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Docket No, 50-313

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

AUG 16 1976

Gentlemen:

In response to your request dated March 3, 1976, the Commission has issued the enclosed Amendment No. 13 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One - Unit No. 1 (ANO-1) facility.

This amendment revises the provisions in the Technical Specifications relating to limiting conditions for operation and surveillance requirements for the facility emergency pond. In addition, it corrects a reference error in Specification 4.4.1.3.

Copies of the related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,
Original Signed by:
Dennis L. Ziemann

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

Enclosures:
1. Amendment No. 13 to License No. DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

V. Stello OK'd issuance of this (Thru Pat King) 8/13/76 at 3:15

OFFICE	OR:ORB #2	OR:ORB #2	OELD	OR:ORB #2	DSS&EA:HMB	
SURNAME	RMDiggs	WEConverse:ro	Davis LCHULMAN 8/9/76	DLZiemann		
DATE	7/19/76	7/21/76	8/6/76	8/16/76	8/9/76	

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

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

cc w/enclosures:
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Mr. William Cavanaugh, III
Manager, Nuclear Services
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72201

Arkansas Polytechnic College
Russellville, Arkansas 72801

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

cc w/enclosures and cy of AP&L
filing dtd. 3/3/76:
Mr. E. F. Wilson
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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 13
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 March 3, 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.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 13

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace existing pages 66a, 83 and 110a of the Technical Specifications contained in Appendix A to the license with the attached revised pages bearing the same numbers. The changed areas on the revised pages are reflected by a marginal line. Also, pages 66b and 110b are enclosed as a matter of convenience in updating the Technical Specifications. There are no changes on these pages.

3.11 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To assure the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

Specification

3.11.1 The reactor shall not be taken critical unless the emergency cooling pond is operable with:

1. An average water depth $\geq$ 3'0", and

2. An average water temperature $\leq$ 105F.

3.11.2 With the requirements of Specification 3.11.1 not satisfied, be in the hot shutdown condition within 6 hours and in the cold shutdown condition within the following 30 hours.

Bases

The requirements of Specification 3.11.1 provide for sufficient water inventory in the emergency cooling pond to mitigate within acceptable limits the effects of a DBA with a concurrent failure of the Dardanelle Reservoir. The minimum water depth takes into account (1) water loss from evaporation due to heat load and climatological conditions, (2) pond bottom irregularities and (3) suction pipe level at the pond. The minimum water depth and maximum water temperature correspond to conservative values used to determine pond response and assure the applicability of the calculational model for pond evaporation.

3.12 MISCELLANEOUS RADIOACTIVE MATERIALS SOURCES

Applicability

Applies to byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

3.12.1 The source leakage test performed pursuant to Specification 4.14 shall be capable of detecting the presence of 0.005 µCi of radioactive material on the test sample. If the test reveals the presence of 0.005 µCi or more of removable contamination, it shall immediately be withdrawn from use, decontaminated, and repaired, or be disposed of in accordance with Commission regulations. Sealed sources are exempt from such leak tests when the source contains 100 µCi or less of beta and/or gamma emitting material or 10 µCi or less of alpha emitting material.

3.12.2 A complete inventory of licensed radioactive materials in possession shall be maintained current at all times.

4.4.1.2.5 Test Frequency

Local leak detection tests shall be performed at a frequency of at least each refueling period, but in no case at intervals greater than two years except that:

(a) The equipment hatch and fuel transfer tube seals shall be additionally tested after each opening.

(b) If a personnel hatch or emergency hatch door is opened when reactor building integrity is required, the affected door seal shall be tested. In addition, a pressure test shall be performed on the personnel and emergency hatches every six months.

4.4.1.3 Reactor Building Modifications

Any major modification or replacement of components affecting the reactor building integrity shall be followed by either an integrated leak rate test or a local leak test, as appropriate, and shall meet the acceptance criteria specified in 4.4.1.1 and 4.4.1.2 respectively.

4.4.1.4 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The latter valves shall be tested during each refueling period.

4.4.1.5 Visual Inspection

A visual examination of the accessible interior and exterior surfaces of the reactor building structure and its components shall be performed during each refueling shutdown and prior to any integrated leak test, to uncover any evidence of deterioration which may affect either the reactor building's structural integrity or leak-tightness. The discovery of any significant deterioration shall be accompanied by corrective actions in accord with acceptable procedures, nondestructive tests, and inspections, and local testing where practical prior to the conduct of any integrated leak test. Such repairs shall be reported as part of the test results.

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285 F. Prior to initial operation, the reactor building will be strength tested at 115% of design pressure and leak rate tested at the design pressure. The reactor building will also be leak tested prior to initial operation at not less than 50% of

4.13 EMERGENCY COOLING POND

Applicability

Applies to the emergency cooling pond.

Objective

To verify the availability of a sufficient supply of cooling water inventory in the emergency cooling pond.

Specification

4.13.1 The emergency cooling pond shall be determined operable by verifying:

1. At least once per 24 hours that the average water depth is $\geq$ 3'0".
2. At least once per 24 hours during the period from June 1 to September 30 that the average water temperature at the point of discharge from the pond is $\leq$ 105F.

4.13.2 Soundings shall be made annually of the emergency cooling pond bottom to ensure that the required volume of water is available.

Bases

The requirements of Specification 4.13 provide for verification of a surficient water inventory in the emergency cooling pond to handle a DBA with a concurrent failure of the Dardanelle Reservoir. This specification ensures that Specification 3.11.1 is met. Monitoring temperature only during the period June 1 to September 30 of each year ensures that, during the hot summer months, the pond temperature limit is not exceeded. During other periods of the year the pond temperature will not have the potential to reach temperatures near 105F.

4.14 RADIOACTIVE MATERIALS SOURCES SURVEILLANCE

Applicability

Applies to leakage testing of byproduct, source, and special nuclear radioactive material sources.

Objective

To assure that leakage from byproduct, source, and special nuclear radioactive material sources does not exceed allowable limits.

Specification

Tests for leakage and/or contamination shall be performed by the licensee or by other persons specifically authorized by the Commission or an agreement State, as follows:

1. Each sealed source, except startup sources subject to core flux, containing radioactive material, other than Hydrogen 3, with a half-life greater than 30 days and in any form other than gas shall be tested for leakage and/or contamination at intervals not to exceed six months.

2. The periodic leak test required does not apply to sealed sources that are stored and not being used. The sources excepted from this test shall be tested for leakage prior to any use of transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate from a transferor indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to and following any repair or maintenance and before being subjected to core flux.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 13 TO FACILITY LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT NO. 1

DOCKET NO. 50-313

INTRODUCTION

By letter dated March 3, 1976, Arkansas Power & Light Company (AP&L) requested changes to the Technical Specifications appended to Facility License No. DPR-51 for the Arkansas Nuclear One - Unit No. 1 facility. The proposed changes involve revision of the limiting conditions for operation and surveillance requirements for the facility emergency pond (Specification 3.11 and 4.13). This safety evaluation considers the proposed technical specifications and the correction of a reference error in Specification 4.4.1.3.

DISCUSSION

The existing limiting conditions for operation for the emergency pond require a pond depth "equal to or greater than elevation 344 feet 0 inches, corresponding to a 3-foot pond depth." The corresponding surveillance requirements require a daily check of the pond elevation and annual soundings of the pond bottom to determine that an adequate volume of water is available for cooling. By letter dated September 18, 1975, AP&L reported an occurrence in which the annual soundings revealed that a pond elevation of 344 feet 0 inches corresponded to a pond depth of less than 3 feet as the result of bottom swelling.

The licensee's March 3, 1976 letter requested a change to the emergency pond limiting conditions for operation (LCO) to require only the 3 feet minimum depth. In addition, the proposed LCO would permit a maximum pond temperature of 105°F. Also, provisions are made in the Specification to place the plant in cold shutdown within 36 hours if the pond level and temperature requirements are not met. A corresponding surveillance requirement would require daily measurements of pond depth and pond temperature (temperature from June 1 through September 30 only). The annual pond sounding specification would be retained.

AP&L has proposed these specifications to eliminate possible confusion arising from the wording of the existing Specifications 3.11 and 4.13 and to assure that sufficient pond cooling water is available when needed.

EVALUATION

We have evaluated the licensee's proposed changes to Technical Specifications 3.11 and 4.13 related to the emergency cooling pond. The salient safety aspects of this review involve (1) the proposed method to assure and verify the availability of sufficient water in the pond and (2) the calculation of the maximum allowable pond ambient water temperature that would meet the cooling requirements during the design basis accident (DBA).

Because of pond bottom swelling encountered at the facility, the minimum pond depth (3 feet) specified in the ANO-1 Final Safety Analysis Report (FSAR) may not be available if the pond surface is at the existing technical specification limit of 344' 0". In lieu of changing this elevation limit each time a change in pond bottom elevation is identified, the licensee proposed measuring the average pond depth to assure a sufficient quantity of water is available. The annual measurement of pond water volume by sounding techniques would be retained as a backup to the proposed daily depth measurement. We have concluded that these provisions would assure that a sufficient quantity of pond water would be available when needed.

AP&L also proposes to revise the specifications to allow a maximum ambient water temperature in the pond (that is, the water temperature that the pond would attain without any heat input from ANO-1) of 105°F in lieu of the 94°F specified in the ANO-1 FSAR. The licensee performed a transient heat analysis of the pond using the calculational model developed for ANO-Unit 2 to assure that the return water temperature to the service water system would not exceed 120°F, the maximum allowable. Pertinent information used in the analysis were: initial pond depths of 3, 4 and 5 feet, service water flow of 5250 gpm (one service water loop) or 10,500 gpm (two service water loops), pond surface area of 14 acres, DBA heat load for Unit 1 only, and maximum allowable return water temperature of 120°F. We performed an independent analysis of the response of the pond to the above conditions and concluded that for Unit 1 only the water temperature would not exceed 120°F under severe (summer) meteorological conditions. We therefore conclude that the maximum allowable ambient pond water temperature of 105°F is acceptable.

The proposed LCO requiring the plant to be placed in the cold shutdown conditions within 36 hours if the pond depth and/or pond temperature limits cannot be met is consistent with NRC requirements for emergency cooling ponds.

On the basis of the above considerations, we conclude that the proposed Specifications 3.11 and 4.13 would assure that sufficient plant cooling would be available for ANO-1 in the event of a DBA. The proposed Specifications 3.11 and 4.13 are therefore acceptable.

We note, however, that the maximum allowable return temperature to the service water system for ANO-Unit 2 (presently in the operating license review stage) is 129.6°F. An independent analysis of the emergency cooling pond when both units are operating indicates that the maximum water temperature will be significantly in excess of 120°F, even with an ambient water temperature of 94°F. This disparity of maximum allowable return water temperatures should be resolved prior to ANO-Unit 2 operation.

Specification 4.4.1.3 references Specification 4.3.1.1 and 4.3.1.2 which do not exist. These references should be corrected to Specifications 4.4.1.1 and 4.4.1.2. This change has been discussed with the licensee's staff and they agree with the change.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: AUG 1 6 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 13 to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised Technical Specifications for operation of the Arkansas Nuclear One - Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment (1) revised the provisions in the Technical Specifications relating to limiting conditions for operation and surveillance requirements for the facility emergency pond and (2) corrected a reference error in Specification 4.4.1.3.

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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated March 3, 1976, (2) Amendment No. 13 to License No. DPR-51, and (3) the Commission's related Safety Evaluation. 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 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

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by:
Dennis L. Ziemann

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

OFFICE ➤						
SURNAME ➤						
DATE ➤						

JUN 11 1976

Docket No. 50-313

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:

By letter dated April 20, 1976, you requested an exemption from the requirements of 10 CFR Part 50, Appendix H, Section II.C.2, to permit the operation of Arkansas Nuclear One - Unit No. 1 (ANO-1) during the remainder of Cycle 1 operation with the reactor vessel surveillance specimens and holder tubes removed from the reactor vessel.

As required by 10 CFR Part 50, Appendix H, and discussed in your letter of April 20, 1976, the surveillance specimens contained within the surveillance capsules receive a higher neutron flux than the reactor vessel inner surface. As noted in the enclosed Safety-Evaluation, this difference results in the surveillance specimens being irradiated at a rate 2.4 times higher than the vessel. At this rate, the specimens would continue to lead the vessel in accumulated neutron flux exposure even if removed during the remainder of Cycle 1.

We have therefore concluded that if the reactor vessel surveillance specimens are removed from ANO-1 for the remainder of Cycle 1 operation and reinstalled prior to Cycle 2 operation, the reactor vessel surveillance program would continue to fulfill the purpose of 10 CFR Part 50, Appendix H.

We have also concluded that an exemption from the requirements of 10 CFR Part 50, Appendix H, for the remainder of Cycle 1 operation will not endanger life or property or the common defense and security and is in the public interest. Thereby, the actions requested in your letter of April 20, 1976, are approved.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

JUN 11 1976

In response to your letters of May 19, June 4 and June 10, 1976, the Commission has issued the enclosed Amendment No. 12 to Facility Operating License No. DPR-51 for the ANO-1. This amendment (1) provides for the removal of surveillance capsules during the remainder of Cycle 1 operation, and the revision of Technical Specification 4.2.7 to conform with Appendix H to 10 CFR Part 50, (2) alters the requirements of Amendment No. 10 to your license to permit reactor operation until the end of Cycle 1 with available spare charcoal filters installed in the Hydrogen Purge System and (3) increases the frequency of gross radioiodine determination and establishes reporting requirements for significant changes in the gross radioiodine concentration.

Copies of the Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Original Signed by
V. Stello

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 12 to DPR-51
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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TBAbernathy
JRBuchanan
DRoss

NRR:D/DIR
EGCase
6/11/76

DOR:AD/OT
DEisenhut
6/11/76

Concur on condition that Fed. Register notice be revised to since notice of Appendix H exemption also done
Davis

OFFICE	DOR:ORB #2	DOR:ORB #2	DOR:ORB #2	DOR:AD/OR	OELD	DOR:DIR
SURNAME	WEConverse:rc	RMDiggs	DLZiemann	KRGoller	Grossman	VStello
DATE	6/11/76	6/11/76	6/11/76	6/11/76	6/11/76	6/11/76

cc w/enclosures:
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Little Rock, Arkansas 72201

Mr. William Cavanaugh, III
Manager, Nuclear Services
Arkansas Power & Light Company
Post Office Box 551
Little Rock, Arkansas 72201

Arkansas Polytechnic College
Russellville, Arkansas 72801

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

cc w/enclosures and cy of Arkansas' filings dtd. 4/20/76; 5/19/76 and 6/4/76:
Mr. E. F. Wilson
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 NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 12
License No. DPR-51

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

   A. The applications for amendment by Arkansas Power & Light Company (the licensee) dated May 19, June 4 and June 10, 1976, comply 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.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment.

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

FOR THE NUCLEAR REGULATORY COMMISSION

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: JUN 11 1976

OFFICE ➤						
SURNAME ➤						
DATE ➤						

Table 4.1-3

MINIMUM SAMPLING AND ANALYSIS FREQUENCY

Item	Test	Frequency
1. Reactor Coolant Samples	a. Gamma Isotopic Analysis	a. Bi-weekly (7)
	b. Gross Activity Determination	b. 3 times/week and at least every third day (1)(6)(7)
	c. Gross Radioiodine Determination	c. Weekly (3)(6)(7)
	d. Dissolved Gases	d. Weekly (7)(11)
	e. Chemistry (Cl, F, and O_2)	e. 3 times/week (8)
	f. Boron Concentration	f. 3 times/week
	g. Radiochemical Analysis for $\bar{E}$ Determination (2)(4)	g. Monthly (7)
2. Borated Water Storage Tank Water Sample	Boron Concentration	Weekly and after each makeup
3. Core Flooding Tank Sample	Boron Concentration	Monthly and after each makeup
4. Spent Fuel Pool Water Sample	Boron Concentration	Monthly and after each makeup (9)
5. Secondary Coolant Samples	a. Gross Radioiodine Concentration	a. Weekly (5)(7)(10)
	b. Isotopic Radioiodine Concentration (4)	b. Monthly (7)(10)
6. Sodium Hydroxide Tank Sample	Sodium Hydroxide Concentration	Quarterly and after each makeup
7. Sodium Thiosulfate Tank Sample	Sodium Thiosulfate Concentration	Quarterly and after each makeup

Notes:

(1) A gross radioactivity analysis shall consist of the quantitative measurement of the total radioactivity of the primary coolant in units of μCi/gm. The total primary coolant activity shall be the sum of the degassed beta-gamma activity and the total of all identified gaseous activities 15 minutes after the primary system is sampled. Whenever the gross radioactivity concentration exceeds 10% of the limit specified in the Specification 3.1.4.1 or increases by 10 μCi/gm from the previous measured level, the frequency of sampling and analyzing shall be increased to a minimum of once/day until a steady activity level is established.

(2) A radiochemical analysis shall consist of the quantitative measurement of the activity for each radionuclide which is identified in the primary coolant 15 minutes after the primary system is sampled. The activities for the individual isotopes shall be used in the determination of $\bar{E}$. A radiochemical analysis and calculation of $\bar{E}$ and iodine isotopic activity shall be performed if the measured gross activity changes by more than 10 μCi/gm from the previous measured level. The gamma energy per disintegration for those radioisotopes determined to be present shall be as given in "Table of Isotopes" (1967) and beta energy per disintegration shall be as given in USNRDL-TR-802 (Part II) or other references using the equivalent values for the radioisotopes.

(3) In addition to the weekly measurement, the radioiodine concentration shall be determined if the measured gross radioactivity concentration changes by more than 10 μCi/gm from the previous measured level.

(4) Iodine isotopic activities shall be weighted to give I-131 dose equivalent activity.

(5) In addition to the weekly measurement, the radioiodine concentration shall be determined if there are indications that the primary to secondary coolant leakage rate has increased by a factor of 2.

(6) Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 1 percent but less than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken within 24 hours of any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by the above.

Whenever the steady state radioiodine or gross radioactivity concentration of prior operation is greater than 10 percent of Specification 3.1.4.1, a sample of reactor coolant shall be taken prior to any reactor criticality and analyzed for radioactive iodines of I-131 through I-135 and gross radioactivity as well as the coolant sample and analyses required by above.

(7) Not required when plant is in the cold shutdown condition or refueling shutdown condition.

(8) O_2 analysis is not required when plant is in the cold shutdown condition or refueling shutdown condition.

(9) Required only when fuel is in the pool and prior to transferring fuel to the pool.

(10) Not required when not generating steam in the steam generators.

(11) The following shall be required only during Cycle 1 operation after 340 effective full power days:

a. Gross radioiodine shall be determined at least three times per week during power operation.

b. If the steady state gross radioiodine concentration increases by a factor of ten or more, the NRC shall be promptly notified with a written followup per Specification 6.12.3.1.

IS-261 Item	Component	Exception
6.4	Bolting 2ø	Not Applicable
6.6	Integrally Welded Valve Supports	Not Applicable

4.2.3 The structural integrity of the reactor coolant system boundary shall be maintained at the level required by the original acceptance standards throughout the life of the station. Any evidence, as a result of the tests outlined in Table IS-261 of Section XI of the code, that defects have developed or grown, shall be investigated.

4.2.4 To assure the structural integrity of the reactor internals throughout the life of the unit, the two sets of main internals bolts (connecting the core barrel to the core support shield and to the lower grid cylinder) shall remain in place and under tension. This will be verified by visual inspection to determine that the welded bolt locking caps remain in place. All locking caps will be inspected after hot functional testing and whenever the internals are removed from the vessel during a refueling or maintenance shutdown. The core barrel to core support shield caps will be inspected each refueling shutdown.

4.2.5 Sufficient records of each inspection shall be kept to allow comparison and evaluation of future inspections.

4.2.6 Complete surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that within a 10 year period after start-up all four reactor coolant pump flywheels will be examined.

4.2.7 Reactor vessel specimens shall be removed and examined, to determine changes in material properties, at specimen exposure (E>1Mey) equivalent to 3, 9.5, 16 and 22.5 Effective Full Power Years (EFPY) of operation. This withdrawal schedule may be modified to coincide with those refueling outages or plant shutdowns, when the reactor head is removed, most closely approaching the withdrawal schedule. Results of these examinations shall be used to update Technical Specification 3.1.2. Specimens not subjected to destructive testing after the first 0.93 EFPY of Cycle 1 may be removed and stored during the remainder of Cycle 1, but shall be re-installed prior to Cycle 2.

Bases

The surveillance program has been developed to comply with Section XI of the ASME Boiler and Pressure Vessel Code Inservice Inspection of Nuclear Reactor Coolant Systems, 1971, including 1971 Winter Addenda edition.

The number of reactor vessel specimens and the frequencies for removing and testing these specimens are provided to assure compliance with the requirements of Appendix H to 10 CFR Part 50.

4.3 TESTING FOLLOWING OPENING OF SYSTEM

Applicability

Applies to test requirements for Reactor Coolant System integrity.

Objective

To assure Reactor Coolant System integrity prior to return to criticality following normal opening, modification, or repair.

Specification

4.3.1 When Reactor Coolant System repairs or modifications have been made, these repairs or modifications shall be inspected and tested to meet all applicable code requirements prior to the reactor being made critical.

4.3.2 Following any opening of the Reactor Coolant System, it shall be leak tested at not less than 2285 psig prior to the reactor being made critical.

4.3.3 The limitations of Specification 3.1.2 shall apply.

Bases

Repairs or modifications made to the Reactor Coolant System are inspectable and testable under applicable codes, such as B 31.7, and ASME Boiler and Pressure Vessel Code, Section IX, IS-400.

For normal opening, the integrity of the Reactor Coolant System, in terms of strength, is unchanged. If the system does not leak at 2285 psig (operating pressure +100 psi; ±50 psi is normal system pressure fluctuation), it will be leak tight during normal operation.(1)

REFERENCES

FSAR, Section 4

4.12 HYDROGEN PURGE SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the hydrogen purge system.

Objective

To verify an acceptable level of efficiency and operability of the hydrogen purge system.

Specification

4.12.1 At least once per refueling period (not to exceed 18 months) the following conditions shall be demonstrated:

a. The pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 16 inches of water at system design flow rate (± 10%).

b. Each system inlet heater unit operates at rated power.

4.12.2.a. The tests and sample analysis of Specification 3.14.1.a,b, & c. shall be performed initially* and at least once per refueling period (not to exceed 18 months) or after every 720 hours of system operation and following significant painting, fire or chemical release in any ventilation zone communicating with the system.

b. Cold DOP testing shall also be performed after each complete or partial replacement of a HEPA filter bank or after any structural maintenance on the system housing.

c. Halogenated hydrocarbon testing shall also be performed after each complete or partial replacement of a charcoal adsorber bank or after any structural maintenance on the system housing.

4.12.3 Each circuit shall be operated at least 10 hours each month.

4.12.4 Hydrogen concentration instruments shall be calibrated at least once per refueling period (not to exceed 18 months) with proper consideration to moisture effect.

Bases

Since the hydrogen purge system is not normally operated, a periodic test is required to show that the system is available for hydrogen control following an accident. During this test the system will be inspected for such things as water, oil, or other foreign material, gasket deterioration in the HEPA units, and unusual or excessive noise or vibration when the fan motor is running.

*Initial tests shall be performed within 9 months of the date of issuance of Amendment 10 to License No. DPR-51.

ATTACHMENT TO LICENSE AMENDMENT NO. 12

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace existing pages 74, 75, 77 and 109b of the Technical Specifications contained in Appendix A to the license with the attached revised pages bearing the same numbers and additional page 75a. The changed areas on the revised pages are identified by a marginal line. Also, page 78 is enclosed as a matter of convenience in updating the Technical Specifications. There are no changes on this page.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 12 TO FACILITY LICENSE NO. DPR-51

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated April 20, 1976, Arkansas Power & Light Company (the licensee) requested an exemption from the requirements of 10 CFR Part 50, Appendix H, Section II.C.2 to permit operation of the Arkansas Nuclear One - Unit No. 1 (ANO-1) facility for the remainder of Cycle 1 operation with the reactor vessel surveillance specimens removed from the vessel. By letters dated May 19, June 4, and June 10, 1976, the licensee requested changes to the Technical Specifications appended to Facility License No. DPR-51 for ANO-1. The changes would:

1. Revise Specification 4.2.7 to comply with the requirements of Appendix H to 10 CFR Part 50 and to permit operation for the remainder of Cycle 1 with the reactor vessel surveillance specimens removed.

2. Revise Specifications 3.14 and 4.12 to permit reactor operation using charcoal filters in the Hydrogen Purge System that do not meet the requirements of Amendment No. 10 to the facility license issued February 18, 1976, and

3. Revise Table 4.1-3, "Minimum Sampling and Analysis Frequency", to require an increased frequency of gross radioiodine determinations during the remainder of Cycle 1 operation and reporting of significant increase in gross radioiodine concentration.

The licensee also submitted a report by letter dated June 1, 1976, describing the surveillance specimen holder tube (SSHT) failure at ANO-1, and subsequent inspections, cleanup, and repairs.

This safety evaluation considers the:

1. Proposed exemption to Appendix H, 10 CFR Part 50, Section II.C.2,

2. Proposed Technical Specification changes described above, and

3. Adequacy of the inspection, cleanup, and repair of the SSHTs.

2.0 DISCUSSION

The Arkansas Nuclear One - Unit No. 1 design includes three reactor vessel surveillance specimen holder tubes located adjacent to the reactor inside vessel wall. Each holder tube contains two surveillance capsules which hold the specimens to be irradiated in accordance with the requirements of the reactor vessel material surveillance program as described in Appendix H to 10 CFR Part 50. The purpose of the surveillance program is to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel regions resulting from their exposure to neutron irradiation and the thermal environment.

The SSHTs (3 inch diameter, 120 mil thickness, type 304 stainless steel) are mounted to the reactor thermal shield by pintles which support, position, and lock the holder tubes in place. An outer shroud tube (3 1/2 inch diameter, 120 mil thickness, type 304 stainless steel) surrounds the upper portion of the SSHT to protect the tube from the impact of coolant flow. Two surveillance specimens are held at the bottom of each SSHT by a spring loaded push rod assembly. Spacers are attached to the push rod at four axial locations to position the push rod in the holder tube. During normal operation, reactor downcomer water flows up the SSHT to cool the surveillance specimens.

Because of observed SSHT failures at other B&W plants, the licensee shut down the ANO-1 facility on March 19, 1976 to inspect the SSHTs. The inspection, performed on March 25 and 26, 1976, revealed the following damage to the ANO-1 SSHT. One holder tube was severed at the second spacer from the top but the holder tube itself was retained in the shroud tube. The lower section of this SSHT was intact and in place as were the associated surveillance capsules. The other two SSHTs were found to be completely severed. Portions of these holder tubes and the associated holddown mechanisms (push rod, spacers, and holddown springs) had fallen to the bottom of the reactor vessel. The lower portion of both of these SSHTs and the associated surveillance capsules also were found to be intact and in place. Subsequent inspections revealed that wear damage had also occurred at the inner surfaces of the shroud tubes of all three SSHTs.

To preclude the possibility of further SSHT damage occurring during the remainder of Cycle 1 operation, the surveillance specimens, holder tubes, push-rod and holddown assemblies, and portions of the shroud tube will be removed prior to continuation of Cycle 1 operation. Engineering of new holder tube and holddown assembly design modifications and material procurement will be completed during the remainder of Cycle 1 operation to allow installation of the revised holder tubes prior to the start of Cycle 2 operation.

The licensee has requested a change to Technical Specification 4.2.7 to update this specification to comply with 10 CFR Part 50, Appendix H, Section II.C.3.C and to reflect removal of the reactor vessel surveillance specimens during the remainder of Cycle 1 operation. The existing Specification 4.2.7 is based on a withdrawal schedule recommended in ASTM-E-185-70 and was adopted prior to publication of Appendix H to 10 CFR Part 50.

The licensee also has requested a waiver of the requirements of Technical Specifications 3.14.1.b and 4.12.2.a (involving Hydrogen Purge System charcoal filters) to permit reactor operation with filters that do not fully meet the requirements of the existing Technical Specifications.

The licensee's June 10, 1976 request for a change to the sampling frequency and reporting requirements for the gross radioiodine determination (Table 4.1-3) was made to meet NRC staff requirements.

3.0 EVALUATION

3.1 Exemption to Appendix H, 10 CFR Part 50

As required by Paragraph II.C.2 of Appendix H to 10 CFR Part 50, the surveillance capsules of ANO-1 are positioned during reactor operation so that the neutron flux received by the specimens is at least as high, but not more than three times as high as that received by the vessel inner surface. A recent calculation by Babcock and Wilcox Company[1/], applicable to ANO-1 and acceptable to the staff, indicates that the neutron flux is 2.4 times greater

1/ BAW - 10100A, February 1975.

at the specimen location than at the reactor vessel wall at 1/4 wall thickness (1/4 t). To date, Cycle 1 operation of ANO-1 accumulated approximately 0.93 EFPY* of actual exposure to the reactor vessel wall at 1/4t; therefore, the specimens accumulated approximately 2.23 EFPY (0.93 x 2.4) of equivalent irradiation. The total planned Cycle 1 operation would accumulate 1.37 EFPY of actual exposure to the reactor vessel at 1/4t. Therefore, the specimens removed after 0.93 EFPY of Cycle 1 operation have already received an irradiation equivalent to 2.23 EFPY which is more irradiation than the vessel wall at 1/4t would accumulate during all of Cycle 1 operation (1.37 EFPY). The irradiation effects accumulated during the 0.93 EFPY of Cycle 1 operation would not be altered in the specimens stored for the remainder of Cycle 1 operation.

On the basis of the above considerations, we have concluded that removal of the surveillance capsules during the remainder of Cycle 1 operation is acceptable and that the surveillance specimens not subjected to destructive testing after the 0.93 EFPY of Cycle 1 operation may be stored until the beginning of Cycle 2 to permit redesign of the capsule holders.

3.2 Proposed Technical Specification 4.2.7

The licensee has proposed a surveillance specimen withdrawal schedule at specimen exposures of 3, 9.5, 16, and 22.5 EFPY referenced to 1/4 reactor vessel wall thickness. Provisions are also included for updating Technical Specification 3.1.2 (relating to reactor heatup, cooldown, and pressure-temperature limits) based on the results of specimen tests. We have reviewed this proposal and determined that it meets the requirements of Appendix H to 10 CFR Part 50, Section II.C.3.C and therefore is acceptable.

3.3 Proposed Specification Regarding Hydrogen Purge System Charcoal Filters

The licensee's June 4, 1976 letter requested a waiver of the requirements of Technical Specifications 3.14.1.b and 4.12.2a which require a carbon sample analysis of the charcoal filters in the Hydrogen Purge System (HPS). These requirements were established by Amendment No. 10 to DPR-51 issued February 18, 1976. That amendment required initial inspections of the filters to be completed within 90 days of issuance of the amendment (i.e., by May 18, 1976). Specification 3.14.1.b established the carbon

*Effective Full Power Years

sample analysis criterion of ≥90% radioactive methyl iodide removal when tested at an air velocity within 20% of system design flow, 0.15 to 0.5 mgm/m^3 inlet methyl iodide concentration, ≥70% relative humidity (R.H.) and ≥ 190°F.

The licensee proposed to use replacement filters currently on hand which have shown ≥99% methyl iodine-131 removal tested at rated air flow, 50 mgm/m^3 inlet non-radioactive methyl iodide concentration, 70% R.H. and 150°F. The licensee has indicated that:

1. the on-hand filters meet all requirements of Specifications 3.14 and 4.12 except the carbon sample analysis criteria, described above,

2. carbon sample tests of the on-hand filters would require complete disassembly of the filters because of their design (normally the carbon sample tests for these filters are performed before manufacture of the filter), and

3. replacement filters which meet all the requirements of the Technical Specifications have been ordered and will be available by the end of Cycle 1 operation.

The licensee concluded in the June 4, 1976 letter that the difference in test requirements would not affect the performance of the filter units. We, however, do not fully agree with that position. The test parameters vary with regard to the temperature of the test and the inlet ion concentration. The lower temperature test performed on the on-hand filters would result in a less conservative test than the existing Technical Specifications require whereas the higher inlet concentration would result in a more conservative test than that required by the existing Technical Specifications. The on-hand filter test criterion (i.e., ≥99% methyl iodine-131 removal) is more conservative than the 90% radioactive methyl iodide removal required by the existing Technical Specifications. Considering these facts collectively, we have concluded that the health and safety of the public would not be endangered by permitting the licensee to use the on-hand filters; however, because the test criterion and parameters cannot be compared

quantitatively, we would permit use of the on-hand filters only until the next refueling shutdown (presently scheduled for October, 1976). We have discussed our conclusions with the licensee's staff and the licensee had agreed with our recommended changes to the June 4, 1976 request; namely, that the initial tests of the HPS filters be deferred until the end of Cycle 1 operation. The licensee has agreed to install fully qualified filters at the end of Cycle 1 operation.

3.4 Inspections, Cleanup, and Repairs

The licensee has completed a thorough inspection and cleanup program as described in the "Surveillance Specimen Holder Tube Repair Report", dated June 1, 1976. Visual inspections by remote television camera were performed on the reactor vessel interior wall and bottom surfaces, the core guide lugs and incore guide tubes, the reactor internals (including assembly bolts and locking caps), and core support assembly (including distribution plate, lower grid forging, incore guide tube support plate, and flow distribution head). Both video and ultrasonic test equipment were used to inspect the SSHT pintles and shroud tubes.

In the course of the inspections, debris from the failed SSHTs was removed from the vessel and internals. Large pieces were removed by grappling; smaller pieces were removed by flushing and vacuum cleaning. In addition all of the intact portions of the SSHTs were removed (including the surveillance specimens) and the shroud tubes were cut and removed in areas where significant wear had occurred (wall thickness <100 mils). The licensee reports that only minor and insignificant damage (primarily small scratches and dents) was observed on the surfaces of the reactor vessel and core internals. The licensee also reported that the "as left" condition of the shroud tubes had been evaluated and that no safety or structural problem will result from continued operation.

The licensee performed an underwater video inspection on all 177 fuel assemblies which were in the reactor at the time of the March 19, 1976 shutdown. The inspection included the "bottom grillage of the lower end fitting plus the bottom 2.75 inch section of fuel rods between the grillage and lower spacer grid."

The inspection was limited to this region because the licensee concluded that the filtering action of the lower end fitting and the lowest spacer grid would prevent recoverable debris from passing through this region. All observed debris in the lower end fitting grillage and spacer grid was removed during this inspection. The licensee indicated that only pieces smaller than 1/8" could pass through the lower region and into the active portion of the fuel assembly. Such pieces would either pass through or be lodged in the fuel assembly. The licensee also inspected two fuel assemblies over the entire length of the assembly. These assemblies had large pieces of debris entrained in the lower end fitting. No indication of debris was observed in the upper region of the bundle. The licensee found no evidence of physical damage to the fuel assemblies during the inspection.

The licensee also inspected the upper tube sheet of both once-through steam generators for possible damage or debris The tube sheets were found to be free of debris and in excellent condition.

All debris removed during the above inspection was collected and weighed. The weight was compared with the calculated weight of the same material determined from the as-built drawings of the SSHTs. The material removed was 4 lbs. less than the calculated dry weight and 6 lbs. less than the calculated wet weight. These differences are less than 1% of the total material recovered.

The licensee has evaluated the possible effects on fuel assemblies of small pieces of debris which may still remain in the fuel upon return to operation and concluded that operation of the reactor may be safely resumed. This conclusion is based on the design of the fuel, the results of the licensee's inspections (which showed no indication of damage), and the recent operating history of the reactor. In particular, the licensee has observed no significant increase in iodine activity in the primary coolant prior to shutdown even though the holder tubes were in the damaged condition. This indicates that no fuel damage had occurred as a result of the debris that was in the reactor vessel.

The NRC staff has reviewed the licensee's inspection, cleanup, and repair operations described in the June 1, 1976 report. We have concluded that the actions taken by the licensee constitute an acceptable program and that further inspection of the facility (including the fuel assemblies) is not required. It is our opinion that the small amount of material remaining within the vessel, internals, and fuel is of such small size that feasible inspection techniques to search for these materials would not be effective.

The NRC staff has also reviewed the licensee's analysis of the effects of debris which may remain in the primary coolant system. Since all the larger pieces of the disintegrated holder tubes were removed from the reactor vessel during the cleanup operation, it is very unlikely that a blockage of flow passages could occur. However, the presence of small fragments in the primary coolant could cause some problems. Some of these particles could pass through the end fitting grillage and the lower spacer grid of the fuel assemblies and be lodged in the spacer grid adjacent to the active fuel section of the rod. Under certain circumstances flow induced vibrations could cause some fretting action which could lead to damage of the fuel claddings. Some pieces of the debris could also enter the control rod guide tubes and cause some interference with the control rod operation. These two possibilities were considered by the staff.

The probability of any of these events occurring is proportional to the concentration of the particles in the coolant. This concentration is estimated to be about 1-2 particles per cubic foot of coolant for 1/8" diameter particles. The quantity of debris circulating in the primary coolant system would be continuously decreased as the purification system would remove the particles from the coolant. The half-life of the purification system in the Arkansas Nuclear One - Unit No. 1 plant is about 20 hours, hence in a relatively short time the concentration of particles would be significantly reduced. The probability that the particles would deposit

in the core is, therefore, relatively small and it would continuously decrease during the operation of the plant.

It is the staff's opinion that the possibility of some of the particles being deposited in the core cannot be completely ignored. Frequent monitoring of primary coolant iodine activity would provide indication of fuel failures caused by fretting. The staff has reviewed the licensee's primary coolant surveillance program described in the June 1, 1976 report and found it acceptable. This program, however, differs from the existing Technical Specification requirements. The staff, therefore, would require continued frequent monitoring of the radioiodine in the coolant and prompt reporting on any significant change in the steady state radioiodine concentration while the reactor is operating.

The licensee was informed of these requirements on June 9, 1976, and, by letter dated June 10, 1976, the licensee proposed Technical Specifications to change Table 4.1-3, "Minimum Sampling and Analysis Frequency", to require determination of primary coolant gross radioiodine concentrations three times per week and to require notification within 24 hours if the steady state gross radioiodine concentration increases by a factor of ten or more. This increased monitoring and reporting program would be required only until the end of Cycle 1 operation since cladding damage due to fretting, if it is to occur, should have occurred by that time. The likelihood of this wear mechanism also would be continuously diminished by the removal of the debris by the purification system. We have reviewed this request and concluded that it meets our requirements and therefore is acceptable.

The probability of interference of control rod operation by debris is very remote since the coolant entering the control rod guide tube has to pass through a 1/8" diameter orifice while the clearance between the control rod and the guide tube is approximately 1/16". Particles less than 1/16" would pass through. Thus, the only particles which might lead to interference with control rod motion are between 1/8" and 1/16". Because of the relatively small concentration of particles (of all sizes) in the coolant, the continued reduction of particle concentration by the purification system, and the fact that control rod motion

was not inhibited when the plant was operated with much more debris present (prior to the March 19, 1976 shutdown), we have concluded that interference with control rod motion is very unlikely. Blockage of the control rod guide tube orifice is not considered likely for the same reasons.

The staff has concluded that, with the incorporation of the modified radioiodine monitoring program described above, the operation of the Arkansas Nuclear One - Unit No. 1 plant could be resumed without hazard to the health and safety of the public.

4.0 ENVIRONMENTAL CONSIDERATIONS

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental statement, negative declaration, or environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the changes do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: JUN 11 1976

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 12 to Facility Operating License No. DPR-51, issued to Arkansas Power & Light Company (the licensee), which revised Technical Specifications for operation of the Arkansas Nuclear One - Unit No. 1 (the facility) located in Pope County, Arkansas. The amendment is effective as of its date of issuance.

The amendment (1) provides for the removal of surveillance capsules during the remainder of Cycle 1 operation, (2) revised Technical Specification 4.2.7 to conform with Appendix H to 10 CFR Part 50, (3) alters the charcoal filter requirements for the Hydrogen Purge System to permit reactor operation until the end of Cycle 1 with available spare filters installed in the system and (4) increases the frequency of gross radioiodine determination and establishes reporting requirements for significant changes in gross radioiodine concentration. The Commission also granted an exemption to Section II.C.2 of Appendix H to 10 CFR Part 50 which permits the licensee

to operate the facility during the remainder of Cycle 1 operation with the reactor vessel surveillance specimens and holder tubes removed from the reactor vessel.

The applications for the amendment comply 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. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental statement, negative declaration or environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated May 19, June 4 and June 10, 1976, (2) April 20, 1976 request for exemption, (3) Amendment No. 12 to License No. DPR-51, and (4) the Commission's related Safety Evaluation. 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 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 11th day of June, 1976.

FOR THE NUCLEAR REGULATORY COMMISSION

Original signed by
Dennis L. Ziemann

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