April 15, 1983

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Dockets Nos. 50-313
and 50-368

Mr. John M. Griffin, Vice President
Nuclear Operations
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Dear Mr. Griffin:

The Commission has issued the enclosed Amendments Nos. 76 and 43 to Facility Operating Licenses Nos. DPR-51 and NPF-6 for Arkansas Nuclear One, Units Nos. 1 and 2 (ANO-1 & 2). These amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 5, 1982, supplemented by letters dated February 17, 1983, March 3, 7, 10, 21, 22, 24, 28 and 29, 1983, and April 5 and 7, 1983.

These amendments allow an increase in the storage capacity for the ANO-1 spent fuel pool from 589 to 968 storage locations and of the ANO-2 spent fuel pool from 485 to 988 storage locations.

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

Sincerely,

Original signed by

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Original signed by

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 76 to DPR-51
2. Amendment No. 43 to NPF-6
3. Safety Evaluation
4. Environmental Impact Appraisal
5. Notice/Negative Declaration

OELD M. KARMAN
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Arkansas Power & Light Company

50-313, Arkansas Nuclear One, Unit 1
50-368, Arkansas Nuclear One, Unit 2

cc:

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Manager, Licensing
Arkansas Power & Light Company
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Little Rock, Arkansas 72203

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General Manager
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Region VI Office
ATTN: Regional Radiation
Representative
1201 Elm Street
Dallas, Texas 75270

Mr. Frank Wilson
Director, Division of Environmental
Health Protection
Arkansas Department of Health
4815 West Markman Street
Little Rock, Arkansas 72201

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

April 15, 1983

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Docket No. 50-313, 50-368

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: ARKANSAS NUCLEAR ONE, UNITS NOS. 1 AND 2

Two signed originals of the Federal Register Notice identified below are enclosed for your transmittal to the Office of the Federal Register for publication. Additional conformed copies (12) of the Notice are enclosed for your use.

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- XX Other: Amendments Nos. 76 and 43.
  Referenced documents have been provided PDR.

Division of Licensing, ORB#4
Office of Nuclear Reactor Regulation

Enclosure:
As Stated

OFFICE→	ORB#4:DL					
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DATE→	4/18/83					

NRC FORM 102 (1-76)

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO.1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-51

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated November 5, 1982, as supplemented February 17, 1983, and April 7, 1983, 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. DPR-51 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Chief
Operating Reactors Branch #4
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by Amendment Number and contain vertical lines indicating the areas of change.

Page

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59a
59b
59c (new page)
59d (new page)
116
127

3.8.6 During the handling of irradiated fuel in the reactor building, at least one door on the personnel and emergency hatches shall be closed. The equipment hatch cover shall be in place with a minimum of four bolts securing the cover to the sealing surfaces.

3.8.7 Isolation valves in lines containing automatic containment isolation valves shall be operable, or at least one shall be closed.

3.8.8 When two irradiated fuel assemblies are being moved simultaneously by the bridges within the fuel transfer canal, a minimum of 10 feet separation shall be maintained between the assemblies at all times.

3.8.9 If any of the above specified limiting conditions for fuel loading and refueling are not met, movement of fuel into the reactor core shall cease; action shall be initiated to correct the conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.11 Irradiated fuel shall not be removed from the reactor until the unit has been subcritical for at least 72 hours. In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

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. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.14 Loads in excess of 2000 pounds shall be prohibited from travel over fuel assemblies in the storage pool. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.15 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. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.16 Storage in the spent fuel pool shall be restricted to fuel assemblies having initial enrichment less than or equal to 4.1 w/o U-235. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.17 Storage in Region 2 (as shown on Figure 3.8.1) of the spent fuel pool shall be further restricted by burnup and enrichment limits specified in Figure 3.8.2. The provisions of Specifications 3.0.3 and 3.0.4 are not applicable.

3.8.18 The boron concentration in the spent fuel pool shall be maintained (at all times) at greater than 1600 parts per million.

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 9.6 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 requirement that at least one decay heat removal loop be in operation ensures that (1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel at the refueling temperature (normally 140°F), and (2) sufficient coolant circulation is maintained through the reactor core to minimize the effects of a boron dilution incident and prevent boron stratification.(1)

The requirement to have two decay heat removal loops operable when there is less than 23 feet of water above the core, ensures that a single failure of the operating decay heat removal loop will not result in a complete loss of decay heat removal capability. With the reactor vessel head removed and 23 feet of water above the core, a large heat sink is available for core cooling, thus in the event of a failure of the operating decay heat removal loop, adequate time is provided to initiate emergency procedures to cool the core.

The shutdown margin indicated in Specification 3.8.4 will keep the core subcritical, even with all control rods withdrawn from the core.(2) Although the refueling boron 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: 1) the safety analysis for the fuel handling accident was based on the assumption that the reactor had been shutdown for 72 hours.[3]; and, 2) to assure that the maximum design heat load of the spent fuel pool cooling system will not be exceeded during a full core offload.

Specification 3.8.14 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 3.8.15 is completed.

Specification 3.8.15 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 3.8.15 shall be deleted.

Specifications 3.8.16 and 3.8.17 assure fuel enrichment and fuel burnup limits assumed in the spent fuel safety analyses will not be exceeded.

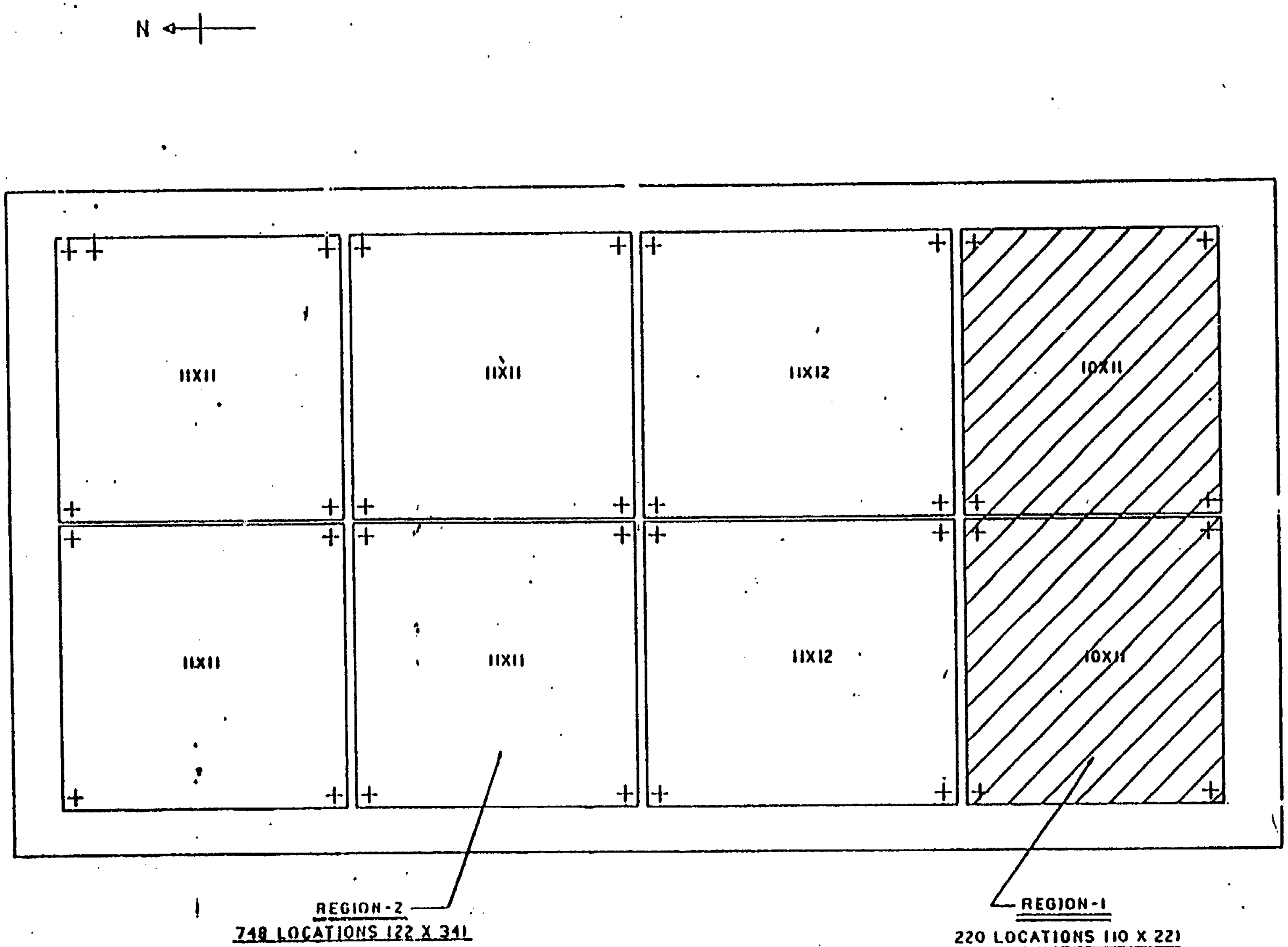
Specification 3.8.18 assures the boron concentration in the spent fuel pool will remain within the limits of the spent fuel pool accident and criticality analyses.

REFERENCES

(1) FSAR, Section 9.5

(2) FSAR, Section 14.2.2.3

(3) FSAR, Section 14.2.2.3.3



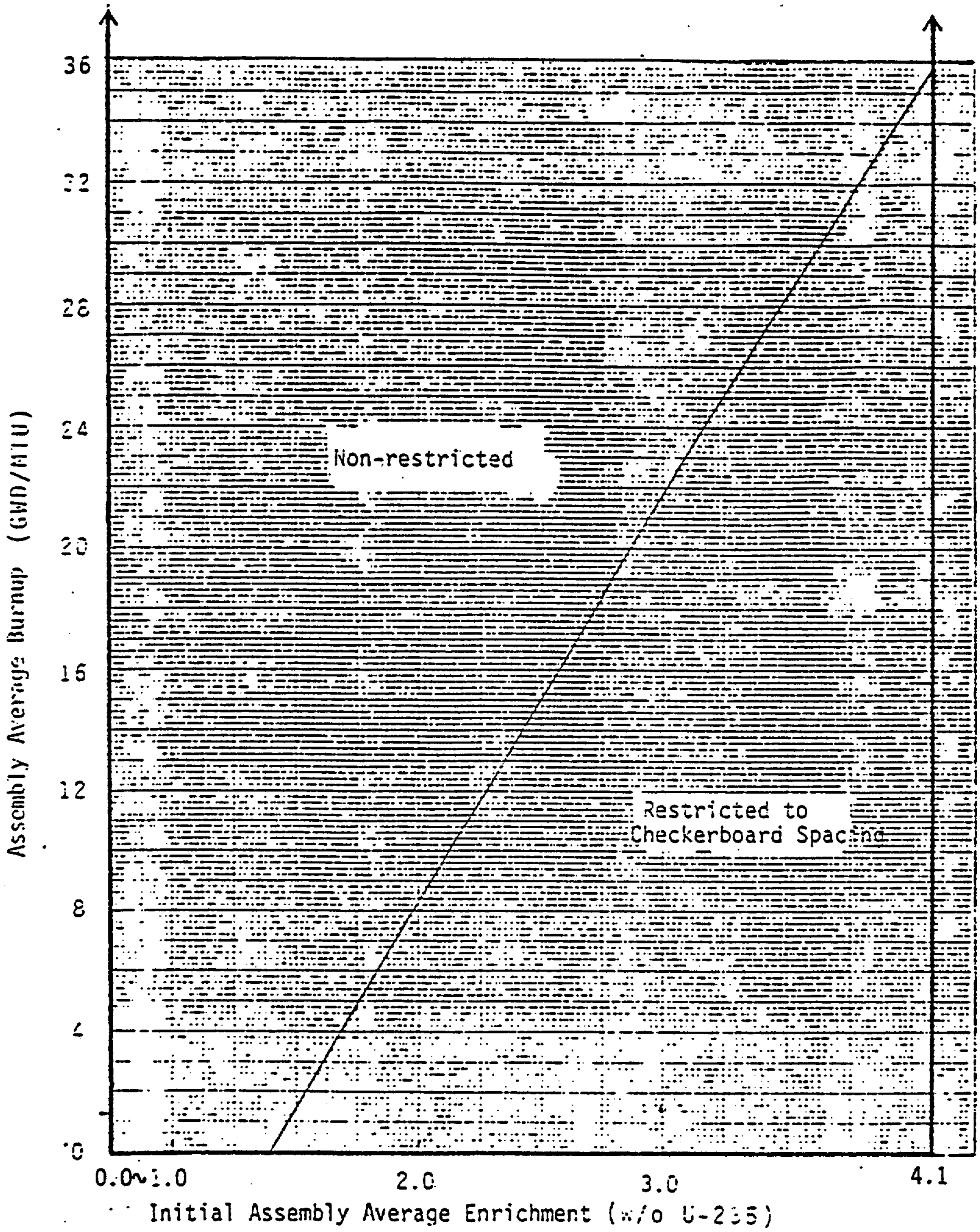
SPENT FUEL POOL ARRANGEMENT UNIT #1

Figure 3.8.1

FIGURE 3.8.2

MINIMUM BURNUP VS. INITIAL ENRICHMENT

FOR REGION 2 ENRICHMENT



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. Fuel assemblies are stored in racks of 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 be stored in the spent fuel pool or in its shipping containers.

5.4.2 Spent Fuel Storage

1. The spent fuel racks are designed and shall be maintained so that the calculated effective multiplication factor is no greater than 0.95 (including all known uncertainties) when the pool is flooded with unborated water.

2. The spent fuel pool and the new fuel pool racks are designed as seismic Class I equipment.

REFERENCES

FSAR, Section 9.6

a. The facility shall be placed in at least hot shutdown within one hour.

b. The Nuclear Regulatory Commission shall be notified and a report submitted pursuant to the requirements of 10 CFR 50.36 and Specification 6.12.3.1.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, November, 1972.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

6.8.3 Temporary changes to procedures of 6.8.1 above may be made provided:

a. The intent of the original procedure is not altered.

b. The change is approved by two members of the plant staff, at least one of whom holds a Senior Reactor Operator's License on the unit affected.

c. The change is documented, reviewed by the PSC and approved by the General Manager within 14 days of implementation.

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO.2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 43
License No. NPF-6

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

   A. The application for amendment by Arkansas Power and Light Company (the licensee) dated November 5, 1982, as supplemented February 17, 1983, and April 7, 1983, 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-6 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 43 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 15, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 43

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change. Corresponding overleaf pages are provided to maintain document completeness.

Pages

VIII

3/4 9-3

3/4 9-14

3/4 9-15

3/4 9-16

6-13

B 3/4 9-1

B 3/4 9-3

LIMITING CONDITIONS FOR OPERATION AND SURVEILLANCE REQUIREMENTS

DECAY TIME AND SPENT FUEL STORAGE

LIMITING CONDITION FOR OPERATION

3.9.3.a The reactor shall be subcritical for at least 72 hours.

3.9.3.b In the event of a complete core offload, a full core to be discharged shall be subcritical a minimum of 175 hours prior to discharge of more than 70 assemblies to the spent fuel pool.

APPLICABILITY: During movement of irradiated fuel in the reactor pressure vessel.

ACTION:

With the reactor subcritical for less than 72 hours, suspend all operations involving movement of irradiated fuel in the reactor pressure vessel. With the reactor subcritical for less than 175 hours, suspend all operations involving movement of more than 70 fuel assemblies from the reactor pressure vessel to the spent fuel pool. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.3.a The reactor shall be determined to have been subcritical for at least 72 hours by verification of the date and time of subcriticality prior to movement of irradiated fuel in the reactor pressure vessel.

4.9.3.b The reactor shall be determined to have been subcritical for at least 175 hours by verification of the date and time of subcriticality prior to movement of the 71st irradiated fuel assembly from the reactor pressure vessel to the spent fuel pool.

SURVEILLANCE REQUIREMENTS (Continued)

2. Verifying with 31 days after removal that laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

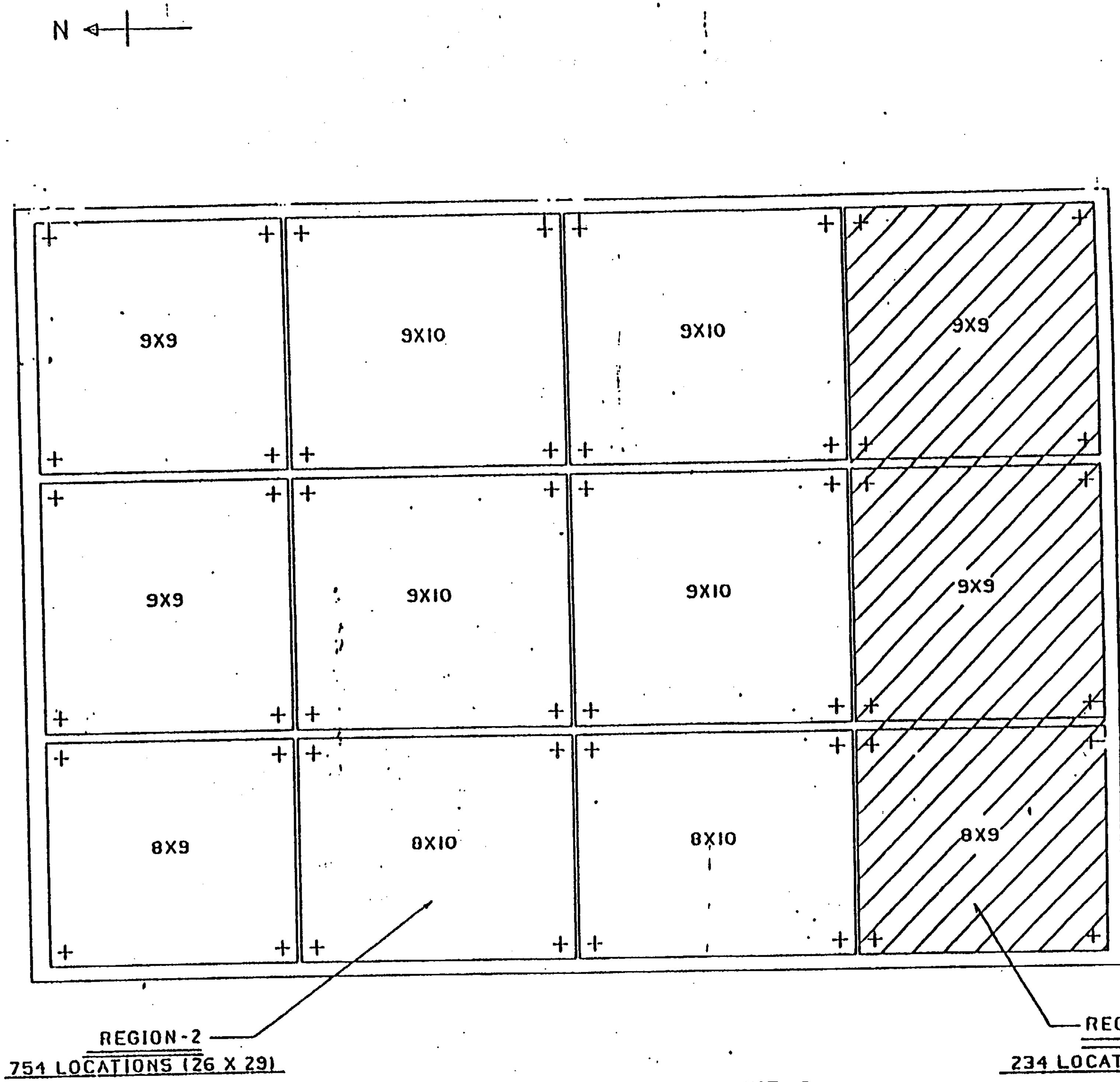
3. Verifying a system flow rate of 39,700 cfm ± 10% during system operation when tested in accordance with ANSI N510-1975.

b. After every 720 hours of charcoal adsorber operation by verifying within 31 days after removal that a laboratory analysis of a representative carbon sample obtained in accordance with Regulatory Position C.6.b of Regulatory Guide 1.52, Revision 2, March 1978, meets the laboratory testing criteria of Regulatory Position C.6.a of Regulatory Guide 1.52, Revision 2, March 1978.

c. At least once per 18 months by verifying that the pressure drop across the combined HEPA filters and charcoal adsorber banks is < 6 inches Water Gauge while operating the system at a flow rate of 39,700 cfm ± 10%.

d. After each complete or partial replacement of a HEPA filter bank by verifying that the HEPA filter banks remove ≥ 99% of the DOP when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm ± 10%.

e. After each complete or partial replacement of a charcoal adsorber bank by verifying that the charcoal adsorbers remove ≥ 99.95% of a halogenated hydrocarbon refrigerant test gas when they are tested in-place in accordance with ANSI N510-1975 while operating the system at a flow rate of 39,700 cfm ± 10%.



SPENT FUEL POOL ARRANGEMENT UNIT #2

Figure 3.9.1

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The unit shall be placed in at least HOT STANDBY within one hour.

b. The Safety Limit violation shall be reported to the Commission, the Vice President, Nuclear Operations and to the SRC within 24 hours.

c. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PSC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon facility components, systems or structures, and (3) corrective action taken to prevent recurrence.

d. The Safety Limit Violation Report shall be submitted to the Commission, the SRC and the Vice-President, Nuclear Operations within 14 days of the violation.

6.8 PROCEDURES

6.8.1 Written procedures shall be established, implemented and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix "A" of Regulatory Guide 1.33, Revision 2, February 1978.

b. Refueling operations.

c. Surveillance and test activities of safety related equipment.

d. Security Plan implementation.

e. Emergency Plan implementation.

f. Fire Protection Program implementation.

g. Modification of Core Protection Calculator (CPC) Addressable Constants
NOTE: Modification to the CPC addressable constants based on information obtained through the Plant Computer - CPC data link shall not be made without prior approval of the Plant Safety Committee.

h. New and spent fuel storage.

6.8.2 Each procedure of 6.8.1 above, and changes thereto, shall be reviewed by the PSC and approved by the General Manager prior to implementation and reviewed periodically as set forth in administrative procedures.

BASES

3/4.9.1 BORON CONCENTRATION

The limitations on reactivity conditions during REFUELING ensure that: 1) the reactor will remain subcritical during CORE ALTERATIONS, and 2) a uniform boron concentration is maintained for reactivity control in the water volume having direct access to the reactor vessel. These limitations are consistent with the initial conditions assumed for the boron dilution incident in the accident analyses.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of the source range neutron flux monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 DECAY TIME

The minimum requirement for reactor subcriticality prior to movement of irradiated fuel assemblies in the reactor pressure vessel ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

The minimum requirement for reactor subcriticality prior to movement of more than 70 irradiated fuel assemblies to the spent fuel pool ensures that sufficient time has elapsed to allow radioactive decay of the short lived fission products such that the heat generated will not exceed the cooling capacity of the spent fuel pool cooling system. This decay time and total assembly limitation is conservatively within the assumptions used in the accident analyses.

3/4.9.4 CONTAINMENT PENETRATIONS

The requirements on containment penetration closure and OPERABILITY of the containment purge and exhaust system HEPA filters and charcoal adsorbers ensure that a release of radioactive material within containment will be restricted from leakage to the environment or filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The OPERABILITY and closure restrictions are sufficient to restrict radioactive material release from a fuel element rupture based upon the lack of containment pressurization potential while in the REFUELING MODE. Operation of the containment purge and exhaust system HEPA filters and charcoal adsorbers and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.9 and 3/4.9.10 WATER LEVEL-REACTOR VESSEL AND SPENT FUEL POOL WATER LEVEL

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. The minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.11 FUEL HANDLING AREA VENTILATION SYSTEM

The limitations on the fuel handling area ventilation system ensure that all radioactive materials released from an irradiated fuel assembly will be filtered through the HEPA filters and charcoal adsorbers prior to discharge to the atmosphere. The operation of this system and the resulting iodine removal capacity are consistent with the assumptions of the accident analyses.

3/4.9.12 FUEL STORAGE

Region 1 of the spent fuel storage racks is designed to assure fuel assemblies of less than or equal to 4.1 w/o U-235 enrichment will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

Region 2 of the spent fuel storage racks is designed to assure fuel assemblies within the burnup and initial enrichment limits of Figure 3.9.2 will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in unborated water. These conditions have been verified by criticality analyses.

The requirement for 1600 ppm boron concentration is to assure the fuel assemblies will be maintained in a subcritical array with $K_{eff} \leq 0.95$ in the event of a postulated accident.

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

SAFETY EVALUATION

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 TO FACILITY OPERATING LICENSE NO. DPR-51

AND

AMENDMENT NO. 43 TO FACILITY OPERATING LICENSE NO. NPF-6

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 & 2

DOCKETS NOS. 50-313 and 50-368

TABLE OF CONTENTS

1.0 Introduction

By letter dated November 5, 1982 (Ref 1), supplemented by References 2 through 14, Arkansas Power and Light Company (the licensee or AP&L) proposed amendments to Facility Operating Licenses Nos. DPR-51 and NPF-6 for Arkansas Nuclear One, Units Nos. 1 and 2 (ANO-1&2). The proposed amendments would revise the provisions in the Technical Specifications (TSs) to allow modifications in the spent fuel design for ANO-1&2 which would increase the spent fuel storage capabilities for ANO-1 from 589 spaces to 968 spaces and for ANO-2 from 485 spaces to 988 spaces. This expansion would be accomplished by replacing the existing spent fuel storage racks with new high density storage racks.

The proposed change would allow refueling capability through the 15th refueling scheduled for the spring of 1998 for ANO-1 and through the 14th refueling scheduled for the spring of 2000 for ANO-2. Present storage capacities would force the shutdown of ANO-1&2 in 1989 due to the inability to refuel.

As addressed below, we have evaluated the safety considerations associated with the proposed changes to the ANO-1&2 spent fuel storage designs. A separate Environmental Impact Appraisal addressing these changes has been prepared.

2.0 Evaluation

2.1 Criticality Considerations

For both ANO-1&2, the spent fuel storage racks are divided into two regions. Region 1 of each unit is designed to accommodate non-irradiated fresh fuel and is sized to permit core offloads. Storage in Region 2 for each unit is restricted by burnup and enrichment limits. Placement of fuel in Region 2 is determined by burnup calculations and controlled administratively by AP&L. Fuel which does not meet the burnup criterion may be placed in Region 2 in a checkerboard arrangement. In these cases, the vacant spaces adjacent to the assembly being inserted will be physically blocked to prevent inadvertent assembly insertion. In addition, the area designated will be subdivided from the normal storage in Region 2 by a row of vacant storage spaces. The criticality aspects of the design of each region are discussed separately below.

2.1.1 Region 1 Design

The Region 1 racks consist of individual stainless steel storage cells with a neutron absorbing material, Boraflex, attached to each cell. There are 234 fuel assembly storage locations with a 10.65 inch center-to-center spacing between assemblies for ANO-1 and 220 fuel assembly storage locations with a 9.8 inch center-to-center spacing between assemblies for ANO-2. The criticality analysis of the racks is decay time. The TURTLE code is used to determine the reactivity equivalence of assemblies with different initial enrichments and burnups. Direct verification of the codes was not possible because no critical experiments have been done with assemblies having large burnups. Therefore, verification of various aspects of the calculation was undertaken. For example, the ability to calculate the isotopic composition of irradiated fuel was verified by comparing the LEOPARD/CINDER calculation to the measured results of irradiations performed on mixed oxide fuel in Saxton. Similar evidence was used to assess the fission product buildup uncertainty and its reactivity effect as well as the reactivity effect of the transuranium isotopes. The result of these uncertainties in addition to uncertainties due to the method, the nominal eigenvalue, construction and material tolerances, and asymmetric assembly positioning give a total 95/95 uncertainty of 2.48 percent reactivity change.

In order to establish burnup criteria for storage in Region 2 for each unit, a constant storage rack infinite multiplication factor (with minimum post-shutdown fission product inventory) contour is constructed as a function of burnup and initial enrichment using LEOPARD and TURTLE. This contour is based on a high enrichment endpoint of 4.10 weight percent and 36,000 MWD/MTU as shown in Figure 3.8.2 from the proposed ANO-1 TSs and in Figure 3.9.2 from the proposed ANO-2 TSs.

The final multiplication factors for Region 2 are determined using the same KENO IV method used for Region 1 with the conditions determined by the zero burnup intercept point in Figure 3.8.2 for ANO-1 and Figure 3.9.2 for ANO-2. In these cases, the intercept points are at 1.4 weight percent U-235. Therefore, the design mode for Region 2 for ANO-1 & 2 is an unirradiated assembly of 1.4 weight percent initial enrichment. LEOPARD and TURTLE are thus used only to calculate relative reactivities as a function of burnup while the KENO IV Monte Carlo method is used to determine the actual storage rack reactivity. The nominal case multiplication factors are calculated to be 0.8892 for ANO-1 and 0.9068 for ANO-2. Increasing these by the above calculated 95/95 uncertainty of 2.48 percent gives final Region 2 multiplication factors of 0.914 for ANO-1 and 0.9316 for ANO-2 which meet our acceptance criterion of less than or equal to 0.95. Based on our review, we conclude that any number of B&W design 15X15 fuel assemblies with burnups in the non-restricted region of Figure 3.8.2 may be stored in Region 2 of the ANO-1 spent fuel storage racks and that any number of CE design 16X16 fuel assemblies with burnups in the non-restricted region of Figure 3.9.2 may be stored in Region 2 of the ANO-2 spent fuel storage racks.

The multiplication factor for Region 2 is also determined assuming a checkerboard storage configuration with unirradiated fuel assemblies at 4.1 weight percent enrichment. The nominal multiplication factors determined by KENO IV are 0.9068 for ANO-1 and 0.8860 for ANO-2. Adding the 95/95 uncertainties due to the nominal eigenvalue, the method bias, tolerances in thickness and asymmetric assembly position results in values of 0.9402 for ANO-1 and

The proposed TSs governing the criticality aspects of the spent fuel pools for ANO-1 & 2 provide for limits on the initial enrichment of fuel assemblies, burnup limits, required boron concentration, limits on the calculated effective multiplication factors and physical blocks in the vacant spaces adjacent to any fuel assembly in Region 2 in the event a checkerboard storage configuration is deemed necessary.

2.1.5 Conclusions

We conclude that the proposed storage racks meet the requirements of General Design Criterion 62 with regard to 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.

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

We also conclude that the proposed modifications to the ANO-1 & 2 TSs are acceptable to allow operation with the proposed expansion of the spent fuel pools' storage capacities.

2.2 Spent Fuel Pool Cooling and Makeup

2.2.1 Introduction

Each ANO unit has an independent spent fuel pool and spent fuel pool cooling and cleanup system (SFPCS). The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The ANO-1 SFPCS is composed of redundant trains, each train containing a pump and heat exchanger. The redundant trains can be cross-connected so that either pump can provide flow through either or both heat exchangers. The heat exchangers are cooled by the component cooling water system. The ANO-2 SFPCS is a closed loop system consisting of two half capacity pumps and one full capacity heat exchanger. The fuel

System." 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 racks full. The design, therefore, also 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, the licensee performed a thermal-hydraulic analysis. In the event of the complete failure of the spent fuel cooling system, for the maximum normal heat load, there is at least four hours available before boiling occurs. The maximum boiloff rate is 50 to 60 gpm. Each of the two assured seismic Category I borated makeup water sources can be initiated in the required time. Sufficient makeup rates are also available from the seismic Category I service water system, condensate tank or demineralized water supply.

2.2.3 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 SFPCS performance has been reviewed, and we conclude that the pool cooling is adequate. The available makeup systems, their respective makeup rates and the time required before makeup if needed have been reviewed and found acceptable. Based on the above, we conclude that the SFPCSs are acceptable for the proposed expansions.

2.3 Installation of Racks and Load Handling

2.3.1 Description

The proposed spent fuel storage modifications will provide storage locations for 968 fuel assemblies for ANO-1 and 988 fuel assemblies for ANO-2. The spent fuel storage racks are divided into two regions. Region I is designed to accommodate non-irradiated fresh fuel and is sized to permit core offloads. Storage in Region 2 is restricted by burnup and enrichment limits. There is no physical barrier between the two regions. Each fuel assembly will be stored in a double walled storage cell of type 304 stainless steel. The annular spaces between the double walls of the cells contain B C (Boroflex) neutron absorber elements positioned at the rack height corresponding to the active fuel length of the fuel assemblies. The individual storage cells are welded into rack arrays. At ANO-1, the storage racks will have three basic module configurations with dimensions of 10 x 11, 11 x 12 and 11 x 11 feet and weigh 27,500 lbs., 19,500 lbs. and 18,000 lbs. respectively. There will be two 10 x 11 modules, two 11 x 12 modules and four 11 x 11 modules. Similarly, at ANO-2 the storage racks

2.3.3 Conclusion

We have reviewed the described load handling operations and equipment needed for the spent fuel rack modifications. We conclude that the lifting devices and other apparatus used for the handling of the storage racks are acceptable.

2.4 Structural Design

2.4.1 Introduction

Both units at ANO are pressurized water reactors (PWRs). The spent fuel pools are similar right and lefthand arrangements. The pools are elevated with the top of the pools at the fueling floor level, elevation 404 feet. The inside bottom of the pools is at elevation 362 feet. The top of the slab-on-grade is at elevation 335 feet. The approximate inside dimensions of the pools are:

	ANO-1	ANO-2
Depth	42 ft.	42 ft.
Length	44 ft.	32.75 ft.
Width	23 ft.	23 ft.

The pool structures are reinforced concrete with floor thickness of about 5.15 feet and walls of various thickness from 4 to 6 feet. The outside walls of the pools are generally continuous to the foundation mat. These walls support the bottom slab of the pool.

Each pool is lined with a continuous, welded, watertight, 3/16 inch thick stainless steel plate.

The new racks are stainless steel "egg-crate" structures. The 9 cell by 9 cell rack is approximately 16 feet high by 7.4 feet long by 7.4 feet wide. The cells of the egg-crate are fabricated of cold-formed gage thickness material. These cells are supported by a heavy welded base and by a welded structural grid near the top of each rack. The racks are each free standing on the pool floor on four corner leveling pads.

2.4.2 Applicable Codes, Standards and Specifications

Structural material of the racks conform to the ASME Boiler and Pressure Vessel Code, Section III, Subsection NF. Computed stresses were compared with the ASME Code, Section III, Subsection NF. Load combinations and acceptance criteria for racks were compared with the "NRC Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" dated April 14, 1978 and amended January 18, 1979 (hereafter referred to as the "NRC Position").

2.4.6 Seismic Analysis of the Pool Structure

A structural analysis of the reinforced concrete pool structures was conducted by the licensee, and it was found that each pool structure is adequate to withstand the effects of added loads due to the new racks under seismic loads. The analysis consisted of a detailed finite element examination of the pools including thermal and seismic loads as well as other applicable loads. No overstress conditions exist in the pool structures or liners for the proposed installations.

2.4.7 Conclusion

It is concluded that the proposed rack installations will satisfy the requirements of 10 CFR 50, Appendix A, GDC 2, 4, 61 and 62 as applicable to structures, and are therefore acceptable.

2.5 Materials

2.5.1 Materials Description

The proposed spent fuel storage racks have been fabricated of type 304 stainless steel, which is used for all structural components. The storage pool in each of the two units is divided into two regions. Region 1 in each case utilizes Boraflex as a neutron absorber material, attached to the active portion of each fuel assembly cell by a thin wrapper which is welded in place. Placement of the wrapper provides for venting the Boraflex to the pool environment, thereby eliminating potential pressure buildup, for example by radiolysis of entrained water vapor. Depending on criticality requirements, Boraflex is deployed on either all four sides, three sides or two sides of a cell. Region 2 features storage racks consisting of cells assembled in a checkerboard pattern, producing a honeycomb-type structure. Each cell has attached to its outer wall a stainless steel wrapper plate creating a pocket opened at the top and bottom. The spacer pockets are designed to accept poison inserts if future need arises. The 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 III (and therefore, by reference, Section IX).

2.5.2 Chemical Compatibility

The spent fuel pools of ANO-1 and 2 are fabricated of materials that will have good compatibility with the borated water chemistry of the spent fuel pool. The corrosion rate of type 304 stainless steel in this water is sufficiently low to defy our ability to measure it. Since all materials in the pools are stainless steel, no galvanic

We therefore conclude that the compatibility of the materials and coolant used in the spent fuel storage pools 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.

2.6 Spent Fuel Pool Cleanup System

2.6.1 Introduction

The spent fuel pool cleanup systems for ANO-1 and 2 consist of a demineralizer for each unit (mixed bed resin), filters, and associated piping, valves and fittings. The systems are designed to remove corrosion products, fission products, and impurities from the pool water. Pool water purity is monitored by monthly chemical and radiochemical analyses. Demineralizer resin will be replaced on the basis of an increase in differential pressure or when pool water samples show reduced decontamination effectiveness. However, these resins are routinely changed on an annual basis as a preventive measure even though they may not show reduced decontamination effectiveness. The licensee indicated that no change or equipment addition to the spent fuel pool cleanup systems is necessary to maintain pool water quality for the augmented storage facilities.

2.6.2 Evaluation

The spent fuel pool cleanup systems have been reviewed in accordance with Section 9.1.3 of the Standard Review Plan (NUREG-0800, July 1981).

Past experience showed that the greatest increase in radioactivity and impurities in spent fuel pool water occurs only during refueling and spent fuel handling. The refueling frequency, amount of the core to be replaced for each fuel cycle, and frequency of operating the spent fuel pool cleanup systems at ANO-1 and 2 are not expected to increase as a result of expansion of the spent fuel pools. There is no reason to believe that the chemical and radionuclide composition of the spent fuel pool waters will change as a result of the proposed modifications. Past experience also indicated that there is not any significant leakage of fission products from spent fuel stored in pools after the fuel has cooled for several months. Thus, the increased quantity of spent fuel to be stored at ANO-1 and 2 will not contribute significantly to the amount of radioactivity from fission products in the spent fuel pool waters.

The licensee does not expect any significant increase in dose rates due to the buildup of crud along the sides of the pools. If crud buildup eventually becomes a major contributor to pool dose rates, measures will be taken to reduce such dose rates. The purification system for the pools includes filters and demineralizers to remove crud and will be operating during the modifications of the pools.

The licensee has presented four alternative plans for removal and disposal of the old racks. These are (1,2) burial with or without volume reduction; (3) decontaminate to releasable criteria of Regulatory Guide 1.86 and disposal; (4) to have an outside vendor chemically decontaminate and dispose of the intact racks. The disposal methodology will follow ALARA guidelines for each of the alternatives.

The licensee has an ALARA committee, which reviews all work in radiological controlled areas when the estimated collective dose for any job will exceed 1 man-rem. Some of the actions that will be taken by the licensee to assure that occupational doses during each task of the pool modifications will be ALARA are:

1. A health physicist and diving supervisor will be in direct communication with the divers at all times during the re-racking to monitor for excessive exposure by utilizing portable or hand-held radiation monitoring instruments. The dose rates will not be permitted to exceed 1 rem/hr whole body.

2. Personnel monitoring devices will be used by all personnel working in the radiologically-controlled area. Additional monitoring of the underwater divers will be done by multiple whole body TLDs and extremity TLDs.

3. Personnel shall be required to wear appropriate protective clothing as determined by the health physicist to preclude contamination.

4. As the racks are pulled out of the water, they will be washed.

5. Area radiation monitors will be used to alarm on a high radiation signal. Actual dose rates can be read locally and in the control room.

6. A portable filtered water vacuum system will be available to remove loosely deposited contamination from the fuel rack surfaces, pool floor and walls near divers' working areas to reduce the radiation exposure.

7. Contamination control measures will be used to prevent the spread of contamination and to protect personnel from internal exposure from radioactive material

2.9 Radiological Consequences of Rack Module Assembly Drop, Cask Drop and Fuel Handling Accidents

2.9.1 Introduction

We have reviewed the licensee's plans for the expansion of the storage capacity of the spent fuel pools at ANO-1 & 2 regarding radiological consequences of rack module drop, cask drop and fuel handling accidents. The review was conducted according to the guidance of Standard Review Plans 15.7.4 and 17.7.5, and Regulatory Guide 1.25.

2.9.2 Evaluation and Findings

Rack Module Assembly Drop Accident

The overhead cranes in the auxiliary buildings at ANO-1 & 2 will be used for removing the existing rack modules and lowering the new modules into the pools. The licensee has stated in Section 8.1, Rack Modules Assembly Handling Considerations, of the November 5, 1982 submittal that "no loads exceeding 2000 lbs. will be allowed over the fuel assemblies at any time." The TSs for ANO-1 & 2 also prohibit the travel over fuel assemblies in the storage pool of loads in excess of 2000 lbs. Since the weight of a rack module is much greater than 2000 lbs., we conclude that the rack modules will not be carried over the fuel assemblies and that there is reasonable assurance that an accident impacting assemblies in the pools would not occur. The assessment of the radiological consequences of a rack module assembly drop accident is not required.

Fuel Handling Accident

The maximum weight of loads which may be transported over spent fuel in the pool is limited by TSs to that of a single assembly (2000 lbs.). The proposed spent fuel pool modifications do not increase the radiological consequences of fuel handling accidents considered in our SERs of June 1973 (ANO-1) and November 1977 (ANO-2), since this accident would still result in, at most, the release of the gap activity of one fuel assembly due to the limitation on the available impact kinetic energy.

Cask Drop Accident

In the evaluation of the cask drop accident, the licensee states in the November 5, 1982 submittal that the administrative procedures prevent a spent fuel cask from being moved over the spent fuel pools. We conclude that the proposed spent fuel pool modifications do not affect the result of the cask drop accident considered in the SERs.

4. References

1. Arkansas Power and Light Company (AP&L) letter to U. S. Nuclear Regulatory Commission (USNRC) dated November 5, 1982 (OCAN118205).

2. AP&L letter to USNRC dated February 17, 1983 (OCAN028302).

3. AP&L letter to USNRC dated March 3, 1983 (OCAN028310).

4. AP&L letter to USNRC dated March 3, 1983 (OCAN028311).

5. AP&L letter to USNRC dated March 3, 1983 (OCAN038304).

6. AP&L letter to USNRC dated March 7, 1983 (OCAN038307).

7. AP&L letter to USNRC dated March 10, 1983 (OCAN038311).

8. AP&L letter to USNRC dated March 21, 1983 (OCAN038312).

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10. AP&L letter to USNRC dated March 24, 1983 (OCAN038326).

11. AP&L letter to USNRC dated March 28, 1983 (OCAN038331).

12. AP&L letter to USNRC dated March 29, 1983 (OCAN048336).

13. AP&L letter to USNRC dated April 5, 1983 (OCAN048302).

14. AP&L letter to USNRC dated April 7, 1983 (OCAN048305).

15. J. R. Weeks, "Corrosion of Materials in Spent Fuel Storage Pools," BNL-NUREG-23201, July, 1977.

16. J. S. Anderson, Brand Industries, Inc., Reports 748-2-1 (August 1978), 748-10-1 (July 1979), 748-30-1 (August 1979).

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

ENVIRONMENTAL IMPACT APPRAISAL

BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THE MODIFICATION OF THE

SPENT-FUEL STORAGE POOLS

FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 AND 2

DOCKET NOS. 50-313 AND 50-368

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1.0 INTRODUCTION

The storage capacity of the spent fuel pools at Arkansas Nuclear One, Unit 1 (ANO-1) and Unit 2 (ANO-2) is 589 fuel assemblies for ANO-1 and 485 fuel assemblies for ANO-2. These limited storage capacities were in keeping with the expectation generally held in the industry that spent fuel would be kept onsite for a few years and then shipped offsite for reprocessing and recycling of the fuel.

Commercial reprocessing of spent fuel has not developed as had been originally anticipated. In 1975 the Nuclear Regulatory Commission directed the staff to prepare a Generic Environmental Impact Statement (GEIS, the Statement) on spent fuel storage. The Commission directed the staff to analyze alternatives for the handling and storage of spent light water power reactor fuel with particular emphasis on developing long range policy. The Statement was to consider alternative methods of spent fuel storage as well as the possible restriction or termination of the generation of spent fuel through nuclear power plant shutdown.

A Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel (NUREG-0575), Volumes 1-3 (the FGEIS) was issued by the NRC in August 1979. In the FGEIS, consistent with long range policy, the storage of spent fuel 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 onsite fuel storage capacity by modification of the existing spent fuel pools. Since the issuance of the FGEIS, applications for approximately 95 spent fuel pool capacity expansions have been received and 81 have been approved. The remaining 14 are still under review. The finding in each case has been that the environmental impact of such increased storage capacity is negligible. However, since there are variations in storage 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 spent fuel pools, the FGEIS discusses in detail other spent fuel storage alternatives. 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 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, that of shutting down the

1.3 Fuel Reprocessing History

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; in September, 1976, NFS informed the Commission that it was 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 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 contractual arrangements with NFS. On May 4, 1982, the license held by GE for spent fuel storage activities at its Morris operation was renewed for another 20 years; however, GE is also not accepting any additional spent fuel for storage at this facility.

2.0 FACILITY

The principal features of the spent fuel storage and handling at ANO as they relate to the proposed modifications are described here to aid understanding of the evaluations provided in subsequent sections of this EIA.

2.1 Spent Fuel Pool (SFP)

Spent fuel assemblies are intensely radioactive due to their fresh fission product content when initially removed from the core; also, they have a high thermal output. The SFP is designed for storage of these assemblies to allow for radioactive and thermal decay prior to shipping them to a reprocessing facility. Space permitting, the assemblies may be stored for longer periods, allowing continued fission product decay and thermal cooling. The ANO-1 SFP is approximately 23 ft. wide x 44 ft. long x 42 ft. deep and the ANO-2 SFP is approximately 23 ft. wide x 32 3/4 ft. long x 42 ft. deep. The SFP structures are reinforced concrete lined with a continuous, watertight stainless steel plate.

2.2 Spent Fuel Pool Cooling and Cleanup System

Each ANO Unit has an independent spent fuel pool and spent fuel pool cooling and cleanup system. The spent fuel pool cooling and cleanup system is designed to remove the decay heat generated by the stored spent fuel assemblies and to maintain the water quality and clarity of the pool water. The ANO-1 spent fuel pool cooling system is composed of redundant trains,

2) Additional storage will not result in measurable increase in non-radiological chemical waste discharges to the receiving water. The licensee does not propose any change in chemical usage or change to the NPDES permit.

3) Additional SFP heat output will not cause measurable thermal effects to the receiving water. The increase in the heat load due to this modification is less than one tenth percent of the present SFP design heat load.

We conclude, based on the above evaluations, that the SFP modifications will not result in non-radiological environmental effects significantly greater or different from those already reviewed and analyzed in the FES for ANO-1 and ANO-2.

4.0 RADIOLOGICAL ENVIRONMENTAL IMPACTS OF PROPOSED ACTION

4.1 Introduction

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

During the storage of the spent fuel under water, both volatile and non-volatile 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 SFPs 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, however, that there is little radionuclide leakage from spent fuel stored in pools after the fuel has cooled for several months. The predominance of radionuclides in the SFP water appear to be radionuclides that were present in the reactor coolant system prior to refueling (which becomes mixed with water in the SFP during refueling operations) or crud dislodged from the surface of the spent fuel during transfer from the reactor core to the SFP.

During and after refueling, the SFP purification system reduces the radioactivity concentrations considerably. It is theorized that most failed fuel contains small, pinhole-like perforations in the fuel cladding

Most airborne releases of tritium and iodine result from evaporation of reactor coolant, which contains tritium and iodine in higher concentrations than the pool water. Therefore, even if there were a higher evaporation rate from the spent fuel pool, the increase in tritium and iodine released from the plant as a result of the increased stored spent fuel would be small compared to the amount normally released from the plant and that which was previously evaluated in the FESs. In addition, the station radiological effluent Technical Specifications limit the total releases of gaseous activity.

Based on the foregoing considerations, implementation of the proposed increased spent fuel storage capability will not result in significantly increased amounts of radioactivity being released to the atmosphere.

4.3 Solid Radioactive Wastes

The concentration of radionuclides in the pool water is controlled by the filters and the demineralizer and by decay of short-lived isotopes. The activity is highest during refueling operations when reactor coolant water is introduced into the pool, and decreases as the pool water is processed through the filters and demineralizer. The increase of radioactivity, if any, due to the proposed modification, should be minor because of the capability of the cleanup system to continuously remove radioactivity in the SFP water to acceptable levels.

The licensee does not expect any significant increase in the amount of solid waste generated from the spent fuel pool cleanup systems due to the proposed modification. While we 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 two resin beds (104 cubic feet wet) and two spent filter cartridges (20 cubic feet wet) per year from both units due to the increased operation of the spent fuel pool cleanup systems. The annual average volume of solid wastes shipped offsite for burial from a typical PWR with deep bed condensate demineralizer system is approximately 18,800 cubic feet. If the storage of additional spent fuel does increase the amount of solid waste from the SFP cleanup systems by about 124 cubic feet (250 cubic feet solidified) per year from both units, the increase in total waste volume shipped from Arkansas Nuclear One would be less than 1% and would not have any significant additional environmental impact.

The present spent fuel racks to be removed from the SFPs because of the proposed modification are contaminated and may be disposed of as low level solid waste. We have estimated that approximately 14,000 cubic feet of solid radwaste will be removed from the plant because of the proposed modifications. Averaged over the lifetime of the plant this would increase the total waste volume shipped from the facility by less than 2%. This will not have any significant additional environmental impact.

5.0 ENVIRONMENTAL IMPACTS OF POSTULATED ACCIDENTS

5.1 Rack Module Assembly Drop Accident

The overhead cranes in the auxiliary building at ANO will be used for removing the existing rack modules and lowering the new modules into the pool. The licensee has stated in Section 8.1, Rack Modules Assembly Handling Considerations, of the November 5, 1982 submittal that "no loads exceeding 2000 lbs will be allowed over the fuel assemblies at any time." The Technical Specifications for ANO-1 and ANO-2 also prohibit the travel over fuel assemblies in the storage pool of loads in excess of 2000 lbs. Since the weight of a rack module is much greater than 2000 lbs, we conclude that the rack modules will not be carried over the fuel assemblies and that there is reasonable assurance that an accident impacting assemblies in the pool would not occur. Therefore, the assessment of the radiological consequences of a rack module assembly drop accident is not required.

5.2 Fuel Handling Accident

The maximum weight of loads which may be transported over spent fuel in the pool is limited by Technical Specifications to that of a single assembly (≃2000 lbs). The proposed spent fuel pool modification does not increase the radiological consequences of fuel handling accidents considered in the staff Safety Evaluation report of June 1973 (ANO-1) and November 1977 (ANO-2), since this accident would still result in, at most, the release of the gap activity of one fuel assembly due to the limitation on the available impact kinetic energy. In the evaluation of the cask drop accident, the licensee states in the November 5, 1982 submittal that the administrative procedures prevent the spent fuel cask from being moved over the spent fuel pool. The staff concludes that the proposed spent fuel pool modification does not affect the result of the cask drop accident considered in the staff's Safety Evaluation Reports.

5.3 Conclusion

Based upon the above evaluation, the staff concludes that the likelihood of a rack module assembly drop accident is sufficiently small - since the rack module assembly will not be allowed over the fuel at any time - that this accident need not be considered. Also, a fuel handling accident involving a dropped assembly or cask would not be expected to result in radionuclide releases leaking to offsite radiological consequences exceeding those of the fuel handling accident evaluated in the staff Safety Evaluation Reports of June 1973 (ANO-1) and November 1977 (ANO-2); that is, doses would be well within 10 CFR Part 100 values. We conclude therefore, that the proposed modifications are acceptable.

8.0 REFERENCES

1. Arkansas Power and Light Company (AP&L) letter to U. S. Nuclear Regulatory Commission (USNRC) dated November 5, 1982 (OCAN118205).
2. AP&L letter to USNRC dated February 17, 1983 (OCAN028302).
3. AP&L letter to USNRC dated March 3, 1983 (OCAN028310).
4. AP&L letter to USNRC dated March 3, 1983 (OCAN028311).
5. AP&L letter to USNRC dated March 3, 1983 (OCAN038304).
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UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKETS NOS. 50-313 AND 50-368

ARKANSAS POWER AND LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

AND NEGATIVE DECLARATION

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendments Nos. 76 and 43 to Facility Operating Licenses Nos. DPR-51 and NPF-6, issued to Arkansas Power and Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Units Nos. 1 and 2, respectively (ANO-1&2), located in Pope County, Arkansas. The amendments are effective as of the date of issuance.

The amendments allow an increase in the spent fuel storage capacity from 589 spaces to 968 spaces for ANO-1 and from 485 spaces to 988 spaces for ANO-2 through the use of high density storage racks.

The application for the amendments 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 amendments. Notice of Consideration of Issuance of Amendments to Facility Operating Licenses in connection with this action was published in the FEDERAL REGISTER on December 22, 1982 (47 FR 57154).

No request for a hearing or petition for leave to intervene was filed following notice of the proposed action. The Commission has prepared an