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Alabama Power
the southern electric system

December 13, 1984

Docket Nos. 50-348
50-364

Director, Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Attention: Mr. S. A. Varga

Joseph M. Farley Nuclear Plant - Units 1 and 2
Results of Review of
Proposed Technical Specification Changes

Gentlemen:

By letter dated October 23, 1984, Alabama Power Company advised the NRC Staff of an administrative error in a previously submitted technical specification change request, dated December 12, 1983. By letter dated November 7, 1984, the NRC Staff issued the corrected technical specification page and requested that Alabama Power Company verify that each proposed technical specification change, which had not been issued by the NRC as an Amendment, be "proofed" to preclude further typographical errors.

Alabama Power Company has recently completed a review of each of the proposed technical specification changes currently being reviewed by the NRC Staff. Of the approximately 200 proposed Technical Specification and Bases pages reviewed, 5 pages were found to have a typographical error. None of these typographical errors altered the meaning of any of the proposed changes or made the proposed technical specifications difficult to understand. The corrected pages are provided as an Attachment to this letter. It is requested that the pages in the Attachment be incorporated into any subsequent amendments.

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Alabama Power Company recognizes and appreciates the efforts of the NRC Staff and the professionalism that has been exhibited in the review and approval of 33 License Amendments for the Farley Nuclear Plant thus far in 1984. These efforts demonstrate that technical specification changes, which enhance the safe and orderly operation of a nuclear plant, can be submitted, adequately reviewed and approved by the NRC Staff in a timely manner.

If there are any questions, please advise.

Yours very truly,

R. P. McDonald

RPM/CJS:bdh-D4
Attachment
cc: Mr. L. B. Long
Mr. J. P. O'Reilly
Mr. E. A. Reeves
Mr. W. H. Bradford

ATTACHMENT

Corrected Pages Resulting From
Review of Proposed Technical Specification Changes

Unit No.	Page	Submittal Date
Unit 1	3/4 8-12	May 3, 1983
Unit 2	3/4 8-15	May 3, 1983
Unit 1	B3/4 4-7	April 20, 1984
Unit 2	B3/4 4-7	February 10, 1984
Unit 2	B3/4 4-8	February 10, 1984

SERVICE WATER BUILDING D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The following D.C. distribution systems shall be OPERABLE and energized:

Train "A" consisting of 125-volt D.C. Distribution Cabinet 1M, 125-volt battery bank No. 1 and a full capacity charger.

Train "B" consisting of 125-volt D.C. Distribution Cabinet 1N, 125-volt battery bank No. 2 and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: With one of the 125-volt distribution trains inoperable*, restore the inoperable distribution system to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.5.2 Each 125-volt D.C. battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

   1. The parameters in Table 4.8-2 meet the Category A limits, and

   2. The total battery terminal voltage is greater than or equal to 121.2-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

   1. The parameters in Table 4.8-2 meet the Category B limits,

*Except during performance of Surveillance Requirements 4.8.2.5.2.d, 4.8.2.5.2.e, and 4.8.2.5.2.c.5. During this test, one train may be inoperable until the battery is recharged following completion of the battery discharge test.

SERVICE WATER BUILDING D.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.5 The following D.C. distribution systems shall be OPERABLE and energized:

Train "A" consisting of 125-volt D.C. Distribution Cabinet 2M, 125-volt battery bank No. 1 and a full capacity charger.

Train "B" consisting of 125-volt D.C. Distribution Cabinet 2N, 125-volt battery bank No. 2 and a full capacity charger.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION: With one of the 125-volt distribution trains inoperable*, restore the inoperable distribution system to OPERABLE status within 2 hours or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.8.2.5.1 Each D.C. train shall be determined OPERABLE and energized at least once per 7 days by verifying correct breaker alignment and indicated power availability.

4.8.2.5.2 Each 125-volt D.C. battery bank and charger shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:

 1. The parameters in Table 4.8-2 meet the Category A limits, and

 2. The total battery terminal voltage is greater than or equal to 121.2-volts on float charge.

b. At least once per 92 days and within 7 days after a battery discharge with battery terminal voltage below 110-volts, or battery overcharge with battery terminal voltage above 150-volts, by verifying that:

 1. The parameters in Table 4.8-2 meet the Category B limits,

*Except during performance of Surveillance Requirements 4.8.2.5.2.d, 4.8.2.5.2.e, and 4.8.2.5.2.c.5. During this test, one train may be inoperable until the battery is recharged following completion of the battery discharge test.

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82 and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of 7 effective full power years of service life. The 7 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{ndt}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{ndt}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{ndt} at the end of 7 EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

4) The pressurizer heatup and cooldown rates shall not exceed 100°F/hr and 200°F/hr respectively. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 320°F.

5) System preservice hydrotests and in-service leak and hydrotests shall be performed at pressures in accordance with the requirements of ASME Boiler and Pressure Vessel Code, Section XI.

The fracture toughness properties of the ferritic materials in the reactor vessel are determined in accordance with ASTM E185-82 and in accordance with additional reactor vessel requirements. These properties are then evaluated in accordance with Appendix G of the 1976 Summer Addenda to Section III of the ASME Boiler and Pressure Vessel Code and the calculation methods described in WCAP-7924-A, "Basis for Heatup and Cooldown Limit Curves, April 1975."

Heatup and cooldown limit curves are calculated using the most limiting value of the nil-ductility reference temperature, RT_{ndt}, at the end of 5 effective full power years of service life. The 5 EFPY service life period is chosen such that the limiting RT_{ndt} at the 1/4T location in the core region is greater than the RT_{ndt} of the limiting unirradiated material. The selection of such a limiting RT_{ndt} assures that all components in the Reactor Coolant System will be operated conservatively in accordance with applicable Code requirements.

The reactor vessel materials have been tested to determine their initial RT_{ndt}; the results of these tests are shown in Table B 3/4.4-1. Reactor operation and resultant fast neutron (E greater than 1 MEV) irradiation can cause an increase in the RT_{ndt}. Therefore, an adjusted reference temperature, based upon the fluence and copper content of the material in question, can be predicted using Figure B 3/4.4-1 and the recommendations of Regulatory Guide 1.99, Revision 1, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials." The heatup and cooldown limit curves of Figures 3.4-2 and 3.4-3 include predicted adjustments for this shift in RT_{ndt} at the end of 4.3 EFPY (as well as adjustments for possible errors in the pressure and temperature sensing instruments).

Values of ΔRT_{ndt} determined in this manner may be used until the results from the material surveillance program, evaluated according to ASTM E185-82, are available. Capsules will be removed in accordance with the requirements of ASTM E185-82 and 10CFR50, Appendix H. The surveillance specimen withdrawal schedule is shown in Table 4.4-5. The heatup and cooldown curves must be recalculated when the ΔRT_{ndt} determined from the surveillance capsule exceeds the calculated ΔRT_{ndt} for the equivalent capsule radiation exposure.

Allowable pressure-temperature relationships for various heatup and cooldown rates are calculated using methods derived from Appendix G in Section III of the ASME Boiler and Pressure Vessel Code as required by Appendix G to 10CFR Part 50 and these methods are discussed in detail in WCAP-7924-A.

The general method for calculating heatup and cooldown limit curves is based upon the principles of the linear elastic fracture mechanics (LEFM) technology. In the calculation procedures a semi-elliptical surface defect with a depth of one-quarter of the wall thickness, T, and a length of 3/2T is assumed to exist at the inside of the vessel wall as well as at the outside of the vessel wall. The dimensions of this postulated crack, referred to in Appendix G of ASME Section III as the reference flaw, amply exceed the current capabilities of inservice inspection techniques. Therefore, the reactor operation limit curves developed for this reference crack are conservative and provide sufficient safety margins for protection against non-ductile failure. To assure that the radiation embrittlement effects are accounted for in the calculation of the limit curves, the most limiting value of the nil ductility reference temperature, RT_{ndt}, is used and this includes the radiation induced shift, ΔRT_{ndt}, corresponding to the end of the period for which heatup and cooldown curves are generated.