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JUN 23 1982

Docket No. 50-364

Mr. F.L. Clayton
Senior Vice President
Alabama Power Company
Post Office Box 2641
Birmingham, Alabama 35291

Dear Mr. Clayton:

The Commission has issued the enclosed Amendment No. 14 to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant, Unit No. 2. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated December 18, 1981, supplemented by letters dated February 1, March 19, April 5, and April 21, 1982.

The amendment allows an increase in the storage capacity of the Spent Fuel Pool from 675 to 1407 storage locations.

Copies of the Safety Evaluation, Environmental Impact Appraisal, and the Notice of Issuance are also enclosed.

Sincerely,

ORIGINAL SIGNED
Edward A. Reeves, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 14 to NPR-8
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice of Issuance

cc: w/enclosures
See next page

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

ALABAMA POWER COMPANY

DOCKET NO. 50-364

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 14
License No. NPF-8

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

   A. The application for amendment by Alabama Power Company (the licensee) dated December 18, 1981, supplemented by letters dated February 1, March 19, April 5, and April 21, 1982, 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. NPF-8 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 14 , are hereby incorporated in the 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

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 23, 1982

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 14 TO FACILITY OPERATING LICENSE NO. NPF-8

DOCKET NO. 50-364

Revise Appendix A as follows:

Remove Pages	Insert Pages
5-7	5-7

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases based on a maximum enrichment of 4.3 weight percent U-235.

b. A nominal 10.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that K_{eff} will not exceed 0.98, based on a maximum enrichment of 3.5 weight percent U-235, assuming aqueous foam moderation.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 149.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1407 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limits of Table 5.7-1.

ENCLOSURE 2

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

SAFETY EVALUATION BY THE OFFICE OF
NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE
SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. NPF-8
ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT - UNIT 2
DOCKET NO. 50-364

CONTENTS

1.0 Introduction

By letter dated December 18, 1981, as supplemented February 1, March 19, April 5, and April 21, 1982, Alabama Power Company (APCo) (the licensee) requested an amendment to Facility Operating License No. NPF-8 for Joseph M. Farley Nuclear Plant Unit No. 2. The request would revise the radiological Technical Specifications to allow an increase in the spent fuel pool (SFP) storage capacity from 675 to a maximum of 1407 fuel assemblies through the use of neutron absorbing "poison" spent fuel storage racks.

The expanded storage would allow Farley Unit 2 to operate until the year 2006 with capability for a full core discharge, assuming annual one-third core reloads.

The major safety considerations associated with the proposed expansion of SFP storage capacity are addressed below. A separate Environmental Impact Appraisal has been prepared as part of this licensing action.

2.0 Background

The SFP currently contains racks with a capacity of 675 fuel assemblies. The proposed expansion modification will increase the existing storage capacity to 1407 fuel assemblies. For simplification of the work involved, APCo has proposed a schedule for reracking that would allow all modifications to be completed before the first scheduled refueling outage (November 1982). This simplifies the modification because currently the SFP and racks are clean, dry and uncontaminated. Thus, standard construction procedures can be used.

3.0 Discussion and Evaluation

APCo proposed to replace the existing storage racks in the SFP with high density, stainless steel, fixed poison type, free standing storage racks. The storage racks will have three basic module configurations with dimensions of 6 x 7, 7 x 7, and 7 x 8 feet, and weights of 6 3/4 tons, 7 9/10 tons, and 9 tons, respectively. There will be two 6 x 7 modules, nineteen 7 x 7 modules and seven 7 x 8 modules.

The individual poison cans or cannisters of the modules are formed using 0.024 inch thick sheets of stainless steel wrapped around a neutron absorbing material vented boraflex sandwiched in between the two sheets of stainless steel. The center-to-center spacing of the cans will be 10.75 inches. A water plenum is provided by supporting the modules at their four corners by stainless steel support feet equipped with large leveling screws. Pool water will flow down along the pool walls where it will enter the water plenum, and then travel laterally where it enters the bottom of the storage cans through the bottom grid.

3.1 Criticality Considerations

The criticality aspects of the proposed high density spent fuel racks have been analyzed using the PDQ-7 diffusion theory code for purposes of scoping and design. The KENO-IV Monte Carlo code with AMPX cross section code has been used to verify the final design. These codes have been benchmarked against experiment and a calculational bias, as well as calculational and mechanical uncertainties were obtained.

The effective multiplication factor for the racks was calculated under the assumption of fresh fuel of 4.3 weight percent U-235 enrichment (54.25 grams of U-235 per centimeter of assembly length) at a pool temperature of 68 degrees Fahrenheit. No credit is taken for control rods or any noncontained burnable poison in the Westinghouse 17 x 17 fuel assemblies and the fuel racks are assumed to be infinite in extent. Under these assumptions the nominal effective multiplication factor for the storage racks in their design configuration is 0.9217 as determined by the KENO-IV code. To this value must be added a calculational bias of 0.0027 (obtained from benchmark comparisons) and a total uncertainity of 0.0159 (obtained by a statistical combination of the calculational and mechanical uncertainties). The mechanical uncertainity accounts for variations in center-to-center spacing, B-10 loading in the poison plates, and U-235 enrichment. After all uncertainties are added, the resulting value of the effective multiplication factor is 0.9403. This meets our acceptance criteria for criticality calculations of 0.95 including all uncertainties. The calculational uncertainty is such that the true multiplication factor will be less than the calculated value with a 95 percent probability of a 95 percent confidence level.

The effect of credible accidents has been calculated and the most consequential one is the dropping of a single fuel assembly outside the rack between the periphery of the storage racks and the side walls of the pool. The effective multiplication factor remains below 0.95 for this accident with all uncertainties and biases included. The pool water was assumed to contain soluble boron for this analysis. This is permitted by the double contingency principle of ANSI N16.1-1975 "American National Standard for Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors" which states that two unlikely, independent, concurrent events are required to produce a criticality accident. The staff has accepted this principle in previous safety evaluations.

We conclude that the proposed storage racks meet the requirements of General Design Criterion 62 regards criticality. This conclusion is based on the following considerations:

1. state-of-the-art calculation methods which have been verified by comparison with experiment have been used;

2. conservative assumptions have been made about the enrichment of the fuel to be stored and the pool conditions;

3. credible accidents have been considered;

4. suitable uncertainties have been considered in arriving at the final value of the multiplication factor; and

5. the final effective multiplication factor value meets our acceptance criterion.

We also conclude that the proposed modifications to Technical Specification 5.6.1.1 increasing the maximum allowable enrichment in the spent fuel pool to 4.3 weight percent U-235 and reducing the nominal center-to-center distance between fuel assemblies in the storage racks to 10.75 inches are acceptable. The proposed Technical Specification 5.6.3 which allows an increase in the spent fuel storage pool capacity from 675 to 1407 fuel assemblies is also acceptable for the high density storage racks described in the Farley Unit 2 Spent Fuel Pool Modification Report dated December 1981. The maximum fuel enrichment presently allowed in the new fuel pit storage racks is 3.5 weight percent U-235 (Technical Specification 5.6.1.2). Therefore, the higher enriched, extended cycle fuel of 4.3 weight percent enrichment can be stored only in the proposed high density spent fuel storage racks at present.

Our evaluation is based on PWR fuel pins and fuel assemblies similar in design to the Westinghouse fuel presently installed in Farley Unit 2. Fuel designs differing from this would require a reevaluation even though the U-235 enrichment and fuel assembly spacing specifications are not violated.

3.2 Spent Fuel Cooling

3.2.1 Introduction

The Spent Fuel Pool Cooling System consists of two pumps and two heat exchangers. One pump and one heat exchanger is used for normal operation and the second pump and heat exchanger serves as a backup. The heat exchangers are cooled by the component cooling water system. The SFP cooling connections to the pool are provided with anti-siphon holes or located in such a manner than protects against inadvertent drainage of the pool to less than 4 feet below the normal level of 24 feet above the fuel. In event of a loss of the cooling system, makeup is available from the seismic Category I reactor water makeup system.

The future refueling cycle for Farley Unit 2 will be a twelve month period and one-third of the core will be removed and stored in the SFP after each cycle. To limit the decay heat load, the 1/3 core will be removed from the reactor vessel and stored in the SFP, 100 hours after reactor shutdown. In the event of a full-core discharge, the decay heat load will be limited by requiring a 10 day decay time after shutdown before core discharge.

3.2.2 Evaluation

To calculate the heat loads for the discharges of spent fuel to the pool, APCo used Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling." The maximum normal heat load which occurs after the twenty-seventh refueling discharge, was calculated to be 19.71 x 10^6 BTU/HR. The normal heat load resulted in a maximum bulk pool temperature of approximately 139°F with one cooling train operating which is in compliance with Standard Review Plan Section 9.1.3, "Spent Fuel Pool Cooling and Cleanup System." The maximum abnormal heat load results from a full-core discharge after the last normal refueling discharge was calculated to be 30.33 x 10^6 BTU/HR. The abnormal heat load resulted in a maximum bulk pool temperature of approximately 158°F with one train operating and 131°F with two trains operating. The American National Standard 57.2 "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations" indicates that the maximum pool temperature should not exceed 150°F under normal operating conditions with all storage full. The design, therefore, meets this standard.

To verify that natural circulation of the pool water for the proposed expanded rack configuration provides adequate cooling of all fuel assemblies in the event of a loss of external cooling, APCo performed a thermal-hydraulic analysis. The adequacy of natrual circulation was verified using the computer code BPOOL. BPOOL is a proprietary program of Nuclear Associates Incorporated (NAI). The code is based on the assumption that boiling takes place near the top of the fuel channel. BPOOL evaluates the saturation properties of the coolant on the basis of the static pressure at the top of the storage racks. These properties include water density, temperature, and steam density. The steam is assumed to separate and flow out of the pool. The water at the saturation temperature corresponding to the pressure at the top of the racks flows downward to the inlet of the storage racks. The static pressure at this location is higher than the pressure at the top of the storage racks and as a result the fluid is subcooled as it enters the fuel assembly. The fluid becomes less dense as it passes up the fuel channel. Near the top of the fuel channel the fluid reaches saturation conditions and net boiling occurs. Thus natural circulation will be maintained and the flow is adequate as verified by the computer code BPOOL.

Under normal cooling conditions (external cooling available) natural circulation cooling of the spent fuel was verified using the NAI computer code HPOOL. HPOOL calculates the pressure loss through a fuel assembly for a given flow rate. This pressure loss is compared with the buoyant head resulting from the difference between the average density of the fluid in the fuel channel and the average density of the fluid in the downcomer (space between the pool wall and the racks). If the density difference results in a buoyant head greater than the pressure loss, the flow rate through the fuel assembly is increased and a new average density of the fluid is determined. This iterative process is continued until the buoyant head and pressure loss in the fuel assembly are equal. Using this flow rate, HPOOL determines the fuel temperature.

In the event of a complete failure of the SFP cooling system, even for the maximum abnormal heat load there is at least 4 hours available before makeup water to the pool is required. The maximum required makeup rate is between 50 and 60 gpm. Each of four makeup water sources can be initiated in the required time. The reactor water makeup tank supply can be provided to the pool by either of two 165 gpm reactor water makeup pumps. The reactor water makeup tank, piping, and the makeup pumps are seismic Category I. Sufficient makeup rates are also available from the refueling water storage tank (via two paths) and the demineralized water system; however neither source is completely seismic Category I.

3.2.3 Conclusion

We have reviewed the calculated decay heat values and conclude that the heat loads are consistent with the Branch Technical Position ASB 9-2 and therefore are acceptable. The SFP cooling system performance and the natural circulation assumptions have been reviewed and we conclude that the pool cooling is adequate. The available makeup systems, their respective makeup rates and the time required before makeup is needed has been reviewed and found acceptable. Based on the above, we conclude that the SFP cooling system is acceptable.

3.3 Installation of Racks and Fuel Handling

During the expansion program, a temporary crane will be used to remove all of the present racks and insert the new racks. Since there is no fuel in the present racks, which are uncontaminated, the installation will be done dry. There is also no equipment, essential in the safe shutdown of the reactor or essential to mitigate the consequence of an accident, which is beneath, adjacent to or otherwise within the area of influence or any loads that will be handled during the expansion modification. Based on the above, we conclude that handling of the present racks and new racks is acceptable.

3.4 Structural and Seismic Loadings

3.4.1 Introduction

The Farley Unit 2 SFP is an existing reinforced concrete box structure. The walls of the pool very in thickness from about 3.5 feet to 7.5 feet. The floor is 5 feet thick and rests on 9.5 foot long columns surrounded by fill concrete, which in turn, are supported by a 5 foot thick base slab, which rests on rock. The inside dimensions are approximately 40.5 feet deep by 27 feet wide by 45 feet long. The pool is lined with a water-tight, continuous, 1/4 inch thick, stainless steel plate.

The new spent fuel storage racks are to be constructed of 300 series stainless steel with vented "Boraflex" poison material sandwiched between stainless steel sheets. The racks are vertical "egg-crate" structures, each of which is free-standing on four pads on the pool floor. A 7 x 8 rack (56 cells) would be approximately 14.9 feet high by 72 feet wide by 6.3 feet long. The pitch of all cells will be

10.75 inches, center-to-center. The racks are individually installed with the bottom grids of adjacent racks butting to one another leaving a nominal 5/8 inch gap at the top. The minimum clearance between a rack and the pool wall is to be approximately 3 inches while the maximum is about 9 inches.

The design, fabrication, installation and quality assurance standards for the new spent fuel racks are compared with the staff's "OT Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications" dated April 1978 and including revisions dated January 1979 (to be referred to henceforth as the "OT Position").

The racks are designed in accordance with the requirements of the American Institute of Steel Construction (AISC) Manual which is an acceptable alternative in the OT Position.

3.4.2 Seismic and Impact Loads

The SFP floor response spectra used for the seismic analysis were as provided in the Farley Unit 2 FSAR and approved as part of the license review. A computer program, "SIMQKE", was then used to develop artificial time histories from these spectra. Damping values of 2 percent for OBE and 5 percent for SSE, which are plant specific values and previously approved in the plant license review were used. The dynamic model, consisting of springs, masses, gaps, and damping elements for a double rack system includes the potential for rack-to-rack interaction, fuel-to-rack interaction and floor-to-rack interaction. The seismic time history analysis was conducted using a coefficient of friction between the pool and rack of 0.2 in order to define maximum credible sliding. The analysis was also performed using a coefficient of friction of 0.8 in order to define a worst case loading condition.

The spacing of the racks is such that rack-to-rack impacts may occur in some modes; however, in all cases, stresses are maintained within allowable limits.

Fuel casks cannot be transported over the pool due to built in physical constraints. The old racks will be removed and the new racks installed before any radioactive material is placed in the pool.

Existing Technical Specification 3.9.7.1 prohibits transporting loads greater than 3000 pounds over the spent fuel pool; therefore, the heaviest load that will be carried over the pool is a fuel bundle. Impact loading on the racks from a fuel bundle drop was considered for the required conditions and combined with dead loads and live loads at suitable thermal levels. Results were satisfactory.

3.4.3 Load and Load Combinations

Loads, load combination were compared with the criteria outlined in SRP Section 3.8.4 and found to be acceptable.

3.4.4 Design and Analysis Procedures

As described above, dynamic analyses of the rack and pool were conducted using lumpsed masses, spring elements, gap elements and damping elements to model the systems. Hydrodynamic effects were considered. Various loading configurations of fuel in the racks were considered in order to define worst-case conditions. In addition, a finite element analysis of the racks, using forces developed from the dynamic analysis, was accomplished. The racks are not attached to the pool walls and the pool itself is founded on bedrock, therefore, any motion of the pool walls will not directly amplify the rack seismic motions. Seismic loads were imposed simultaneously in three orthogonal directions on the computer models in the dynamic analyses.

APCo's analysis includes consideration of the loads, acting upward, of stuck fuel assembly as it is being lifted out of the rack. For this case, no permanent deformation of the rack is allowed.

3.4.5 Structural Acceptance Criteria

The structual acceptance criteria outlined in the applicant's submittal was compared to that outlined in SRP Section 3.8.4 II.5 and was found to be in conformance.

3.4.6 Materials, Quality Control, and Special Construction Techniques

With the exception noted previously, all materials are in accordance with the ASME Code, as are fabrication, and inspection procedures.

3.4.7 Conclusion

We find that the subject modification with respect to Structural and Seismic Loadings, proposed by the licensee is acceptable and satisfies the applicable requirements of the General Design Criteria 2, 4, 61, and 62 of 10 CFR, Part 50, Appendix A, regarding such structures.

3.5 Materials Evaluation

3.5.1 Structural Aspects

APCo proposes to use austinitic stainless steel conforming to ASTM A666, Grade B, for certain portions of the racks. This material specification is not found in the ASME Code. The staff's position is that all rack material should conform to all applicable requirements of Section III, Division 1, Subsection NF of the ASME Code.

APCo has committed to qualify the rack material in question to ASME Code Subsection NF (material specification SA240) in all respects, and in addition, to obtain valid test results to justify the higher yield stress allowed by ASTM-A666, Grade B. APCo has alos furnished test results and cited experience with this material to satisfy staff concerns. Complete documentation of material quality will be maintained. This is acceptable to the staff.

3.5.2 Corrosive Aspects

3.5.2.1 Introduction

We have reviewed the compatability and chemical stability of the materials, except the fuel assemblies, wetted by the pool water. The proposed SFP storage racks are fabricated primarily of Type 304 stainless steel, which is used for all structural components, except for part of the bottom grid where Type 17-4 PH given the H-1100 heat treatment and a cast stainless steel CF8 are used in selected components. The neutron absorber material is boraflex, which is held firmly between a stainless steel structural can and a stainless steel inner wrapper. The compartments in the storage racks containing the boraflex are exposed to the spent fuel pool environment through small openings formed during fabrication in the top and bottom of each tube assembly. The water chemistry in the SFP has been reviewed elsewhere and found to meet NRC specifications. Type 304 stainless steel rack modules have been welded and inspected by nondestructive examinations performed in accordance with the applicable provisions of ASME Boiler and Pressure Vessel Code, Section 9. APCo will perform a materials compatability monitoring program consisting of 10 coupons which duplicate the condition of boraflex which is encased in the poison canisters. These coupons are to be hung alongside the high density fuel racks and will be subjected to the maximum neutron, gamma, and heat fluxes. Sufficient coupons are included to permit destructive examination of a sample on inspection intervals of 1 to 5 years over the life of the facility.

3.5.2.2 Evaluation

The Unit 2 SFP is fabricated of materials that will have good combatibility with the borated water chemistry of the spent fuel pool. The corrosion rate of Type 304 stainless steel in this water is sufficiently low to defy our ability to measure it. Since all materials in the pools are stainless steel, no galvanic corrosion effects are anticipated. No instances of corrosion of stainless steel in spent fuel pools containing boric acid has been observed throughout the country(1). Boraflex has been shown to be resistant to radiation doses in excess of any anticipated in the SFP. The venting of the cavities containing the boraflex to the SFP environment will ensure that no gaseous buildup will occur in these cavities that might lead to distortion of the racks. The type 17-4 PH stainless steel in the threaded feet of the racks has been given an H-1100 heat treatment, in which condition it is resistant to stress corrosion cracking in SFP environments. The Codes and Standards used in fabricating and inspecting these new fuel storage racks should ensure their integrity and minimize the likelihood that any stress corrosion cracking will occur during service. The materials surveillance program proposed by APCo will reveal any instances of deterioration of the boraflex that might lead to the loss of neutron absorbing power during the life of the new spent fuel racks. We to not anticipate that such deterioration will occur.

This monitoring program will ensure that, in the unlikely situation that the boraflex will deteriorate in this environment, the licensee and the NRC will be aware of it in sufficient time to take corrective action.

3.5.2.3 Conclusion

From our evaluation as discussed above, we conclude that the corrosion that will occur in Unit 2 SFP will be of little significance during the remaining life of the unit. Components of the spent fuel storage pool are constructed of alloys which are known to have a low differential galvanic potential between them, and that have performed well in spent fuel storage pools at other pressurized water reactor sites where the water chemistry is maintained to comparable standards to those in force at Farley. The proposed materials surveillance program is adequate to provide

(1) J. R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-23021, July, 1977.

warning in the unlikely event that deterioration of the neutron adsorbing properties of the boraflex will develop during the design life of the racks. Therefore, with the selection of the materials we believe that no significant corrosion should occur in the spent fuel storage racks at Farley Unit 2 for a period well in excess of the 40 years design life of the unit.

Therefore, we conclude that the compatability of the materials and coolant used in the spent fuel storage pool is adequate based on tests, data, and actual service experience in operating reactors. We find that the selection of appropriate materials by the licensee meets the requirements of 10 CFR Part 50, Appendix A, Criterion 61, by having a capability to permit appropriate periodic inspection and testing of components, and Criterion 62, by preventing criticality by maintaining structural integrity of components, and is therefore acceptable.

3.6 Occupational Radiation Exposure

We have reviewed APCo's plan for the removal and disposal of the low density racks and the installation of the high density racks with respect to occupational radiation exposure. Since the SFP for Farely Unit 2 has never had spent fuel stored in it and is currently dry, clean and uncontaminated, there will be no additional radiation exposure to worker due to the SFP modification. Thus, the staff concludes that SFP modification exposure to workers is as low as is reasonably achievable (ALARA) and acceptable.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies at Farley Unit 2 on the basis of information supplied by the licensee and by utlizing relevant assumptions for occupancy times and for dose rates in the SFP area from radionuclide concentrations in the SFP water. The spent fuel assemblies themselves contribute a negligible amount to dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the SFP area, we estimate that the proposed modification should add less than one percent to the total annual occupational radiation dose at the unit. This small increase in radiation dose in the SFP area should not affect the licensee's ability to maintain individual occupational doses to ALARA levels and within the limits of 10 CFR Part 20. Therefore, we conclude that storing additional fuel in the Unit 2 SFP will not result in any significant increase in doses received by workers.

3.7 Radioactive Waste Treatment

3.7.1 Introduction

The SFP cleanup system is designed to remove corrosion products, fission products and impurities from the pool water with mixed bed demineralizers and filters. Pool water purity is monitored weekly by chemical and radiochemical analysis. Demineralizer resin will be replaced when pool water samples show demineralizer reduced decontamination effectiveness. The SFP filters will be exchanged when ΔP exceeds 20 psia. The licensee indicated that no change or equipment addition to the SFP cleanup system is necessary to maintain pool water quality and optical clarity for high density fuel storage.

3.7.2 Evaluation

Past experience showed that the greatest increase in radioactivity and impurities in SFP water occurs during refueling and spent fuel handling. The refueling frequency, the amount of core to be replaced for each fuel cycle, and frequency of operating the SFP cleanup system are not expected to increase as a result of high density fuel storage. The chemical and radionuclide composition of the SFP water is not expected to change as a result of the proposed high density fuel storage. Past experience also shows that no significant leakage of fission products from spent fuel stored in pools occurs after the fuel has cooled for several months. To maintain water quality, the licensee has established the frequency of chemical and radionuclide analysis that will be performed to monitor the water quality and the need for SFP cleanup system demineralizer resin and filter replacement. In addition, the licensee has also set the chemical and radiochemical limits to be used in monitoring the SFP water quality and initiating corrective action.

On the basis of the above, we determined that the proposed expansion of the SFP will not appreciably affect the capability and capacity of the SFP cleanup system. More frequent replacement of filters or demineralizer resin, required when the differential pressure exceeds 20 psid or decontamination effectiveness is reduced to less than 10 (decontamination factor), can offset any potential increase in radioactivity and impurities in the pool water as a result of the expansion of stored spent fuel. Thus we have determined that the existing fuel pool cleanup system with the proposed high density spent fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool and thus meets the requirements of General Design Criterion 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate systems to spent fuel storage; (2) is capable of reducing

occupational exposures to radiation by removing radioactive products from the pool water, and thus meet the requirements of Section 20.1(c) of 10 CFR Part 20, as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water with the filters and demineralizers, and thus meets Regulatory Position (C.2.f(2) of Regulatory Guide 8.8, as it relates to reducing the spread of containments from the sources; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action.

3.7.3 Conclusion

On the basis of the above evaluation, we conclude that the existing spent fuel pool cleanup system meets GDC 61, Section 20.1(c) of 10 CFR Part 20 and the appropriate sections of Regulatory Guide 8.8 and, therefore, is acceptable for the proposed high density spent fuel storage.

4.0 Conclusions

On the basis of the foregoing analysis, it is concluded that there will be no significant environmental impact attributable to the proposed action. Having made this conclusion, the Commission has further concluded that no environmental impact statement for the proposed action need be prepared and that a negative declaration to this effect is appropriate.

We have concluded, based on the considerations discussed above, that: (1) there 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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ENCLOSURE 3

ENVIRONMENAL IMPACT APPRAISAL BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE

SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. NPF-8

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2

DOCKET NO. 50-364

CONTENTS

1.0 INTRODUCTION AND DISCUSSION

A Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575, Volumes 1-3) was issued by the Nuclear Regulatory Commission (NRC) August 1979. The NRC staff evaluated and analyzed alternative handling and storage of spent light-water power-reactor fuel with emphasis on long range policy. Consistent with the long range policy, the storage of spent fuel addressed in the FGEIS is considered to be interim storage to be used until the issue of permanent disposal is resolved and implemented.

One spent fuel storage alternative considered in detail in the FGEIS is the expansion of the onsite fuel storage capacity by modification of the existing spent fuel pools (SFPs). On the date of issuance of the FGEIS (August 1979), 40 applications for SFP capacity expansions were approved with the finding in each case that the environmental impact of the proposed increased storage was negligible. However, since there are variations in storage pool designs and limitations caused by the spent fuel already stored in some of the pools, the FGEIS recommends that licensing reviews be done on a case-by-case basis to resolve plant specific concerns.

In addition to the alternative of increasing the storage capacity of the existing SFPs, other spent fuel storage alternatives are discussed in detail in the FGEIS. The finding of the FGEIS is that the environmental impact-costs of interim storage are essentially negligible, regardless of where such spent fuel is stored. A comparison of the impact-costs of the various alternatives reflect the advantage of continued generation of nuclear power versus its replacement by coal fired power generation. In the bounding case considered in the FGEIS, where spent fuel generation is terminated, the cost of replacing nuclear stations before the end of their normal lifetime makes this alternative uneconomical.

This Environmental Impact Appraisal (EIA) incorporates the appraisal of environmental concerns applicable to expansion of the Farley Unit 2 SFP.

For additional discussion of the alternatives to increasing the storage capacity of existing SFPs, refer to the FGEIS. This EIA consists of three major parts plus a summary and conclusion. The three parts are: (1) descriptive material, (2) an appraisal of the environmental impacts of the proposed action, and (3) an appraisal of the environmental impact of postulated accidents.

1.1 Description of the Proposed Action

By application dated December 18, 1981, as supported by letters dated February 1, March 19, April 5, and April 21, 1982, Alabama Power Company (APCo) (the licensee) requested an amendment to Facility Operating License No. NPF-8 for the Joseph M. Farley Nuclear Plant Unit 2 (Farley Unit 2). The proposed amendment would allow an increase in the storage capacity of the Farley Unit 2 Spent Fuel Pool (SFP) from 675 to 1407 storage locations.

In order to avoid any unnecessary personnel radiation exposure and to eliminate the generation of additional large quantities of radwaste, APCo has proposed a schedule for re-racking the Farley Unit 2 spent fuel pool prior to the first refueling outage (currently scheduled to begin November 1, 1982). This schedule will allow the pool to be re-racked under dry conditions utilizing conventional construction practices without the possibility of radiation exposure to installation personnel. Re-racking, prior to the first refueling will also eliminate the remote possibility of a fuel handling mishap or damage to a spent fuel assembly which could occur during wet re-racking after the first refueling due to the necessity to shuffle spent fuel within the pool.

The environmental impacts of Farley Unit 2 as designed, were considered in the Final Environmental Statement (FES) issued in December 1974. The purpose of this EIA is to determine and evaluate any additional environmental impacts which are attributable to the proposed increase in the SFP storage capacity of the plant.

1.2 Need for Increased Storage Capacity

Farley Unit 2 is a pressurized water reactor with a licensed power of 2652 MWt. The reactor core contains 157 fuel assemblies.

The modifications evaluated in this EIA are the proposals by the licensee to increase the SFP storage capacity from 675 to 1407 spaces.

The proposed increase would be accomplished by replacing the existing fuel storage racks with new, more compact, neutron absorbing racks. The proposed rack design uses a nominal 10.75-inch center-to-center spacing in each direction. The old racks had nominal 13-inch center-to-center spacing in each direction. This modification would extend spent fuel storage capability in the SFP to the year 2009 compared to the year 1994 with the current design. The increase in capacity would extend the capability for a full core discharge from 1991 to 2006. This added capability, while it is not needed to protect the health and safety of the public, is desirable in the event of a need for a reactor vessel inspection or repair. Such off-load capability would reduce occupational exposures to plant personnel.

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. The Nuclear Fuel Services (NFS) plant at West Valley, New York, was shut down in 1972 for alterations and expansion; on September 22, 1976, NFS informed the Commission that they were withdrawing from the nuclear fuel reprocessing business. The Allied General Nuclear Services (AGNS) proposed plant in Barnwell, South Carolina, is not licensed to operate.

The General Electric Company's (GE) Morris Operation (MO) in Morris, Illinois, is in a decommissioned condition. Although no plants are licensed for reprocessing fuel, the storage pool at Morris, Illinois, and the storage pool at West Valley, New York (on land owned by the State of New York and leased to NFS through 1980), are licensed to store spent fuel. The storage pool at West Valley is not full but NFS is presently not accepting any additional spent fuel for storage, even from those power generating facilities that had contractural arrangements with NFS. GE is accepting additional spent fuel for storage

at the MO only from a limited number of utilities. Construction of the AGNS receiving and storage station has been completed. AGNS has applied for, but has not been granted, a license to receive and store irradiated fuel assemblies in the storage pool at Barnwell prior to a decision on the licensing action relating to the separation of facility. The future of this facility is uncertain.

1.3 Radioactive Wastes

The station contains waste treatment systems designed to collect and process the gaseous, liquid and solid waste that might contain radioactive material. The waste treatment systems are evaluated in the Farley Unit 2 FES dated December 1974. There will be no change in the waste treatment systems described in Section 3.2.3 of the FES because of the proposed modification.

1.4 Spent Fuel Pool Cleanup System

The SFP Cooling and Cleanup System consists of two cooling trains, a purification loop, a surface skimmer loop, and piping, valves and instrumentation. The pumps draw water from the pool. This flow is passed through the heat exchangers and then returned to the pool. While the heat removal operation is in process, a portion of the SFP water is normally diverted through a demineralizer and a filter to maintain SFP water clarity and purity.

We find that the proposed expansion of the SFP will not appreciably affect the capability and capacity of the existing SFP cleanup system. More frequent replacements of filters or demineralizer resin, required when the differential pressure exceeds a predetermined limit or demineralization effectiveness is reduced, can offset any potential increase in radioactivity and impurities in the pool water as a result of the expansion of stored spent fuel. Thus, we have determined that the existing fuel pool cleanup system with the proposed high density fuel storage (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool and thus meets the requirements of General Design Criterion 61 in Appendix A of 10 CFR Part 50 as it relates to appropriate systems to fuel storage; (2) is capable of reducing occupational exposures to radiation by removing radioactive products from the pool water, and thus meets the requirements of Section 20.1(c) of 10 CFR Part 20 as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water into the filters and demineralizers, and thus meets Regulatory Position C.2.f(c) of Regulatory Guide 8.8, as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters, and thus meets Regulatory Position C.2.f(3) of Regulatory Guide 8.8, as it relates to removing crud from fluids through physical action.

On the basis of the above evaluation, we conclude that the existing SFP cleanup system meets GDC 61, Section 20.1(c) of 10 CFR Part 20 and the appropriate sections of Regulatory Guide 8.8. Therefore, the system is acceptable for the proposed high density fuel storage.

2.0 ENVIRONMENTAL IMPACTS OF THE PROPOSED ACTION

2.1 Non-radiological

For the Farley Unit 2 SFP expansion, the new racks will be fabricated offsite, and transported to the facility by truck. Because of this, no unusual terrestrial effects are anticipated or considered likely. An estimate of the maximum increase in the rate of heat addition to the cooling water system is approximately $6.3X10^6$ BTU/hr. This additional thermal output from the expanded fuel pool is the value which would occur at 100 hours after shutdown with all storage cells filled. The rate would decrease exponentially with time after placement in the pool. The enlarged SFP heat rate is less than 1% of the total heat load of $6.5X10^9$ BTU/hr rejected by the station to the atmosphere by the cooling towers and to the receiving water as blowdown. No increase in service water usage is proposed. Thermal effects in the receiving water body will not be measurable by this small increase in the heat output rate. The licensee does not propose any change in chemical usage or any change to the NPDES discharge permit.

We conclude that the SFP expansion will not result in non-radiological environmential effects significantly greater or different from those already reviewed and analyzed in the FES for Units 1 and 2.

2.2 Radiological

2.2.1 Introduction

The potential offsite radiological environmental impacts associated with the expansion of the spent fuel storage capacity was evaluated and determined to be environmentally insignificant as addressed below.

During the storage of the spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides that might be released to the water through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are also predominantly nonvolatile. The primary impact of such nonvolatile radioactive nuclides is their contribution of radiation levels to which workers in and near the SFP would be exposed. The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (xenon and krypton), tritium and the iodine isotopes.

Experience indicates that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP.

During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding at the reactor operating condition of approximately 800°F. A few weeks after refueling, the spent fuel cools in the SFP so that the fuel clad temperature is relatively cool, approximately 180°F. This substantial temperature reduction should reduce the rate of release of fission products from the fuel pellets and decrease the gas pressure in the gap between pellets and clad, thereby tending to retain the fission products within the gap. In addition, most of the gaseous fission products have short half-lives and decay to insignificant levels within a few months. Based on the operational reports submitted by the licensees or discussions with the operators, there has not been any significant leakage of fission products from spent light water reactor fuel stored in the MO (formerly Midwest Recovery Plant) at Morris, Illinois, or at the NFS storage pool at West Valley, New York. Spent fuel which had significant leakage while in operating reactors has been stored in these two pools. After storage in the onsite SFP, this fuel was later shipped to either MO or NFS for extended storage. Although the fuel exhibited significant leakage at reactor operating conditions, there was no significant leakage from this fuel in the offsite storage facility.

2.2.2 Radioactive Material Released to Atmosphere

With respect to gaseous releases, the only significant noble gas isotope attributable to storing additional assemblies for a longer period of time would be Krypton-85. As discussed previously, experience has demonstrated that after spent fuel has decayed 4 to 6 months, there is no significant release of fission products from defective fuel. However, we have conservatively estimated that an additional 220 curies per year of Krypton-85 may be released when the Farley-2 modified pool is completely filled. This increase would result in an additional total body dose to an individual at the site boundary of less than 0.003 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural backgound radiation. The additional total body dose to the estimated population within a 50-mile radius of the plant is less that 0.004 man-rem/year. This is less than the natural fluctuations in the dose this population would receive from natural background radiation. Under our conservative assumptions, these exposures represent an increase of approximately 1% of the exposures from the station evaluated in the FES for the individual at the site boundary and the population. Thus, we conclude that the proposed modification will not have any significant nor measurable impact on exposures offsite.

Assuming that the spent fuel will be stored onsite for several years, Iodine-131 releases from spent fuel assemblies to the SFP water will not be significantly increased because of the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between refueling.

Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above 150°F during normal refuelings as used in the design analysis. Therefore, it is not expected that there will be any significant change in the

annual release of tritium or iodine as a result of the proposed modification from that previously evaluated in the FES. Most airborne releases from the station result from leakage of reactor coolant which contains tritium and iodine in higher concentrations than the SFP. Therefore, even if there were a higher evaporation rate from the SFP, the increase in tritium and iodine released from the station as a result of the increase in stored spent fuel would be small compared to the amount normally released from the station and that which was previously evaluated in the FES. If it is desired to reduce levels of radioiodine, the air can be diverted to charcoal filters for the removal of radioiodine before release to the environment.

2.2.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool is controlled by the SFP Cleanup System filter and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations while reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filter and demineralizer. The increase of radioactivity, if any, should be minor because of the capability of the SFP Cleanup System to remove radioactivity to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the SFP Cleanup System due to the proposed modification. While we generally agree with the licensee's conclusion, as a conservative estimate we have assumed that the amount of solid radwaste may be increased by an additional six resin beds (180 cubic feet) a year due to the increased operation of the SFP Cleanup System. The annual average volume of solid waste shipped from J.M. Farley, Unit 1 during 1978 through 1981 was 21,400 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP Cleanup Systems by about 180 cubic feet of dewatered spent resin per year, the increase in total waste volume shipped would be less than 1% and would have no significant additional environmental impact.

We have reviewed the licensee's plan for the removal and disposal of the low density racks and the installation of the high density racks. Since the SFP for Farley Unit 2 has never had spent fuel stored in it and is currently dry, clean and uncontaminated, there will be no additional radwaste generated by the removal of the low density racks.

2.2.4 Radioactivity Released to Receiving Waters

There should not be a significant increase in the liquid release of radionuclides from the plant as a result of the proposed modification. Since the SFP cooling and cleanup system operates as a closed system, only water originating from cleanup of SFP floors and resin sluice water need be considered as potential sources of radioactivity.

It is expected that neither the quantity nor activity of the floor cleanup water will change as a result of this modification. The SFP demineralizer resin removes soluble radioactive matter from the SFP water. These resins are

periodically flushed with water to the spent resin storage tank. The amount of radioactivity on the SFP demineralizer resin might increase slightly due to the additional spent fuel in the pool, but the soluble radioactivity should be retained on the resins. If any activity is transferred from the spent resin to the flush water, it will be removed by the Liquid Waste Processing System since the sluice water is returned to that system for processing. After processing in the system, the amount of radioactivity released to the environment as a result of the proposed modification would be negligible.

2.2.5 Impacts of Other Pool Modifications

As discussed above, the additional radiological environmental impact in the vicinity of Farley Unit 2 resulting from the proposed modifications are very small fractions (approximately 1%) of the impacts evaluated in the Farley Unit 2 FES. These additional impacts are too small to be considered anything but local in character.

Based on the above, we conclude that a SFP modification at any other facility should not significantly contribute to the environmental impact at Farley Unit 2 and that the Farley Unit 2 SFP modification should not contribute significantly to the environmental impact of any other facility.

2.3 Summary

On the basis of this review we conclude that the environmental impacts associated with modification and operation of the expanded spent fuel pool will have negligible adverse effects.

3.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

3.1 Cask Drop Accidents

The licensee states in the December 18, 1981 submittal that "Protection against a cask drop is assured by the Seismic Category I, single failure-proof lifting device, and by the interlocks and administrative controls described in the Farley FSAR subsection 9.1.4." The staff has concluded that the spent fuel cask crane design, inservice inspection program, and proof test program are at least equal to the staff's requirements in NUREG-0554, May 1979, "Single Failure-Proof Cranes for Nuclear Power Plants." The staff concludes, therefore, that, with respect to a cask drop accident, the likelihood of such an occurrence is sufficiently small that the proposed SFP modification is acceptable, and no additional restrictions on load handling operations in the vicinity of the SFP are necessary.

3.2 Fuel Handling Accidents

The new high-density racks will be installed prior to the first refueling outage; the spent fuel pool is not dry and contains no spent fuel. Even if there were spent fuel in the pool, the maximum weight of loads which may be

transported over spent fuel in the pool would be limited to less than 3000 pounds by Technical Specification 3.9.7.1. The proposed SFP modification does not, therefore, increase radiological consequences of fuel handling accidents considered in the staff Safety Evaluation of May 2, 1975, since this accident would still result in, at most, release of the gap activity of one fuel assembly due to the limitations on available impact kinetic energy.

3.3 Conclusions

Based upon the above evaluation, the staff concludes that the likelihood of a cask drop accident resulting in radionuclide releases is sufficiently small that this accident need not be considered. Additionally, a fuel handling accident would not be expected to result in radionuclide releases leading to offsite radiological consequences exceeding those of the fuel handling accident in the staff Safety Evaluation of May 2, 1975 (9 Rem to the thyroid and 3 Rem whole body at the Exclusion Area Boundary); these conservatively estimated doses are less than a small fraction of 10 CFR Part 100 guideline values and are acceptable.

4.0 SUMMARY

The Final Generic Environmental Impact Statement (FGEIS) on Handling and Storage of Spent Light Water Power Reactor Fuel concluded that the environmental impact of interim storage of spent fuel was negligible and the cost of the various alternatives reflect the advantage of continued generation of nuclear power with the accompanying spent fuel storage. Because of the differences in SFP designs the FGEIS recommended licensing SFP expansions on a case-by-case basis. For Farley Unit 2, expansion of the storage capacity of the SFP does not significantly change the radiological impact evaluated in the FES. As discussed in Section 2.2.2 above, the additional total body dose that might be received by an individual or the estimated population within a 50-mile radius is less than 0.003 mrem/yr and 0.009 man-rem/yr, respectively, and is less than the natural fluctuations in the dose this population would receive from background radiation. Operation of the station with additional spent fuel in the SFP is not expected to increase the occupational radiation exposure by more than one percent of the total annual occupational exposure at the station.

5.0 BASIS AND CONCLUSION FOR NOT PREPARING AN ENVIRONMENTAL IMPACT STATEMENT

We have reviewed the proposed modifications relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6. We have determined, based on this assessment, that the proposed license amendments will not significantly affect the quality of the human environment. Therefore, the Commission has determined that an environmental impact statement need not be prepared and that, pursuant to 10 CFR 51.5(c), the issuance of a negative declaration to this effect is appropriate.

Dated: June 23, 1982

NRC Participants: Dr. T. Cain, M. Fecteau, M. Lamastra, C. Miller, E. Reeves