DISTRIBUTION:
Docket
NRC PDR
Local PDR
ORB-2 Reading
VStello
KRGoller/TJCarter
DLZiemann
RMDiggs
RPSnaider
Attorney, OELD
OI&E (5)
BJones (4)
BScharf (15)
WPasciak
BGrimes
WButler
LShao
DEisenhut
DRoss
JRBuchanan
TBAbernathy
ACRS (16)
CMiles, OPA

BHanlers

Docket No. 50-313

DEC 17 1976

Arkansas Power & Light Company
ATTN: Mr. J. D. Phillips
Senior Vice President
Production, Transmission, and Engineering
Sixth and Pine Streets
Pine Bluff, Arkansas 71601

Gentlemen:

The Commission has issued the enclosed Amendment No. 17 to Facility Operating License DPR-51 for Arkansas Nuclear One - Unit 1. This amendment consists of a change to the license and revises the provisions of the Technical Specifications in response to your request of October 7, 1976, supplemented by letters dated October 18, October 25, November 11, November 16 and November 19, 1976.

This amendment authorizes changes in the design of the spent fuel pool from that reviewed and approved in the operating license review and as described in the Arkansas Nuclear One - Unit 1 Final Safety Analysis Report. The changes will increase spent fuel storage capacity from 253 to 590 fuel assemblies. During our review, we discussed with your staff various modifications to the technical specification changes proposed in your October 7, 1976 submittal. Your staff has agreed to these modifications and they have been incorporated.

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

Notified licensee 12/17/76

Sincerely,

Original Signed by:
Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to License DPR-51
2. Joint Safety Evaluation and Environmental Impact Appraisal
3. Notice

Concurrence provided noted correction made and concurrence by Shao

	Eisenhut	DOR:PSB	DOR:EEB
		WButler	BGrimes
	12/16/76	12/16/76	12/16/76

OFFICE	ORB-2	DOR:ORB-2	DOR:ORB-2	DOR:EEB	OELD	DOR:AD/ORs
SURNAME	RMDiggs	RPSnaider:esp	DLZiemann	LShao		KRGoller
DATE	12/17/76	12/14/76	12/16/76	12/16/76	12/15/76	12/16/76

Form AEC-318 (Rev. 9-53) AECM 0240

U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

DEC 17 1976

cc w/enclosures:
Horace Jewell, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Phillip K. Lyon, Esquire
House, Holms & Jewell
1550 Tower Building
Little Rock, Arkansas 72201

Mr. Donald Rueter
Manager, Licensing
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72201

Arkansas Polytechnic College
Russellville, Arkansas 72801

Chief, Energy Systems Analyses Branch (AW-459)
Office of Radiation Programs
U. S. Environmental Protection Agency
Room 645, East Tower
401 M Street, S. W.
Washington, D. C. 20460

U. S. Environmental Protection Agency
Region VI Office
ATTN: EIS COORDINATOR
1201 Elm Street
First International Building
Dallas, Texas 75270

Honorable Ermil Grant
Acting County Judge of Pope County
Pope County Courthouse
Russellville, Arkansas 72801

cc w/enclosures and cy of AP&L filings dtd. 10/7/76, 10/18, 10/25 and 11/11, 16 & 19/76:
Director, Bureau of Environmental Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-51

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

   A. The application for amendment by Arkansas Power & Light Company (the licensee) dated October 7, 1976, as supplemented by letters dated October 18, November 11, November 16, and November 19, 1976, 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.b(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   b(2) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess, and use at any time any byproduct, source and special nuclear materials as reactor fuel, sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors, in accordance with the limitations for storage and amounts required for reactor operation as described in the Final Safety Analysis Report, as amended and in the application for license amendment dated October 7, 1976, as supplemented by letters dated October 18, November 11, November 16 and November 19, 1976.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by:
Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: DEC 17 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 17

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace existing pages 59, 59a, and 116 of the Appendix A portion of the Technical Specifications with the attached revised pages bearing the same numbers. The changed areas on the revised pages are identified by a marginal line.

3.8.10 The reactor building purge isolation system, including the radiation monitors shall be tested and verified to be operable within 7 days prior to refueling operations.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours.

3.8.12 All fuel handling in the Auxiliary Building shall cease upon notification of the issuance of a tornado watch for Pope, Yell, Johnson, or Logan counties in Arkansas. Fuel handling operations in progress will be completed to the extent necessary to place the fuel handling bridge and crane in their normal parked and locked position.

3.8.13 No loaded spent fuel shipping cask shall be carried above or into the Auxiliary Building equipment shaft unless atmospheric dispersion conditions are equal to or better than those produced by Pasquill type D stability accompanied by a wind velocity of 2 m/sec. In addition, the railroad spur door of the Turbine Building shall be closed and the fuel handling area ventilation system shall be in operation.

3.8.14 For the maximum fuel pool heat load capacity (i.e., seven reload batches (413 assemblies) stored in the pool at the time of discharge of the full core) the full core to be discharged shall be cooled in the reactor vessel a minimum of 175 hours prior to discharge.

3.8.15 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool.

3.8.16 The spent fuel shipping cask shall not be carried by the auxiliary building crane pending the evaluation of the spent fuel cask drop accident and the crane design by AP&L and NRC review and approval.

Bases

Detailed written procedures will be available for use by refueling personnel. These procedures, the above specifications, and the design of the fuel handling equipment as described in Section 9.7 of the FSAR incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. If no change is being made in core geometry, one flux monitor is sufficient. This permits maintenance on the instrumentation. Continuous monitoring of radiation levels and neutron flux provides immediate indication of an unsafe condition. The decay heat removal pump is used to maintain a uniform boron concentration.(1) The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) The boron concentration will be maintained above 1800 ppm. Although this concentration is sufficient to maintain the core $k_{eff} \leq 0.99$ if all the control rods were removed from the core, only a few control rods will be removed at any one time during fuel shuffling and replacement. The k_{eff} with all rods in the core and with refueling boron concentration is approximately 0.9. Specification 3.8.5 allows the control room operator to inform the reactor building personnel of any impending unsafe condition detected from the main control board indicators during fuel movement.

The specification requiring testing reactor building purge termination is to verify that these components will function as required should a fuel handling accident occur which resulted in the release of significant fission products.

Because of physical dimensions of the fuel bridges, it is physically impossible for fuel assemblies to be within 10 feet of each other while being handled.

Specification 3.8.11 is required as the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.[3]

Specification 3.8.14, which requires cooling of the full core for 175 hours prior to discharge to the spent fuel pool when seven reload batches are already stored in the pool, is necessary to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded.

Specification 3.8.15 will assure that damage to fuel in the spent fuel pool will not be caused by dropping heavy objects onto the fuel. Administrative controls will prohibit the storage of fuel in locations adjoining the walls at the north and south ends of the pool, in the vicinity of cask storage area and fuel tilt pool access gates, until the review specified in 3.8.16 is completed.

Specification 3.8.16 assures that the spent fuel cask drop accident cannot occur prior to completion of the NRC staff's review of this potential accident and the completion of any modifications that may be necessary to preclude the accident or mitigate the consequences. Upon satisfactory completion of the NRC's review, Specification 3.8.16 shall be deleted.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3

5.4 NEW AND SPENT FUEL STORAGE FACILITIES

Applicability

Applies to storage facilities for new and spent fuel assemblies.

Objective

To assure that both new and spent fuel assemblies will be stored in such a manner that an inadvertent criticality could not occur.

Specification

5.4.1 New Fuel Storage

1. New fuel will normally be stored in the new fuel storage pool. The fuel assemblies are stored in racks in parallel rows, having a nominal center to center distance of 21 inches in both directions. This spacing is sufficient to maintain a K_{eff} of less than .9 even if flooded with unborated water, based on fuel with an enrichment of 3.5 weight percent U235.

2. New fuel may also be stored in the spent fuel pool or in their shipping containers.

5.4.2 Spent Fuel Storage

1. Irradiated fuel assemblies will be stored, prior to offsite shipment, in the stainless steel lined spent fuel pool, which is located in the auxiliary building. The pool is sized to accommodate a full core of irradiated fuel assemblies in addition to the concurrent storage of two and one-third cores of spent fuel previously discharged.

2. The spent fuel pool is filled with borated water with a minimum concentration of 1800 ppm boron during refueling.

3. One spent fuel storage rack position is designed to accommodate a special container for storage of a leaking fuel assembly.

4. The spent fuel pool and new fuel pool racks are designed as seismic Class 1 equipment.

5. The design is based upon storage of spent fuel containing no more than 45.2 grams of Uranium-235 per longitudinal centimeter of assembly.

REFERENCES

FSAR, Section 9.6

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

SAFETY EVALUATION AND ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

INTRODUCTION

By letter dated October 7, 1976, the Arkansas Power & Light Company (AP&L) submitted an application for a license amendment to increase the storage capacity of the Arkansas Nuclear One - Unit 1 (ANO-1) spent fuel pool (SFP). This application was supplemented by additional information provided by letters dated October 18, October 25, November 11, November 16 and November 19, 1976.

DISCUSSION

The present storage capacity of the ANO-1 SFP is 253 fuel assemblies and there are currently no irradiated fuel assemblies stored in the pool, although irradiated fuel was temporarily stored in the pool during previous investigations of reactor surveillance specimen hold-down tube failures. The proposed modification would replace the existing fuel storage racks with closer spaced racks, which would increase the storage capacity to 590 fuel assemblies. With the present storage capacity, by January 1978 the normal discharge of fuel would fill the existing racks to the extent that the reactor core could not be completely unloaded if required for maintenance or inspection. Furthermore, by January 1980, the existing racks would be completely filled and no storage space would be available to accommodate the normal fuel discharge for the 1981 refueling. The proposed modification would increase the spent fuel storage capacity to accommodate refueling plus a full core unloading until 1984.

The proposed modification does not alter the external physical geometry of the spent fuel pool or involve changes to the SFP cooling or purification systems. The proposed modification will not affect in any manner the quantity of uranium fuel utilized in the reactor over the anticipated operating life of the facility and thus in no way will affect the generation of spent uranium fuel by the facility. The rate and total quantity of spent fuel generated and stored in the SFP during the anticipated operating lifetime of the facility remains unchanged as a result of the proposed expansion. However, the modification will increase the number of spent fuel assemblies stored in the SFP at any given time and the storage time of some of the fuel assemblies will be increased.

Currently, spent fuel is not being reprocessed on a commercial basis in the United States. Although no plants are licensed for reprocessing fuel, two reprocessing facilities are licensed for storing spent fuel and applications have been filed for permission to expand these facilities. A third reprocessing facility has applied for a license to receive and store irradiated fuel assemblies prior to a decision on the licensing action relating to the separation facility. The NRC staff is preparing a generic environmental impact statement on spent fuel storage of light water power reactor fuel and is expected to complete this statement by the Fall of 1977.

I. SAFETY EVALUATION

Reactivity Considerations Discussion

The proposed high density fuel assembly storage racks are an Exxon Nuclear Company, Inc. design which uses square, type 304 stainless steel tubes to hold the fuel assemblies. The nominal wall thickness of all the stainless steel tubes is 0.115 inches. The thickness of stainless steel between all storage lattice positions is 0.230 inches (two tube walls). The rack structure is designed to hold these individual containers, which are 14 feet long with a square cross section of 9-1/16 inches, on a 13.5 inch pitch during safe shutdown earthquake accelerations. Thus, there will be about four inches of water between neighboring storage containers. The 13.5 inch pitch, combined with the overall dimension of the fuel assembly, which is 8.52 inches, gives a fuel region volume fraction of 0.40 for the storage lattice.

AP&L based its criticality analyses for this array on an enrichment of 3.5 weight percent U-235. Assuming a uranium dioxide density of 92.5 percent of the theoretical density in the 15x15 array of 208 fuel rods, this 3.5 percent enrichment results in a fuel loading of 45.2 grams of U-235 per axial (longitudinal) centimeter of fuel assembly.

For the neutron multiplication factor calculations, AP&L utilized CCELL, BRT-1, and GAMTEC-II computer programs to obtain 18 energy group cross sections for use in the KENO-II Monte Carlo program and 90 group cross sections for use in the XMC Monte Carlo program. These calculational methods were verified by calculating five critical experiments which were made up of varying compositions of stainless steel clad fuel elements in water. The thickness of the stainless steel clad used in the experiments was 0.016 inches. The results of these calculations, with the statistical uncertainty of the Monte Carlo calculation included, were all within 2.1 percent of the experimental value for the neutron multiplication factor.

These computer programs were first used to calculate the neutron multiplication factor for an infinite array of fuel assemblies in the nominal storage lattice. This gave a neutron multiplication factor of 0.913 $\pm$.004. The effects of the fabrication tolerances and fuel assembly positioning uncertainties were then calculated by assuming an infinite array of four bundle clusters with 13.125 inch center-to-center spacing. This configuration yielded a neutron multiplication factor of 0.925 $\pm$.004. The maximal effect of varying water temperatures was determined by calculating the worst possible case of 20°C water in the fuel storage containers with 100°C water between the storage containers. This also gave a neutron multiplication factor of 0.925 $\pm$ .004.

Evaluation of Reactivity Considerations

The major uncertainties in the calculations discussed above are in the accuracy of the multigroup cross sections. The critical experiments which were calculated had all of the materials which are in the storage lattice, but only a small amount of stainless steel (the .016 inch thick fuel element clad) was represented. Additionally, the geometry of the experiments was different from that of the actual storage lattice. However, the results of these calculations agree very well with results obtained from parametric calculations using other standard methods. Using AP&L's maximum calculated neutron multiplication factor of 0.929, and adding the total uncertainty of 0.021 which was obtained in calculating the five critical experiments, a maximum neutron multiplication factor of 0.95 is obtained. Since this value includes all uncertainties and is therefore conservative, the NRC staff finds it acceptable.

It has been determined that when any number of fuel assemblies, which are fabricated with uranium dioxide with a density between 90 and 96 percent of theoretical density and which have no more than 45.2 grams of uranium-235 per axial centimeter of assembly, are loaded into the proposed racks, the neutron multiplication factor will be $\leq$0.95. The ANO-1 Technical Specifications are being amended to prohibit the storage of fuel assemblies that contain more than 45.2 grams of uranium-235 per longitudinal centimeter of assembly, thereby assuring that the multiplication factor value of 0.95 will not be exceeded.

Spent Fuel Cooling Discussion

The ANO-1 spent fuel pool cooling system includes two 1,000 gallon per minute pumps and two heat exchangers. Approximately ten percent of the flow is bypassed around the heat exchangers and purified by filters and a demineralizer. The ANO-1 Final Safety Analysis Report (FSAR) states that each heat exchanger is designed to transfer 8.75×10^6 BTU/hr. (2.56 Mw) to 95°F service water when the spent fuel pool (SFP) temperature is 131°F.

In the submittal of October 7, 1976, AP&L utilized the calculational method of ANS Standard 5.1 to derive the pool heat loads for the "normal refueling" and "core discharge" cases. The results of these calculations show for normal refueling, i.e., ten annual discharges of 59 assemblies, assuming 150 hours cooling time for the last batch discharged, the heat generation rate will be 12 x 10^6 BTU/hr. (3.5 Mw) and the spent fuel pool cooling system will maintain the pool water temperature at approximately 120°F. The maximum heat load would occur with seven reload batches (413 fuel assemblies) stored in the pool at the time of discharge of the full core. For this calculation AP&L assumed that the full core is irradiated for 100 days into the eighth cycle and is cooled in the reactor vessel for 150 hours prior to its discharge to the spent fuel pool. Under these conditions the calculated heat generation rate is 28.5 x 10^6 BTU/hr. (8.35 Mw) and the spent fuel pool water temperature could be maintained at approximately 150°F by the spent fuel pool cooling system.

In addition, AP&L calculated the water temperature of the pool as a function of time following a complete loss of pool cooling capability. These calculations indicate that it would take approximately 19 hours for pool temperature to reach 212°F in the case of the normal annual 1/3 core off-load of 59 assemblies. For the maximum heat load case discussed above, AP&L's calculations show that pool temperature would reach 212°F in approximately 5 hours.

Evaluation of Spent Fuel Cooling

The NRC staff independently calculated the heat loads using the same assumptions as AP&L but utilizing the total decay energy curve of that portion of the NRC Standard Review Plan 9.25 (ULTIMATE HEAT SINK) entitled BRANCH TECHNICAL POSITION APCSB 9-2 - RESIDUAL DECAY ENERGY FOR LIGHT WATER REACTORS FOR LONG-TERM COOLING.

This conservative analysis yielded a maximum heat load of 30.5 x 10^6 BTU/hr. (8.9 Mw) based on AP&L's assumed 150 hours of decay time prior to transfer of the fuel to the spent fuel pool. It was determined that an additional 24 hour decay period would be necessary to prevent exceeding the maximum design heat load stated in the ANO-1 FSAR when calculated using the more conservative method of BRANCH TECHNICAL POSITION APCSB 9-2. The ANO-1 Technical Specifications are being amended to require 175 hour cooling time under maximum design heat load conditions prior to transferring fuel to the fuel storage pool. We have concluded this is acceptable.

AP&L has stated that the normal inventory of water in the spent fuel pool, cask pit, and tilt pool is 368,000 gallons. Assuming that all of this inventory of water could be used as a heat sink, calculations show that it would take approximately seven hours after the postulated complete failure of the fuel pool cooling system following full core off-load for the pool water to reach 212°F. This is in substantial agreement with AP&L's calculated five hours. AP&L's analysis did not include consideration for the additional inventory of water in the cask pit and tilt pool. This delay prior to reaching 212°F would allow time to make repairs or obtain an alternate cooling method.

Evaluation of Structural, Mechanical, and Material Design

Design, analysis, fabrication, and installation of the new spent fuel racks are being performed jointly by Bechtel, AP&L and Exxon Nuclear Company.

The replacement spent fuel storage racks will be fabricated from Type 304 stainless steel, totaling approximately 200,000 pounds. The racks do not use a neutron absorbing poison material. The individual fuel assemblies will be stored in square feet guide tubes fabricated from minimum .115 inch thick stainless steel. The 9-1/16 inch square storage fuel guide tubes are mounted in the rack structure on a center-to-center spacing of 13.5 inches. The storage racks consist of frames supporting the guide tubes. The basic structural function of the racks is to maintain safe geometric spacing between spent fuel assemblies during and after all applicable loading combinations and transients.

The fuel storage modules will be supported on a gridwork of fabricated I-beams oriented in both North-South and East-West directions. This gridwork of beams is designed to provide uniform support to the racks and to transmit the weight of the racks to the existing floor embedments. Provision for shimming under all support points has been made to insure a level support for the modules. Each module is located on the floor beams by dowel pins to assure the precise location of the modules required for proper operation of the fuel handling equipment. The dowel pins also transmit horizontal seismic loads to the floor beams. These loads are then transmitted to the pool walls by restraint arms bearing on the pool walls at the ends of all beams.

Our review of the structural and mechanical aspects of the AP&L proposal considered the following:

1. Supporting arrangements for the racks including their restraints.

2. Design, fabrication, and installation procedures.

3. Structural analysis for all loads including seismic and impact loadings.

4. Load combinations.
5. Structural acceptance criteria.
6. Quality Assurance requirements for design, fabrication, and installation.
7. Conformance with applicable industry codes.

Each of these were reviewed in accordance with the criteria of the NRC Standard Review Plan, Sections 3.7, 3.8, and applicable subsections.

AP&L used seismic input in the form of floor response spectra as presented in the ANO-1 FSAR. The analytical model used for seismic design utilized boundary conditions at the rack interfaces and at the support point nodes. The mass of the water enclosed in each fuel cell was lumped together with the masses of the fuel assembly and the storage cell effectively coupling the storage cell and fuel assembly. To support this coupling assumption a detailed nonlinear analysis explicitly including the clearance gap betwen the storage cell wall and the fuel assembly was performed resulting in support reactions which were compared with those of a simplified linear elastic model with no gap between the storage cell walls and the fuel assembly. Two analyses were performed utilizing two different seismic acceleration time histories developed from artificially generated 10 second duration ground motions which enveloped the floor response spectra.

The fuel racks and supporting structures were designed for the extreme conditions of being fully-loaded, undergoing a safe shutdown earthquake, and then remaining in a hot bath (212°F within the time frames discussed above in the spent fuel cooling discussion and evaluation).

AP&L also performed a review of the load carrying ability of the spent fuel pool floor and walls and found that the existing concrete structure is capable of supporting the proposed increased number of fuel assemblies and restraining the spent fuel racks during a seismic event. The temperature limits established in the FSAR for the spent fuel pool are not being changed with the proposed modification (the walls can withstand 212°F inside the pool without failure); therefore the effects of temperature gradients on the pool walls will remain unchanged. Because the original SFP design analysis also assumed that the tilt pool was dry when the SFP water temperature was 212°F, the increased temperatures also will have no effect on the wall in the tilt pool area.

AP&L also evaluated the effect of the proposed racks on the overall structural stability and seismic response of the auxiliary building. The spent fuel pool was modeled as part of the whole auxiliary building structure. It was determined that a 4% increase in mass occurs at the applicable mass point and the total mass of the building is increased by 1.5%. The results of this mass increase were found to have a negligible effect on the structural integrity of the auxiliary building and of the equipment located therein.

The criteria used in the analysis, design, and construction of the new spent fuel racks to account for anticipated loadings and postulated conditions that may be imposed upon the structures during their service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the NRC staff. The use of these criteria provides reasonable assurance that the new fuel pool structures will withstand the specified design conditions without impairment of structural integrity or the performance of required safety functions. We therefore find the structural, mechanical, and material aspects of the design acceptable.

Evaluation of Potential Accidents

Fuel Handling Accidents

Although the new storage racks provide accommodation for a larger inventory of spent fuel, the radiological consequences of a fuel handling accident are not more severe than those previously reported in the ANO-1 FSAR and the AEC's June 6, 1973 Safety Evaluation Report for ANO-1. In their letter of October 18, 1976, AP&L addressed the dropped fuel assembly accident in detail to confirm that the spacing of the fuel assemblies stored in the new racks would have no effect upon the accident.

AP&L concluded that the top 6.5 inches of the fuel storage cells, which extend above the fuel storage rack frame members, would provide energy absorbing capability for a dropped fuel assembly.

The case of a fuel assembly dropped inside a storage cell was also considered. The fuel assembly would impact the 1/2" support plate at the cell bottom. The welds attaching this plate to the storage cell are weaker than the connection of the cell to the rack frame members. The support plate attachment welds would therefore fail, dropping the fuel assembly an additional 2.5 inches where it would be stopped by the fuel storage rack base plate.

AP&L stated that the maximum drop height of a fuel assembly would be limited to 21-7/8 inches because fuel bundles will not be moved over the fuel storage racks at a higher elevation. Water drag effects were conservatively disregarded. Static compression load tests were performed on two-foot-long test samples of representative fuel cell inlet sections to determine the ductility and energy absorbing capability of the storage cells. The tests were performed for a representative selection of impact geometries, representing twisted and off-center fuel assemblies and confirmed the conservatism of the calculated values.

Because the stored fuel would be protected by the rack structure from being impacted by a dropped fuel assembly, the radiological consequences remain the same as those previously evaluated and determined acceptable for the damage to the dropped assembly itself.

Cask Drop Accident

Specification 3.8.16 assures that the spent fuel cask drop accident cannot occur prior to completion of the NRC staff's review of this potential accident and the completion of any modifications that may be necessary to preclude the accident or mitigate the consequences. Upon satisfactory completion of the NRC's review, Specification 3.8.16 shall be deleted.

Construction Accident

Because ANO-1 is currently operating in its first fuel cycle, there is no spent fuel stored in the existing racks. However, the irradiated fuel from the first core was temporarily stored in the initial racks for a short time during 1976. AP&L has completed decontamination of the pool and racks and has disposed of the original racks; therefore no radiation hazards will exist during the installation of the new rack system.

The floor beam support network and the first six rack modules, consisting of approximately 200 storage spaces, are scheduled to be installed in early January 1977 while the pool is dry. These modules will be placed at the south end of the pool. The remaining twelve modules will be installed later during the first quarter of 1977 when the pool is flooded. The installation of the first six modules in the south end of the pool permit the other modules to be installed without being carried over any of the installed modules, thus minimizing the chance of a drop onto the spent fuel to be stored in the installed modules.

In chapter 9 of the ANO-1 FSAR AP&L discussed the consequences of a spent fuel cask drop on the relay panel room ceiling, over which the cask (and therefore the new racks and support structure) must traverse enroute to the pool. The relay panel room ceiling is designed to absorb the energy of a cask drop. Casks in nuclear industry usually range from 25 tons to 100 tons and are thus much heavier than the loads to be transported during the spent fuel pool modifications. The heaviest rack module to be installed will weigh 1400 lb. In addition, the area over which the drop weight of a module or support-structure element would be spread is much greater than that over which the cask would impact. On this basis, we conclude that the loads to be carried during the spent fuel pool modifications can be handled safely and that the proposed construction activities can be performed with reasonable assurance that no damage to stored fuel or any safety-related equipment or structures will occur.

Conclusion

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.

II. ENVIRONMENTAL IMPACT APPRAISAL

On September 16, 1975, the Commission announced (40 F.R. 42801) its intent to prepare a generic environmental impact statement on handling the storage of spent fuel from light water reactors. In this notice, the Commission also announced its conclusion that it would not be in the public interest to defer all licensing actions intended to ameliorate a possible shortage of spent fuel storage capacity pending completion of the generic environmental impact statement. The Commission directed that in the consideration of any such proposed licensing action, the following five specific factors should be applied, balanced, and weighed in the context of the required environmental statement or appraisal.

a. Is it likely that the licensing action here proposed would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity?

The Arkansas Nuclear One Unit No. 1 reactor core contains 177 fuel assemblies. The facility achieved initial criticality on August 6, 1974 and commenced commercial operation on December 19, 1974. The spent fuel pool, SFP, was designed on the basis that a fuel cycle would be established which would require storage of spent fuel for only one year prior to shipment to a reprocessing facility. Therefore, a pool storage capacity for 253 assemblies (1-1/3 cores) was considered adequate. This provided for complete unloading of the reactor even if the spent fuel from the previous refueling was in the pool. Typically, Arkansas Nuclear One Unit No. 1 replaces about one-third of the core at each refueling. With the existing storage racks, without annual shipment of fuel, full core discharge would not be possible after the scheduled January 1978 refueling. If one-third core is discharged each year, the SFP will be filled after the refueling scheduled for January 1980. If spent fuel could not be shipped offsite or stored elsewhere, prior to January 1981, operation of the reactor would have to be terminated.

Since spent fuel reprocessing facilities cannot assuredly be available to Arkansas Power and Light Company prior to the mid-1980's (and, therefore, no spent fuel can be shipped for reprocessing), spent fuel discharges subsequent to 1980 will have to be stored or the facility shut down. The proposed SFP modification to increase the storage capacity for irradiated fuel would provide the licensee with additional operating flexibility which is desirable even if adequate offsite storage facilities hereafter become available to the licensee.

We have concluded that a need for additional spent fuel storage capacity exists at Arkansas Nuclear One Unit No. 1 which is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity.

b. Is it likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity?

With respect to this proposed licensing action, we have considered commitment of both material and nonmaterial resources. The material resources considered are those to be utilized in the expansion of the SFP.

Under the proposed modification, the present spent fuel racks will be replaced by new spent fuel racks that will increase the storage capacity to 590 assemblies. There will be a total of 18 new racks, 10 will contain 36 storage cells, 7 will contain 30 storage cells and one will contain 20 storage cells. The total quantity of stainless to be utilized in the new spent fuel racks is approximately 200,000 pounds. The racks do not use a poison material such as boron impregnated stainless steel, B_4C plates or boral. The amount of stainless steel used annually in the U. S. is about 2.82×10^{11} lbs. The material is readily available in abundant supply. We conclude that the amount of material required for the new racks at ANO-1 is insignificant and does not represent an irreversible commitment of natural resources. In addition, the existing SFP racks -- which are also fabricated from Type 304 Stainless -- will probably be sold as scrap or used in another facility, thus reducing the total consumption. This licensing action would not constitute a commitment of resources that would affect the alternatives available to other nuclear power plants or other actions that might be taken by the industry in the future to alleviate fuel storage problems. No other resources need be allocated because the other design characteristics of the SFP remain unchanged. No additional allocation of land would be made; the land area now used for the SFP would be used more efficiently by reducing the spacings among fuel assemblies.

The increased storage capacity at the ANO-1 spent fuel pool was considered as a nonmaterial resource and was evaluated relative to proposed similar licensing actions within a one year period (the time we estimate is necessary to complete the generic environmental statement) at other nuclear power plants, fuel reprocessing facilities and fuel storage facilities. We have determined that the proposed expansion in the storage capacity of the SFP is only a measure to allow for continued operation and to provide operational flexibility at the facility, and will not preclude similar licensing actions at other nuclear power plants.

We conclude that the expansion of the spent fuel pool at the Arkansas Nuclear One - Unit 1 facility does not constitute a commitment of either material or nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity.

c. Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overlooking any cumulative environmental impacts?

The SFP at ANO-1 was designed principally to store spent fuel assemblies prior to shipment to a reprocessing facility. These assemblies may be transferred from the reactor core to the SFP during a core refueling, or to allow for inspection and/or modification to core internals (which may require the removal and storage of up to a full core). The assemblies are initially highly radioactive due to their fission product content and have a high thermal output. They are stored in the SFP to allow for radioactive and thermal decay.

The major portion of decay occurs during the 150 day period following removal from the reactor core. After this period, the assemblies may be withdrawn and placed into a heavily shielded fuel cask for offsite shipment. Space permitting, the assemblies may be stored for an additional period allowing continued fission product decay and thermal cooling prior to shipment. Presently, the ANO-1 SFP contains no spent fuel assemblies but during the first refueling, scheduled for January 1977, approximately 72 assemblies will be transferred to the SFP.

Since the additional capacity of the SFP is proposed for this site alone and for this licensee only, all the environmental impacts can be assessed within the context of this application. Potential non-radiological and radiological impacts resulting from the fuel rack conversion and subsequent operation of the expanded SFP at this facility were considered by the Staff. No environmental impacts on the environs outside the spent fuel storage building were identified during the proposed construction of the expanded SFP. The impacts within this building are expected to be limited to those normally associated with metal working activities.

No significant environmental impacts, either onsite or offsite, could be identified as resulting from operation of an expanded SFP at this facility.

The only potential offsite nonradiological environmental impact that could arise from this proposed action would be an additional discharge of heat to the Dardanelle Reservoir. Storing spent fuel in the SFP for a longer period of time will add more heat to the SFP water. The spent fuel pool heat exchanger is cooled by the intermediate cooling water system, which in turn is cooled by the service water system. The increase in heat loading is about 2.5 MBTU/Hr. Compared to the existing heat load on the service water system and the total heat rejected to the reservoir by the once through circulating water system (about 5700 MBTU/Hr) the small additional heat load from the SFP cooling system (attributable to the longer storage of additional spent fuel) will be negligible.

The potential offsite radiological environmental impact associated with this expansion (resulting from an incremental addition in the long-lived radioactive effluents released from the facility) was evaluated and has been determined to be environmentally insignificant as addressed below.

The expansion of the SFP will allow spent fuel to be stored for an additional six-year period without shipment offsite and still maintain space to off-load a full core. 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 to radiation levels to which workers in and near the SFP would be exposed. The violatile 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.

The ANO-1 Spent Fuel Pool Cooling System (SFPCS) is designed to maintain the SFP water temperature less than or equal to 120°F during normal refueling operations and at approximately 150°F during full core discharge situations. The SFPCS is described in Section 9.4 of the Final Safety Analysis Report (FSAR). In addition to its primary function, the system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank. This removes radioactive fission and corrosion products and maintains water clarity for fuel handling operations. The two spent fuel pool circulating pumps take suction from the spent fuel pool and recirculate the fluid back to the pool after passing through the two coolers. Part of the flow is diverted through the demineralizer and filters. During refueling operations these pumps are also used for filling the fuel transfer canal and incore instrumentation tank with borated water from the borated water storage tank.

A bypass purification loop is provided to maintain the purity of the water in the spent fuel pool. This loop is also utilized to purify the water in the borated water storage tank following refueling and to maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or fuel transfer canal can be purified by using the borated water recirculation pump. The total volume of water in the spent fuel pool, cask pit and tilt pit at normal pool level is 368,000 gallons. The spent fuel pool purification loop utilizes a filter and a 20ft^3 nonregenerative mixed bed demineralizer at a flow rate of 180 gpm. One volume of the spent fuel pool can thus be processed in approximately 34 hours. Storing additional spent fuel in the SFP may increase the amount of corrosion and fission product nuclides introduced into the SFP water. The purification system is capable of removing the increased radioactivity to maintain acceptable radiation levels above and in the vicinity of the pool. Redesign of the SFP racks increases only the storage capacity of the pool and not the frequency or the amount of the core to be replaced for each fuel cycle. Thus, the amount of corrosion product nuclides released into the pool during any year will be about the same regardless of the length of time or number of assemblies stored in the pool. Expansion of the capacity does increase the potential for increasing the amount of fission products introduced into the SFP water. This could increase the amount of radioactivity accumulated on the filter and demineralizer which are disposed of as solid waste.

The design basis for the filter and demineralizer is to replace these units on high differential pressure at a frequency of about once per year. Arkansas Power and Light Company has indicated that the proposed modification will not change the basis for replacing the filter and demineralizer and that the expected replacement frequency should not be significantly altered by the increase in storage capacity. Thus, the licensee does not anticipate that the radioactive solid waste generated from the ANO-1 facility will be increased as a result of the modification. As a conservative estimate, we have assumed that the amount of solid radwaste may be increased by two additional resin beds a year. During 1975, an average of 9500 cubic feet of solidified waste was shipped offsite from pressurized water reactors. If the increased storage of spent fuel does increase the amount of solid waste by 40 cubic feet per year, the increase in total waste volume would be less than 1% and would not have any significant additional environmental impact.

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee and by utilizing realistic assumptions for radionuclide concentrations in the SFP water and for occupancy times. 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. Our analysis indicates that the occupational radiation exposure resulting from the proposed action represents a negligible burden. Based on present and projected operations in the spent fuel pool area, the proposed modification will add less than one percent to the total annual occupational radiation exposure burden at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses as low as reasonably achievable and within the limits of 10 CFR 20. Thus, we conclude that storing additional fuel in the SFP will not result in any significant increase in doses received by occupational workers.

With respect to gaseous releases from the SFP, the only significant noble gas isotope remaining in the SFP and attributable to storing additional assemblies for a longer period of time would be Krypton-85, since shorter lived noble gases will have decayed to negligible amounts. Based on operating experience for Zircaloy clad fuel (see NUREG-0017), we have assumed that 0.12% of all fuel rods have cladding defects which permit the escape of fission product gases. This value is the weighted average percent defective fuel for nine pressurized water reactors. It is assumed that the fission product gases escape on a relatively linear basis with time. On this basis, we have conservatively estimated that an additional 44 curies per year of Krypton-85 will be released when the modified pool is completely filled. The fuel storage pool area is continuously ventilated. This air is released through the containment vent. If the plant does eventually release an additional 44 curies per year of Kr-85 as a result of the proposed modification, the increase would result in an additional offsite dose of less than 0.1 mrem/year. This dose is insignificant when compared to the approximately 100 mrem/year that an individual receives from natural background radiation. Thus, we conclude that the proposed modification will not have any significant impact on radiation levels or personnel exposure offsite.

Assuming that the spent fuel will be stored onsite for several years (rather than shipped offsite after 6 to 12 months storage as originally planned), Iodine-131 releases will not be significantly increased by the expansion of the fuel storage capacity since the Iodine-131 inventory in the fuel will decay to negligible levels between each annual refueling. Storing additional spent fuel assemblies is not expected to increase the bulk water temperature above the 120°F used in the design analysis during normal refuelings or above approximately 150°F during a full core off-load. Since the temperature of the pool water will normally be maintained below 120°F, it is not expected that there will be any significant change in evaporation rates and the release of tritium as a result of the proposed modification.

The licensee will be prohibited from moving the spent fuel cask in the spent fuel building until either a cask drop/tip analysis is complete or it is determined that the overhead handling system meets the intent of proposed Regulatory Guide 1.104. In addition, the maximum weight of loads which may be transported over spent fuel may not be substantially in excess of that of a single fuel assembly. The consequences of accidents similar to fuel handling accidents therefore remain unchanged from those previously evaluated.

We have considered the potential cumulative environmental impacts associated with the expansion of the SFP and have concluded that they will not result in radioactive effluent releases that significantly affect the quality of the human environment during either normal operation of the expanded SFP or under postulated fuel handling accident conditions.

d. Have all technical issues which have arisen during the review of this application been resolved within that context?

This impact appraisal and the accompanying safety evaluation report point out that all questions concerning health, safety and environmental concerns have been answered.

e. Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?

In regard to this licensing action, the staff has considered the following alternatives: (1) shipment of spent fuel to a fuel reprocessing facility, (2) shipment of spent fuel to a separate fuel storage facility, (3) shipment of spent fuel to another reactor site, and (4) ceasing operation of the facility. These alternatives are considered in turn.

The Arkansas Power & Light Company has entered into a contract with the Exxon Nuclear Company, Inc., of Richland, Washington for the design, analysis, and fabrication of replacement spent fuel storage racks for 590 fuel assemblies. These replacement spent fuel storage racks will provide storage capacity for approximately 3-1/3 cores through 1987. Therefore, 10 annual discharges may be accommodated or 7 annual discharges may be accommodated while still maintaining the capability for a full core discharge until 1984. The contract price for the design and fabrication of the replacement racks is approximately $1,100,000 including estimated freight charges. The current estimate for removal of the existing racks and installation of the new racks is $200,000. This gives a total construction cost of $1,300,000 for the spent fuel rack modification. While this is expensive, the alternatives are more costly.

(1) As discussed earlier, there are no storage and/or reprocessing facilities in the U.S. that are presently able to contract for the storage and reprocessing of spent fuel. With the present spent fuel storage and reprocessing situation, it appears unlikely that shipment of spent fuel to any such facilities could be made within the next several years. Currently, both the Nuclear Fuel Services (NFS) and the General Electric Company's reprocessing plants are in a decommissioned condition. Arkansas Power & Light Company does not have access to this storage. Reprocessing of the first six batches of fuel from ANO-1 is contracted to Allied General Nuclear Services (AGNS). However, the AGNS plant is not licensed to operate and cannot be depended upon for receipt of spent fuel until all issues relating to the Generic Environmental Statement on Mixed Oxide Fuel (GESMO), spent fuel shipment and waste disposal have been settled. Therefore, shipment of spent fuel to a reprocessing plant is not an available alternative for several more years.

(2) An alternative to expansion of onsite spent fuel pool storage is the construction of new "independent spent fuel storage installations" (ISFSI). Such installations could provide storage space in excess of 1000 MTU of spent fuel. This is far greater than the capacities of onsite storage pools. An ISFSI could be designed using dry storage technology. Fuel storage pools at GE Morris and NFS are functioning as ISFSIs although they were not designed to solely serve this purpose.

Regulatory Guide 3.24, "Guidance on the License Application, Siting, Design, and Plant Protection for an Independent Spent Fuel Storage Installation " issued in December 1974, recognizes the possible need for ISFSIs and provides recommended criteria and requirements for water-cooled ISFSIs. Pertinent sections of 10 CFR Parts 19, 20, 30, 40, 51, 70, 71, and 73 would also apply.

It is estimated that at least five years would be required for completion of an independent fuel storage facility. This estimate assumes one year for preliminary design, one year for preparation of the license application, Environmental Report, and licensing review in parallel with one year for detail design, two and one-half years for construction and receipt of an operating license, and one-half year for plant and equipment testing and startup.

Industry proposals for independent spent fuel storage facilities are scarce to date. In late 1974, E. R. Johnson Associates, Inc. and Merril Lynch, Pierce, Fenner and Smith, Inc. issued a series of joint proposals to a number of electric utility companies having nuclear plants in operation or contemplated for operation, offering to provide independent storage services for spent nuclear fuel. A paper on this proposed project was presented at the American Nuclear Society meeting in November 1975. The Commission has not received any license requests for facilities conceived and designed only to store spent fuel. In 1974, E. R. Johnson Associates estimated their construction cost at approximately $9000 per spent fuel assembly. At this rate, it would cost the licensee over $3,000,000 to store the additional 337 spent fuel assemblies that the proposed modification will accommodate, plus there would be additional costs for shipment and safeguarding the fuel. An independent spent fuel storage installation is not a viable alternative based on cost or availability in time to meet the licensee's needs. It is also unlikely that the total environmental impacts of constructing an independent facility and shipment of spent fuel would be less than the minor impacts associated with the proposed action.

(3) According to a survey conducted and documented by the Energy Research and Development Administration, up to 46 percent of the operating nuclear power plants will lose the ability to refuel during the period 1975-1984 without additional spent fuel storage pool expansions or access to offsite storage facilities. Thus, the licensee cannot assuredly rely on any other power facility to provide additional storage capability except on a short-term emergency basis.

(4) Storage in the existing racks is possible but only for a short period of time. The first batch of spent fuel will be discharged in January 1977, and an additional batch of spent fuel will be discharged annually thereafter. Therefore, after the second refueling in January 1978, the reactor will be operating without a full core discharge capability. The fuel could not be discharged from the core in case of an emergency shutdown for inspection of the vessel or removal of the core internals. ANO-1 would be unable to discharge the normal batch of spent fuel by January 1981 because the existing racks would already be filled with spent fuel. In either of the above two situations the purchase of replacement power would be approximately $450,000 for each day the reactor was not operating. Besides being an unacceptable alternative, within three days the cost of replacement power would exceed the cost of the proposed action.

In summary, the alternatives (1) to (3) described above do not offer the operating flexibility of the proposed action nor could they be completed as rapidly as the proposed action. The alternatives of shipping the spent fuel to a reprocessing facility, an independent storage facility or to another reactor would be more expensive than the proposed action and might preempt storage space needed by another utility. The alternative of ceasing operation of the facility also would be more expensive than the proposed action because of the need to provide replacement power. In addition to the economic advantages of the proposed action, we have determined that the expansion of the SFP would have a negligible environmental impact. Accordingly, deferral or severe restriction of the action here proposed would result in substantial harm to the public interest.

Basis and Conclusion for not Preparing an Environmental Impact Statement

We have reviewed this proposed facility modification relative to the requirements set forth in 10 CFR Part 51 and the Council of Environmental Quality's Guidelines, 40 CFR 1500.6 and have applied, weighed, and balanced the five factors specified by the Nuclear Regulatory Commission in 40 FR 42801. We have determined that the license amendment will not significantly affect the quality of the human environment. Therefore, the Commission has found 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.

Date: December 17, 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT 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. 17 to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised the license and its appended Technical Specifications for operation of Arkansas Nuclear One - Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

This amendment authorized changes in the design of the ANO-1 spent fuel storage pool from that reviewed and approved in the operating license review and as described in the ANO-1 Final Safety Analysis Report. The changes will increase spent fuel storage capacity from 253 to 590 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 Consideration of Proposed Modification to Facility Spent Fuel Storage Pool in connection with this action was published in the FEDERAL REGISTER on October 28, 1976 (41 F.R. 47294). 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 there will be no significant environmental impact attributable to the action.

For further details with respect to this action, see (1) the application for amendment dated October 7, as supplemented by letters dated October 18, October 25, November 11, November 16, and November 19, 1976, (2) Amendment No. 17 to Facility Operating License No. DPR-51 and (3) the Commission's related Safety Evaluation and 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 Arkansas Polytechnic College, Russellville, Arkansas 72801. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 17th day of December, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors