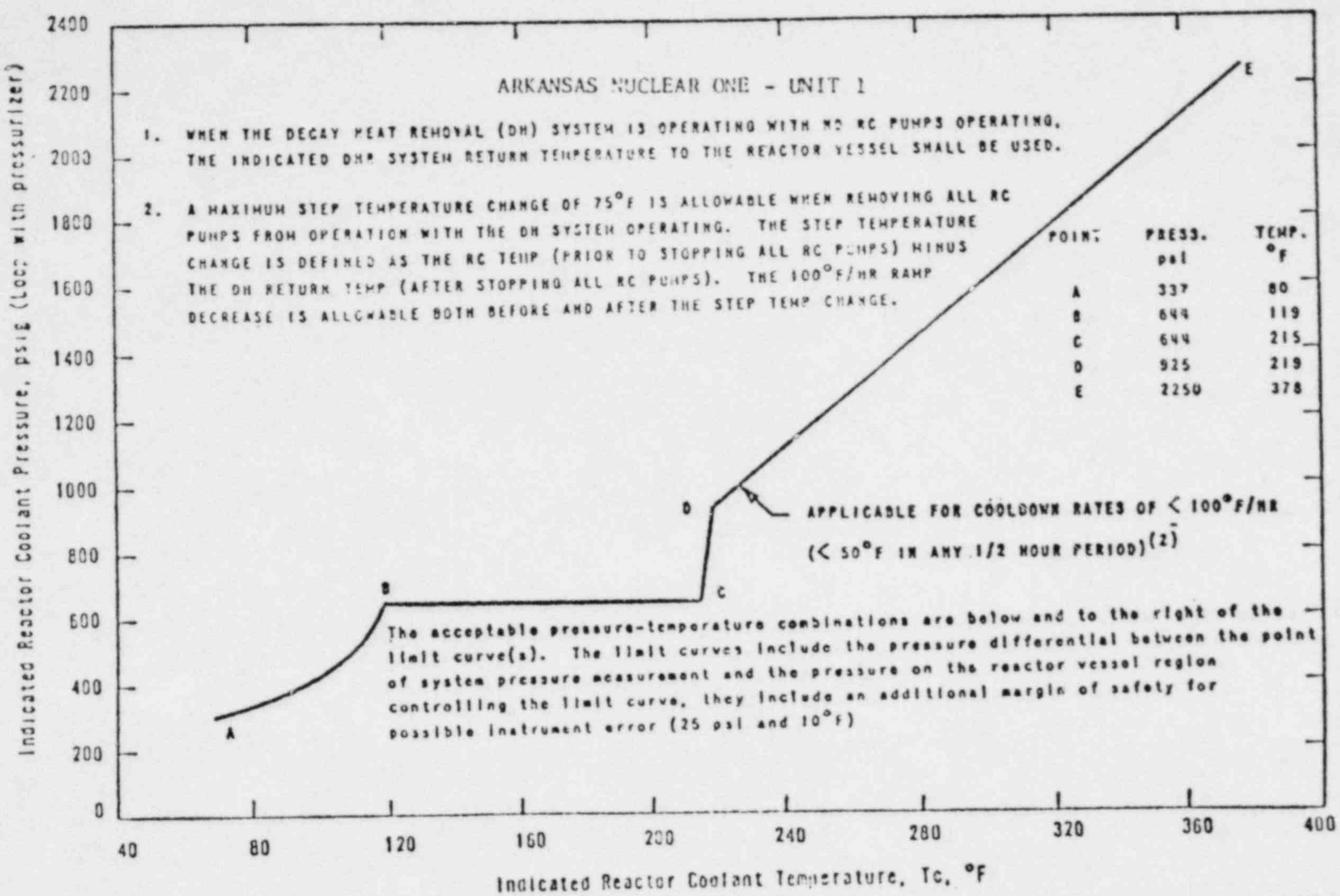
REACTOR COOLANT SYSTEM INSERVICE HYDROSTATIC TEST HEATUP AND COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 5.0 EFFECTIVE FULL POWER YEARS

Figure 3.1.2-1



REACTOR COOLANT SYSTEM, NORMAL OPERATION-HEATUP LIMITATIONS,
APPLICABLE FOR FIRST 5.0 EFFECTIVE FULL POWER YEARS

Figure 3.1.2-2



REACTOR COOLANT SYSTEM, NORMAL OPERATION-COOLDOWN LIMITATIONS APPLICABLE FOR FIRST 5.0 EFFECTIVE FULL POWER YEARS

Figure 3.1.2-3

3.1.3 Minimum Conditions For Criticality

Specification

3.1.3.1 The reactor coolant temperature shall be above 525F except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply.

3.1.3.2 Reactor coolant temperature shall be to the right of the criticality limit of Figure 3.1.2-2.

3.1.3.3 When the reactor coolant temperature is below the minimum temperature specified in 3.1.3.1 above, except for portions of low power physics testing when the requirements of Specification 3.1.8 shall apply, the reactor shall be subcritical by an amount equal to or greater than the calculated reactivity insertion due to depressurization.

3.1.3.4 The reactor shall be maintained subcritical by at least 1 percent $\Delta k/k$ until a steam bubble is formed and an indicated water level between 45 and 305 inches is established in the pressurizer.

3.1.3.5 Except for physics tests and as limited by 3.5.2.1, safety rod groups shall be fully withdrawn and the regulating rods shall be positioned within their position limits as defined by Specification 3.5.2.5 prior to any other reduction in shutdown margin by deboration or regulating rod withdrawal during the approach to criticality.

Bases

At the beginning of life of the initial fuel cycle, the moderator temperature coefficient is expected to be slightly positive at operating temperatures with the operating configuration of control rods.(1) Calculations show that above 525F the positive moderator coefficient is acceptable.

Since the moderator temperature coefficient at lower temperatures will be less negative or more positive than at operating temperature,(2) startup and operation of the reactor when reactor coolant temperature is less that 525F is prohibited except where necessary for low power physics tests.

The potential reactivity insertion due to the moderator pressure coefficient(2) that could result from depressurizing the coolant from 2100 psia to saturation pressure of 900 psia is approximately 0.1 percent $\Delta k/k$.

During physics tests, special operating precautions will be taken. In addition, the strong negative Doppler coefficient(1) and the small integrated $\Delta k/k$ would limit the magnitude of a power excursion resulting from a reduction of moderator density.

The requirement that the reactor is not to be made critical below the limits of Figure 3.1.2-2 provides increased assurances that the proper relationship between primary coolant pressure and temperatures will be maintained relative to the NDTT of the primary coolant system. Heatup to this temperature will be accomplished by operating the reactor coolant pumps.

3.1.8 Lower Power Physics Testing Restrictions

Specification

The following special limitations are placed on low power physics testing.

3.1.8.1 Reactor Protective System Requirements

A. Below 1720 psig, shutdown bypass trip setting limits shall apply in accordance with Table 2.3-1.

B. Above 1800 psig, nuclear overpower trip shall be set at less than 5.0 percent. Other settings shall be in accordance with Table 2.3-1.

3.1.8.2 Startup rate rod withdrawal hold(1) shall be in effect at all times.

3.1.8.3 During low power physics testing the minimum reactor coolant temperature for criticality shall be to the right of the criticality limit of Figure 3.1.2-2. A minimum shutdown margin of 1% $\Delta k/k$ shall be maintained with the highest worth control rod fully withdrawn.

Bases

The above specification provides additional safety margins during low power physics testing.

REFERENCES

(1) FSAR, Section 7.2.2.1.3.

3.1.9 Control Rod Operation

Specification

3.1.9.1 The concentration of dissolved gases in the reactor coolant shall be limited to 100 std. cc/liter of water at the reactor vessel outlet temperature.

3.1.9.2 Allowable combinations of pressure and temperature for control rod operation shall be to the left of and above the limiting pressure versus temperature curve for a dissolved gas concentration of 100 std. cc/liter of water as shown in Figure 3.1.9-1.

3.1.9.3 In the event the limits of Specifications 3.1.9.1 or 3.1.9.2 are exceeded, the center control rod drive mechanism shall be checked for accumulation of undissolved gases.

Bases

By maintaining the reactor coolant temperature and pressure as specified above, any dissolved gases in the reactor coolant system are maintained in solution.

Although the dissolved gas concentration is expected to be approximately 20-40 std. cc/liter of water, the dissolved gas concentration is conservatively assumed to be 100 std. cc/liter of water at the reactor vessel outlet temperature.

The limiting pressure versus temperature curve for dissolved gases is determined by the equilibrium pressure versus temperature curve for the dissolved gas concentration of 100 std. cc/liter of water. The equilibrium total pressure is the sum of the partial pressure of the dissolved gases plus the partial pressure of water at a given temperature. The margin of error consists of the maximum pressure difference between the pressure sensing tap and lowest pressure point in the system, the maximum pressure gage error, and the pressure difference due to the maximum temperature gage error.

If either the maximum dissolved gas concentration (100 std. cc/liter of water) is exceeded or the operating pressure falls below the limiting pressure versus temperature curve, the center CRDM should be checked for accumulation of undissolved gases.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 28 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

Introduction

By letter dated July 8, 1977, (with supportive document BAW-1440), Arkansas Power & Light Company (AP&L) requested changes to the Technical Specification pressure-temperature limit curves for hydrostatic test, normal heatup, and normal cooldown for Arkansas Nuclear One - Unit 1 (ANO-1). This change was required by Technical Specification 3.1.2.7 and was accompanied by the aforementioned Babcock and Wilcox Company document BAW-1440, which detailed the analysis performed by Babcock and Wilcox on ANO-1 surveillance capsule ANI-E. The surveillance capsule had been withdrawn in the Spring of 1976. During our review we determined that certain changes to the proposed specifications were necessary to meet our requirements. These changes were discussed with and agreed to by the AP&L staff and have been incorporated.

Discussion and Evaluation

As noted in BAW-1440, the specimens in Capsule ANI-E were analyzed by Babcock and Wilcox after withdrawal from the ANO-1 vessel in the Spring of 1976. Such capsules are monitored over the life of the reactor vessel to predict the effects of radiation on the fracture toughness properties of the vessel materials. The results are used in the revision of pressure-temperature limit curves in the Technical Specifications. Adherence to the revised curves will provide assurance that no brittle failure of the reactor vessel will occur.

The specimens in Capsule ANI-E received a neutron fluence of 7.27×10^{17} n/cm^2 (neutrons per square centimeter). The test results indicated that vessel weld material is the limiting material. For this weld material, irradiation produced a shift in RT_{NDT} (as defined in paragraph NB-2331 of the ASME Code) of 137°F at the 50 foot-pound level, and a drop in upper-shelf Charpy energy from 70 to 55 foot-pounds. The drop in upper shelf energy is consistent with predicted values of USNRC Regulatory Guide 1.99, Revision 1. However, the shift in RT_{NDT} is larger than the upper limit predictions of the Regulatory Guide. The test results on correlation material specimens contained in the capsule also show a slightly higher amount of radiation damage than has resulted from similar surveillance programs. These correlation materials are part of a research program funded by the NRC and known as the Heavy Section Steel Technology (HSST) program. The specific specimens were designated HSST-PL-02.

AP&L proposed that the revised pressure-temperature limit curves be effective through the end of eight (8) EFPY (effective full power years). This proposal was based upon the RT_{NDT} upper limit adjustment line contained in Regulatory Guide 1.99. However, the NRC staff concluded that, because the shift in RT_{NDT} is greater than that predicted by the upper limit line, the temperature shift should be calculated by extrapolating the surveillance data point parallel to the prediction lines in Regulatory Guide 1.99. This method led to the conclusion that the revised curves would be acceptable through the first five (5) EFPY instead of the eight EFPY limit proposed. AP&L accepted this change, which has been incorporated.

We have concluded that the proposed changes, as modified, are acceptable and are in conformance with Appendix G to Title 10, Code of Federal Regulations, Part 50 (10 CFR 50). Conformance with Appendix G of 10 CFR 50 in establishing operating limitations will assure adequate safety margins during operation, testing, maintenance, and postulated accident conditions and constitute an acceptable basis for satisfying the requirements of NRC General Design Criterion 31 (10 CFR 50, Appendix A).

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

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.

Date: September 14, 1977

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 28 to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One - Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment modified the Technical Specification pressur temperature limit curves for hydrostatic test, normal heatup, and normal cooldown, as required by Technical Specification 3.1.2.7 and as based upon analysis of surveillance capsule ANI-E, which was withdrawn from the Arkansas Nuclear One - Unit 1 reactor vessel in Spring, 1976.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 8, 1977, (2) Amendment No. 28 to Facility Operating License No. DPR-51, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C., and at the Arkansas Polytechnic College, Russellville, Arkansas 72801. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 14th day of September, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors