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April 15, 1983

Docket No. 50-348

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. 31 to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant, Unit No. 1. The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated March 19, 1982, supplemented by letters dated April 21 and September 14, 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 and Negative Declaration are also enclosed.

Sincerely,

/S/

Edward A. Reeves, Project Manager
Operating Reactors Branch No. 1
Division of Licensing

Enclosures:
1. Amendment No. 31 to NPF-2
2. Safety Evaluation
3. Environmental Impact Appraisal
4. Notice of Issuance/Negative Declaration

cc w/enclosures:
See next page

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F.R. NOTICE
AMENDMENT

OFFICE	ORB 1	ORB 1	ORB 1	AD:OR	OELD		
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DATE	3/28/83	3/28/83	3/28/83	3/28/83	3/29/83		

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

ALABAMA POWER COMPANY

DOCKET NO. 50-348

JOSEPH M. FARLEY NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 31
License No. NPF-2

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

   A. The application for amendment by Alabama Power Company (the licensee) dated March 19, 1982, supplemented by letters dated April 21 and September 14, 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-2 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 31 , 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. However, Technical Specification 5.6, both the existing page 5-7 and revised page 5-7, will be effective until completion of the spent fuel pool modifications. When modifications are complete the old Technical Specification 5.6 is cancelled.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing

Attachment:
Changes to the Technical
  Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 31 TO FACILITY OPERATING LICENSE NO. NPF-2

DOCKET NO. 50-348

Revise Appendix A as follows:

Remove Page*	Insert Page*
5-7	5-7

NOTE: Old page 5-7 and new page 5-7 are both effective until completion of the spent fuel pool modifications. At that time remove old page 5-7 which is cancelled.

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 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 1

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-2

ALABAMA POWER COMPANY

JOSEPH M. FARLEY NUCLEAR PLANT - UNIT 1

DOCKET NO. 50-348

CONTENTS

1.0 Introduction

By letter dated March 19, 1982, as supplemented April 21 and September 14, 1982, Alabama Power Company (APCo) (the licensee) requested an amendment to Facility Operating License No. NPF-2 for Joseph M. Farley Nuclear Plant Unit No. 1. The request would revise the 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 1 to operate until the year 2004 with capability for a full core discharge, assuming annual one-third core reloads of 52 assemblies each reload. Total storage capacity would be expended with the discharge in the year 2007, when 29 assemblies would remain in the reactor vessel.

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 contains 136 spent fuel assemblies discharged from cycles 1, 2, and 3. Only 32 assemblies were discharged at the end of cycle 3 since cycle 3 was shortened by several months due to a failure in the main electrical generator which occurred on September 10, 1981. APCo's current plans are to off-load the cycle 4 spent fuel assemblies during the cycle 5 refueling outage which started January 15, 1983. Later these spent fuel assemblies will be located to one end of the fuel pool. Thus, removal of the old used type fuel racks will be accomplished at the opposite end of the pool to minimize radiation exposure to divers. The modification to the spent fuel pool is scheduled to start in July 1983 and to complete about end of September 1983 on a phased basis. We will modify the effective date of the license amendment accordingly.

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). 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.

The racks are in compliance with the applicable portions of the following: Regulatory Guides 1.13 and 1.29; and 10 CFR Part 50 Appendix A General Design Criteria 1, 2, 61, 62, and 63.

Based on the above we conclude that the proposed storage rack design and arrangement is adequate and, therefore, it is acceptable.

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) fully flooded with unborated water 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 ± 0.0044 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 uncertainty of 0.0159 (obtained by a statistical combination of the calculational and mechanical uncertainties). The mechanical uncertainty 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 ± 0.0044. This meets our acceptance criteria for criticality calculations of 0.95 when flooded with unborated water 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 at 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.

3.1.1 Conclusion

We conclude that the proposed storage racks meet the requirements of General Design Criterion 62 as 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 modifications to the 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 revised 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 1 Spent Fuel Pool Modification Report dated March 1982. 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 only be stored 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 the Farley Unit 1 plant. 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 Pool Cooling System

3.2.1 Introduction

The SFP 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 are located in such a manner that protects against inadvertent drainage of the pool to less than 4 feet below the normal level of 23 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 refueling cycles for Farley Unit 1 are twelve month cycles where one-third of the core is removed and stored in the SFP after each cycle. To limit the decay heat load, the one-third core (52 assemblies) will be removed from the reactor vessel and stored in the SFP no sooner than 100 hours after reactor shutdown. In the event of a full-core discharge, the decay heat load is based on a ten 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-ninth refueling discharge, was calculated to be 19.755×10^6 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 30.384×10^6 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. For this event, a complete failure of the SFP cooling system, even for the maximum abnormal heat load, at least four hours is available before boiling occurs. The maximum boiloff rate is between 50 and 60 gpm. Each of three 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, the respective makeup rates, and the time required before makeup is needed have 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

The SFP, a seismic Category I reinforced concrete structure, is housed within the fuel storage area of the auxiliary building. A 125-ton capacity outdoor overhead unequal leg gantry crane is provided to handle heavy loads such as the spent fuel shipping cask. The crane, a seismic Category I, single-failure proof crane, was evaluated and found acceptable by the staff as documented in Supplement No. 2 to the Farley Units 1 and 2 Safety Evaluation Report, NUREG-0117, dated October 1976. The range of travel of this crane is limited by design such that it cannot pass over the SFP. The only crane that can pass over the SFP is the spent fuel bridge crane with a main hoist rated at 4000 pounds. Therefore, the removal and installation of storage racks will require a temporary traveling bridge and hoist installed on the fuel handling bridge rails for the movement of the storage racks to and from the spent fuel cask area. There the racks are handled by the single-failure-proof cask handling crane. This is the same seismic Category I crane and lifting-fixture that was used for reracking Farley Unit 2, thereby demonstrating its ability to safely perform the reracking. Following the installation of the temporary crane and before use in Unit 1, the crane will be tested using a load of 117 percent of rated capacity.

APCo indicated that the movement of all loads into and out of the auxiliary building, associated with this modification, will be accomplished with the single-failure-proof cask crane and double rigging to assure that a single failure will not result in an unanalyzed load-drop event. No heavy loads will be carried over any spent fuel assemblies during the rerack program. The spent fuel assemblies will be moved and located so that no heavy loads will be carried over them.

APCo also stated in response to NUREG-0512, "Control of heavy loads at Nuclear Power Plants," that all crane operators and signalmen will be trained in accordance with ANSI B30.2-1976, and no exceptions are taken regarding training, qualification or operator conduct.

3.3.1 Conclusion

We have reviewed the described load handling operations and equipment needed for the spent fuel rack modifications and conclude that the stored spent fuel and safety related equipment will be adequately protected against a load drop accident. We, therefore, conclude that the health and safety of the public will not be endangered by the expansion program of the Farley Unit 1 SFP. Therefore, the expansion program is acceptable.

3.4 Structural and Seismic Loadings

3.4.1 Introduction

The Farley Unit 1 SFP is an existing reinforced concrete box structure. The walls of the pool vary 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 7.2 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.

3.4.2 Applicable Codes Standard and Specifications

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 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.

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 also 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.

3.4.3 Seismic and Impact Loads

The SFP floor response spectra used for the seismic analysis were as provided in the Farley 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-fuel 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 as described above in Section 3.3.

Technical Specification 3.9.7.1 prohibits transporting loads greater than 3000 pounds over the SFP; 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.

3.4.4 Load and Load Combinations

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

3.4.5 Design and Analysis Procedures

As described above, dynamic analyses of the rack and pool were conducted using lumped 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.

3.4.6 Structural Acceptance Criteria

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

3.4.7 Materials, Quality Control, and Special Construction Techniques

With the exception noted previously in Section 3.4.2, all materials are in accordance with the ASME Code, as are fabrication, and inspection procedures.

3.4.8 Conclusion

We find that the subject modification with respect to structural and seismic loadings, proposed by APCo 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 as discussed in Section 3.4.2 above.

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 SFP 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 (Section 3.7) 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 SFP is fabricated of materials that will have good compatibility with the borated water chemistry of the 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]

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

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 do 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, APCo 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 the Unit 1 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 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 1 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

3.6.1 Radiation Exposure to Workers During Modifications

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. The occupational exposure for this operation is estimated by APCo to range from 6.1 to 8.1 person-rems. This estimate is based on a detailed breakdown of occupational exposure for each phase of the modification. APCo has considered the number of individuals performing a specific job, their occupancy time while performing this job, and the average dose rate in the area where the job is being performed.

Throughout the SFP modification operation, personnel exposure controls will be administered in accordance with APCo's radiological control procedures to assure exposures as low as is reasonable achievable (ALARA) to workers. These procedures include pre-job planning and worker briefings, checking water clarity for visibility, extensive surveys of the work area, physical barriers to prevent divers entering prohibited areas, and use of local filtered ventilation when necessary. In addition, APCo has developed specific operating procedures for divers to assure that their doses are ALARA.

APCo has presented two alternative plans for the removal and disposal of the old racks. These are (1) transfer of the old racks to another utility for use as spent fuel racks or (2) decontamination of the old racks prior to disposal. APCo will follow ALARA guidelines for workers regardless of which disposal method chosen.

3.6.2 Conclusion

Based on the manner in which APCo proposes to perform the modifications, and relevant experience from other operating reactors that have performed similar SFP modifications, we conclude that the SFP modifications can be performed in a manner that will ensure as low as is reasonably achievable (ALARA) exposures to workers.

3.6.3 Onsite Radiation Exposure During Normal Operations

We have estimated the increment in onsite occupational dose during normal operations after the SFP modifications which will result in an increase in stored spent fuel assemblies at Unit 1. This estimate is based on information supplied by APCo for occupancy times and for dose rates in the SFP area from radionuclide concentrations in the SFP water. The spent fuel assemblies contribute a neglible 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 exposure at Unit 1. The small increase in radiation exposure should not affect the licensee's ability to maintain individual occupational doses to ALARA levels and within the limits of 10 CFR Part 20.

3.6.4 Conclusion

Thus, based on these considerations, we conclude that storing the additional fuel in the modified SFP during normal operations of Farley Unit 1 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 psid. APCo 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, APCo 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, APCo established 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 SFP 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 Overall Safety Conclusion

On the basis of the foregoing analysis, we conclude 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.

Date: APR 5 1983

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ENCLOSURE 3

ENVIRONMENTAL IMPACT APPRAISAL BY THE
OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE
SPENT FUEL STORAGE POOL

FACILITY OPERATING LICENSE NO. NPF-2
ALABAMA POWER COMPANY
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1

DOCKET NO. 50-348

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 1 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 March 19, 1982, as supported by letters dated April 21 and September 14, 1982, Alabama Power Company (APCo) (the licensee) requested an amendment to Facility Operating License No. NPF-2 for the Joseph M. Farley Nuclear Plant Unit 1 (Farley Unit 1). The proposed amendment would allow an increase in the storage capacity of the Farley Unit 1 from 675 to 1407 storage locations.

The environmental impacts of Farley Unit 1 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 1 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 2007 compared to the year 1993 with the current design. The increase in capacity would extend the capability for a full core discharge from 1990 to 2004. 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 1 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

We have reviewed potential non-radiological environmental impacts associated with the amendment proposed by APCo. Increased storage capacity would be achieved by re-racking the fuel in the Unit 1 spent fuel pool to increase the storage capacity from 675 to 1407 cells. Our principal objective of a review of this type is to determine whether it will result in environmental effects significantly greater or different than those already reviewed and analyzed in the FES for the facility. Additionally, the need for action to mitigate environmental impacts may be identified.

The following three potential impacts are reviewed for SFP modifications. Criteria for the evaluation are indicated for each.

1) Construction impact: Does the manufacture or transportation of rack material create measurable effects to areas not previously disturbed during site preparation and plant construction?

2) Impact due to discharge of decay heat: Does the increase in heat output from the SFP require an increase in cooling water usage (flow)? Does additional SFP heat output cause additional measurable thermal effects to the receiving waters, or an increase in cooling tower fogging and icing or makeup water withdrawal?

3) **Impact due to other chemical discharges: Does additional storage result in an increase in non-radiological chemical waste discharges to the receiving water? Is a change to the facility NPDES permit necessary?**

For the Unit 1 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. APCo does not propose any change in chemical usage or change to the NPDES discharge permit.

We conclude that the SFP expansion will not result in non-radiological environmental 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.

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 1 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 than 0.1 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.

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 Occupational Exposure

We have reviewed APCo's plan for the removal and disposal of the low density racks and the high density racks with respect to occupational radiation exposure. The occupational exposure for the entire operation is estimated by the licensee to be between 6.1 to 8.1 person-rem. We consider this to be a conservative estimate because this is based on conservative dose rates and occupancy factors for individuals performing a specific job during the modification. This operation is expected to be a small fraction of the total annual person-rems from occupational exposure.

We have estimated the increment in onsite occupational doses resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by APCo and by utilizing relevant assumptions for occupancy times and for dose rates in the spent fuel pool area from radionuclide concentrations in the SFP water. The spent fuel assemblies contribute a negligible amount to the dose rates in the pool area because of the depth of water shielding the fuel. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modifications should add less than one percent to the total annual occupational radiation exposure and conclude that storing additional spent fuel in the pool will not result in any significant increase in doses received by workers.

2.2.6 Impacts of Other Pool Modifications

As discussed above, the additional radiological environmental impact in the vicinity of Farley Unit 1 resulting from the proposed modifications are very small fractions (approximately 1%) of the impacts evaluated in the Farley 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 1 and that the Farley Unit 1 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 SFP will have negligible adverse effects.

3.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

The review was conducted according to the guidance of Standard Review Plan 15.7.4, NUREG-0612, and NUREG-0554 with respect to accident assumptions.

3.1 Cask Drop Accidents

The licensee states in the March 19, 1982 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," Therefore, the staff concludes with respect to a cask drop accident, that 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 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. Therefore, the proposed SFP modification does not 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 the fuel assembly due to the limitations on available impact kinetic energy.

3.3 Conclusion

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 1, 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 0.1 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 and storing additional spent fuel in the pool will not result in any significant increase in doses received by workers.

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 amendment will not significantly affect the quality of 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: April 15, 1983

NRC Participants:
Dr. T. Cain
M. Wohl
M. Lamastra
P. Stoddart
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-348

ALABAMA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND NEGATIVE DECLARATION

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 31 to Facility Operating License No. NPF-2, issued to Alabama Power Company (the licensee), which revised Technical Specifications for operation of the Joseph M. Farley Nuclear Plant, Unit No. 1, (the facility) located in Houston County, Alabama. The amendment is effective as of the date of issuance.

The amendment revises the Technical Specification to enlarge the capacity of the spent fuel pool from 675 fuel assemblies to 1407 assemblies.

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. Notice of Proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 4, 1982 (47FR5371). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has prepared an environmental impact appraisal for the revised Technical Specifications and has concluded that an environmental impact statement for this particular action is not warranted because the proposed action will not significantly affect the quality of the human environment.

For further details with respect to this action, see (1) the application for amendment dated March 19, 1982, as modified by letters dated April 21, and September 14, 1982, (2) Amendment No. 31 to License No. NPF-2, (3) the Commission's related Safety Evaluation and (4) the Commission's Environmental Impact Appraisal. 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 Houston Memorial Library, 212 W. Burdeshaw Street, Dothan, Alabama 36303. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 15th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch No. 1
Division of Licensing