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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 81 TO FACILITY OPERATING LICENSE NO. NPF-2

AND AMENDMENT NO. 73 TO FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-348 AND 50-364

1.0 INTRODUCTION

By letter dated August 11, 1986, as supplemented July 22, 1987, the Alabama Power Company (the licensee) requested amendments to Facility Operating Licenses NPF-2 and NPF-8 for the Joseph M. Farley Nuclear Plant, Units 1 and 2 (Farley Units 1 and 2). The proposed amendments would extend the expiration dates of these licenses from August 16, 2012 to June 25, 2017 for Unit 1, and from August 16, 2012 to March 31, 2021 for Unit 2. The July 22, 1987 submittal clarified certain aspects of the original request and the substance of the changes noticed in the Federal Register and the proposed no significant hazards determination were not affected.

2.0 DISCUSSION

Title 10 CFR 50.51 specifies that each license will be issued for a fixed period of time not to exceed 40 years from the date of issuance. The current terms shown in the licenses for Farley Units 1 and 2 are 40 years commencing with the issuance of the construction permits. Those permits were issued on August 16, 1972. Accounting for the time that was required for construction, the effective operating license terms were about 35 years for Unit 1 and about 31 years for Unit 2. Consistent with Section 10 CFR 50.51 of the Commission's regulations, the licensee, by the August 11, 1986 application, requested extensions of the operating license terms for Farley Units 1 and 2. This request would set the fixed periods of the licenses to be from the dates of issuance of the operating licenses rather than from the date of the construction permits.

3.0 EVALUATION

We evaluated the safety issues associated with issuance of the proposed license amendments. These proposed amendments would allow approximately five additional years of operation for Unit 1 and approximately nine additional years of operation for Unit 2. The issues addressed consist of additional radiation exposure to the licensee's operating staff, potential increases in evaluated impacts on the offsite population, and the general increase in the aging of plant structures and equipment. The impact of additional radiation exposure to the facility operating staff and the impact on the general population in the vicinity of the Farley Nuclear Plant are addressed in the NRC staff's Environmental Assessment dated May 12, 1989.

3.1 MECHANICAL EQUIPMENT

The components of the reactor coolant system pressure boundary were designed, built and tested to the appropriate ASME Boiler and Pressure Vessel Codes, Regulatory Standards, and supplemental criteria in compliance with the requirements of 10 CFR Part 50, Section 50.55a, "Codes and Standards." The initial inservice inspection (ISI) program was described in the Final Safety Analysis Report (FSAR). The FSAR description and the associated Technical Specifications comply with the requirements of Section 50.55a(g).

Subsequently, for Farley Unit 1 the licensee revised their first 10-year (120-month) ISI Program including the portion on testing of pumps and valves. Those programs were in accordance with Article IWA-6220 of Section XI of the 1974 Edition through the Summer 1975 Addenda of the ASME Boiler and Pressure Vessel Code. Following the NRC staff review of the first 10-year ISI Program for Farley Unit 1, certain reliefs from code requirements were granted by our letter dated December 7, 1979. That program was satisfactorily completed, and the second 10-year updated program was submitted by licensee letter dated May 27, 1987. This updated program, in accordance with the ASME Code, Section XI, 1983 Edition through the Summer 1983 Addenda, is under review by the NRC staff. Interim approval of that second 10-year ISI Program was granted by our letter dated March 31, 1989.

For Farley Unit 2, our review of the first 10-year ISI Program, provided by licensee letter dated July 25, 1980, was completed and certain ASME Code reliefs granted by our letter dated September 22, 1983. That first 10-year ISI Program was in effect until superseded by an upgraded program put into effect for the sixth refueling outage (April 1989). The upgraded program is in accordance with ASME Code, Section XI, 1983 Edition through Summer 1983 Addenda, which also applies to Farley Unit 1. By letter dated March 31, 1989, the NRC staff granted interim approval of the upgraded programs including the testing program for pumps and valves.

We conclude from our evaluations that compliance with the codes, standards, and regulatory requirements to which the mechanical equipment for Farley Units 1 and 2 was originally analyzed, constructed, tested and inspected, (including Boiler and Pressure Vessel Code and the other augmented inspections of austenitic stainless steel piping) provides adequate assurance that the structural integrity of components important to safety will be maintained during additional periods authorized by these proposed amendments. Any significant degradation by an active mechanism would be discovered during the required testing of equipment or components. Thereafter, the equipment or component would be restored to an acceptable condition. Therefore, the age of the mechanical equipment and components should be satisfactory for the proposed extensions of the operating licenses for Farley Units 1 and 2.

3.2 STRUCTURES

The concrete and steel Category I structures for Farley Units 1 and 2 were designed and constructed in accordance with the Commission's General Design Criteria, Appendix A, 10 CFR Part 50, as amended July 7, 1971. The design bases, fabrication, construction, and quality assurance criteria for the plant were previously reviewed by our staff. These staff evaluations are contained in the Safety Evaluation Report (SER), NUREG 75/034, "Safety Evaluation Report Joseph M. Farley Nuclear Plant Units 1 and 2," dated May 2, 1975, through Supplement No. 6, dated March 1981. Industrial experience with concrete and steel structures confirms that a service life in excess of forty years may be anticipated without significant degradation.

The major codes and specifications used in the design and construction of the Category I concrete and steel structures were American Concrete Institute (ACI) 318-63/71, "Building Code Requirements for Reinforced Concrete," and the American Institute of Steel Construction (AISC) Specification, "Specification for the Design, Fabrication, and Erection of Structural Steel for Building." Support structures were constructed to the requirements of Subsection NF, Section III, of the ASME Boiler and Pressure Vessel Code. Section 3.8 of our SER states that the criteria used in the analysis, design and construction of the Farley plant account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetimes. These criteria are in conformance with established criteria, codes standards and specifications acceptable to the NRC staff.

The use of the indicated codes, standards, and specifications in the design, analyses, and construction; Appendix B of 10 CFR Part 50 for quality assurance; and the identified testing and inservice surveillance requirements provide reasonable assurance that the concrete and steel structures would withstand continued service for the proposed license extension without significant degradation of structural integrity.

3.3 REACTOR VESSELS

The FSAR states that the reactor vessels (RV) for Farley Units 1 and 2 were designed and fabricated for a service life of forty years at 80% plant capacity (32 effective full power years). The vessels are Safety Class 1. They were designed, fabricated and inspected in accordance with the requirements of Section III, Class 1, of the ASME Boiler and Pressure

Vessel Code edition, addenda, and Code Cases applicable at the time of purchase. Operating limitations of the ASME Boiler and Pressure Vessel Code and Appendix G of 10 CFR Part 50 are also applicable. The inservice inspection program is periodically upgraded to comply with the requirements of Section 50.55a(g), 10 CFR Part 50, that incorporates Section XI of the ASME Boiler and Pressure Vessel Code.

The integrity and performance capability of the ferritic materials in the RV for Farley Units 1 and 2 is assured because the fracture toughness is monitored with a surveillance program in conformance, to the extent practical, with the requirements of Appendix H, 10 CFR Part 50, "Reactor Vessel Materials Surveillance Program Requirements," and ASTM, "Standard Practice for Conducting Surveillance Tests for Light-Water Cooled Nuclear Power Reactor Vessels." The ferritic materials must meet the fracture toughness properties of Section III of the ASME Boiler and Pressure Vessel Code and Appendix G, 10 CFR Part 50, "Fracture Toughness Properties."

To date three material specimen capsules have been removed from the Farley Unit 1 RV through refueling outage number eight (May 1988). The third capsule is being removed from Farley Unit 2 during the sixth refueling outage (April 1989). These specimen removals are required by Technical Specification 4.4.10.1.2 to determine the changes in RV material properties in accordance with Appendix H, 10 CFR Part 50. Changes in these RV material properties affect the reactor coolant system heatup and cooldown limits. These limits are revised, as necessary, to be consistent with requirements of the ASME Code, Section III, and Appendix G, 10 CFR Part 50. Thus, the integrity of the RV must remain in compliance with applicable safety codes throughout the proposed operating lifetime.

In addition, regulatory changes made to 10 CFR 50.61 (50 FR 29944 dated July 23, 1985) relating to continued compliance of the RV to the Pressurized Thermal Shock (PTS) rule have been evaluated for Farley Units 1 and 2. By letter dated January 20, 1986, the licensee submitted WCAP-11047, dated January 1986 for NRC staff review. Our review concluded that on Farley Unit 1, for the projected 32 effective full power years (EFPY) (an estimated 40 years at 80% plant capacity), the RV fluence is almost six times smaller than that fluence required to reach the PTS screening criteria. Thus, the Farley Unit 1 RV meets our criteria and is acceptable. For the Farley Unit 2 RV, the RV fluence factor would have to increase by almost two prior to reaching the screening criteria. Therefore, the Farley Unit 2 RV meets our criteria and is acceptable, also, for the extension in license term proposed.

Based on these considerations, we conclude that no special considerations would exist to indicate unexpected RV degradation for Farley Units 1 and 2 due to the proposed operating lifetime extensions. The structural integrity of the RV is assured because they were originally designed for 32 EFPY usage (40 years at 80% plant capacity); they are monitored, inspected and tested to detect degradation processes at an early stage of development; and they are operated with procedures to assure that design conditions are not exceeded during later periods of operation, as described herein.

3.4 ELECTRICAL EQUIPMENT

The safety related electrical equipment and components have been evaluated to meet the NRC requirements of IE Bulletin 79-01B, NUREG-0588 and 10 CFR 50.49. The environmental qualification (EQ) programs submitted to us for Farley Unit 1 by letter dated June 30, 1980 and for Farley Unit 2 by letter dated September 12, 1980 document service life expectancy as 40 years. This includes operation in the most severe normal environment, as well as the environment which could exist in a potential design basis accident.

On the basis of our review of the EQ Programs and their continuing administrative controls to assure that safety-related electrical equipment and components would continue to perform their designed safety functions, the Farley Units 1 and 2 electrical equipment is acceptable for the extended time periods proposed for the licenses.

3.5 SUMMARY OF FINDINGS

The NRC staff has concluded in its associated Environmental Assessment that the annual radiological effects during the additional years of operation proposed are not significantly greater than were previously estimated in the Final Environmental Statement. These radiological effects are acceptable.

The Exclusion Area for the Farley site consists of property wholly owned by Alabama Power Company. The licensee controls all activities within the exclusion area and anticipates no changes to the exclusion area boundary during the extended license periods. Projected changes in population within the Low Population Zone (LPZ), nearest population center distances and 10 mile radius Emergency Planning Zone (EPZ) have been found not to be significant for the period of the license extensions. Accordingly, the Commission's conclusions regarding 10 CFR Part 100 siting criteria for Farley Units 1 and 2 are that the exclusion area, the LPZ, and population center distances meet the guidelines of 10 CFR Part 100 and are not changed by the proposed license extensions.

We also conclude from our considerations of the design, operation, testing and monitoring of the mechanical equipment, structures, reactor vessels, and electrical equipment and components that an extension of the operating licenses for Farley Units 1 and 2 to a 40-year service life is consistent with the FSAR, SER, as supplemented, and submittals made by the licensee. There is reasonable assurance that these units will continue to operate safely for the additional periods authorized by these amendments. The plants are operated in compliance with the Commission's regulations, and issues associated with plant degradation have been adequately addressed herein and in the previously issued evaluations relating to this matter.

4.0 ENVIRONMENTAL CONSIDERATION

A Notice of Issuance of Environmental Assessment and Finding of No Significant Impact relating to the proposed extension of the Facility Operating License termination dates for Farley Units 1 and 2 was published in the Federal Register on May 19, 1989 (54 FR 21686).

5.0 CONCLUSION

The Commission made a proposed determination that the amendments involve no significant hazards consideration which was published in the Federal Register September 24, 1986 (51 FR 33939), and consulted with the State of Alabama. No public comments were received, and the State of Alabama did not have any comments.

The staff has concluded, based on the considerations discussed above, that: (1) three 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 these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: Edward A. Reeves

Dated: May 19, 1989