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

APR 4 1977

Gentlemen:

By letter dated August 17, 1976, you requested an exemption from the provisions of Appendix H to Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) which would permit future operation of Arkansas Nuclear One Unit No. 1 (ANO-1) while irradiating the remaining portion of the reactor vessel surveillance specimens at Davis-Besse Unit No. 1. By the same letter, you requested approval of proposed changes to the ANO-1 Technical Specifications consistent with the requested exemption. Some revisions to the proposed modified Technical Specifications were included in your letters of December 20 and 22, 1976, and January 13, 1977.

Irradiating the remaining ANO-1 surveillance specimens at Davis-Besse Unit No. 1 will cause the ANO-1 program to be out of conformance with the provision of Appendix H which requires the irradiation program to be performed within the ANO-1 vessel. However, as noted in the enclosed Safety Evaluation, the nominal dimensions of the ANO-1 reactor vessel and internals are identical to those at Davis-Besse Unit No. 1, and the operating conditions at these two reactors are very similar so that with the exception of actual operating history and minor differences in power distribution, for which adjustments can be provided, the technical aspects of the material surveillance program will be achieved to the satisfaction of the NRC staff.

Based on these considerations, we have concluded that an exemption for ANO-1 from this requirement for a continuing in-vessel material surveillance program as set forth in Appendix H to 10 CFR Part 50 for a period of five years will not be detrimental to life or property or the common defense and security and is in the public interest. Therefore, the exemption requested in your letter of August 17, 1976, is approved for a period of five years from the date of this letter.

If an extension of this exemption beyond this initial five year term is desired, you should submit an application for extension to the Commission no later than six months prior to expiration of the exemption. This application should provide a justification for extending the term of the exemption based on operating experience. 60

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APR 1 1977

In addition to granting this exemption, the Commission has issued the enclosed Amendment No. to Facility Operating License No. DPR-51 for ANO-1. This amendment provides for performance of the ANO-1 reactor vessel material surveillance program at Davis-Besse Unit No. 1 and for the submission of specified reports. Certain changes were required in the proposed Technical Specifications submitted by your staff relative to this program. These have been discussed with and agreed to by your staff.

Copies of our related Safety Evaluation and the Notice of Issuance 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. to License No. DPR-51
2. SER Report
3. Notice of Issuance

cc w/enclosures:
See next page

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NOTIFIED LICENSEE OF APPROVAL (DAVE MARDIS)
4/1/77

			DOR:AD/OT	OELD	DOR:DIR	NRR:D/DIR
			DEisenhut	3/30/77	VStello	ECase
			3/30/77	3/30/77	3/31/77	/ /
OFFICE➤	DOR:ORB #2	DOR:ORB #2	DOR:ORB #4	DOR:EB/OT	DOR:ORB #2	DOR:AD/OR
SURNAME➤	RMDiggs	RPSnaider:ro	GZwetzig	LShao	DKDavis	KRGoller
DATE➤	3/30/77	3/30/77	3/30/77	3/30/77	3/30/77	3/30/77

AS 4/1/77

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

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

cc w/enclosures and copy of AP&L
filings dtd. 8/17/76, 12/20/76,
12/22/76 and 1/13/77:
Director, Bureau of Environmental
Health Services
4815 West Markham Street
Little Rock, Arkansas 72201

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE - UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.
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 August 17, 1976, as supplemented by letters dated December 20 and 22, 1976, and January 13, 1977, 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:

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SURNAME➤						
DATE➤						

(2) Technical Specifications

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

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance:

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO.

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A portion of the Technical Specifications with the attached revised pages. The changed areas on the revised pages are identified by a marginal line.

REMOVE	ADD
17	17*
18	18
18a	18a
19	19
20	20
77	77
- -	77a
- -	77b

*There were no changes on this page. It is included as a matter of convenience in updating the Technical Specifications.

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SURNAME ➤						
DATE ➤						

A reactor coolant pump or decay heat removal pump is required to be in operation before the boron concentration is reduced by dilution with makeup water. Either pump will provide mixing which will prevent sudden positive reactivity changes caused by dilute coolant reaching the reactor. One decay heat removal pump will circulate the equivalent of the reactor coolant system volume in one half hour or less.[1]

The decay heat removal system suction piping is designed for 300 F thus, the system can remove decay heat when the reactor coolant system is below this temperature. (2,3)

One pressurizer code safety valve is capable of preventing overpressurization when the reactor is not critical since its relieving capacity is greater than that required by the sum of the available heat sources which are pump energy, pressurizer heaters, and reactor decay heat.[4] Both pressurizer code safety valves are required to be in service prior to criticality to conform to the system design relief capabilities. The code safety valves prevent overpressure for a rod withdrawal accident.[5] The pressurizer code safety valve lift set point shall be set at 2500 psig $\pm$ 1 percent allowance for error and each valve shall be capable of relieving 300,000 lb/h of saturated steam at a pressure not greater than 3 percent above the set pressure.

The internals vent valves are provided to relieve the pressure generated by steaming in the core following a LOCA so that the core remains sufficiently covered. Inspection and manual actuation of the internals vent valves (1) ensure operability, (2) ensure that the valves are not open during normal operation, and (3) demonstrate that the valves begin to open and are fully open at the forces equivalent to the differential pressures assumed in the safety analysis.

REFERENCES

(1) FSAR, Tables 9-10 and 4-3 through 4-7.

(2) FSAR, Section 4.2.5.1 and 9.5.2.3.

(3) FSAR, Section 4.2.5.4.

(4) FSAR, Section 4.3.10.4 and 4.2.4.

(5) FSAR, Section 4.3.7.

3.1.2 Pressurization, Heatup, and Cooldown Limitations

Specification

3.1.2.1 Hydro Tests:

For thermal steady state system hydro tests the system may be pressurized to the limits set forth in Specification 2.2 when there are fuel assemblies in the core and to ASME Code Section III limits when no fuel assemblies are present provided:

a. Prior to initial criticality the reactor coolant system temperature is 100°F or greater, or

b. After initial criticality and prior to the accumulation of 1.7×10^6 thermal megawatt-days operation the reactor coolant system temperature is 215°F or greater.

3.1.2.2 Leak Tests

a. Leak tests may be conducted under the provisions of 3.1.2.1 above or

b. After initial criticality and prior to the accumulation of 1.7×10^6 thermal megawatt-days operation the system may be tested to a pressure of 1150 psig provided that the system temperature is 175°F or greater.

3.1.2.3 The reactor coolant pressure and the system heatup and cooldown rates (with the exception of the pressurizer) shall be limited in accordance with Figure 3.1.2-1 and Figure 3.1.2-2, and are as follows:

Heatup:

Allowable combinations of pressure and temperature shall be to the right of and below the limit line in Figure 3.1.2-1. The heatup rates shall not exceed those shown on Figure 3.1.2-1.

Cooldown:

Allowable combinations of pressure and temperature for a specific cooldown shall be to the left of and below the limit line in Figure 3.1.2-2. Cooldown rates shall not exceed those shown in Figure 3.1.2-2.

3.1.2.4 The secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator shell is below 100°F.

3.1.2.5 The pressurizer heatup and cooldown rates shall not exceed 100 F/hr. The spray shall not be used if the temperature difference between the pressurizer and the spray fluid is greater than 430°F.

3.1.2.6 Prior to exceeding 1.7.x 10^6 thermal megawatt-days of operation, Figures 3.1.2-1 and 3.1.2-2 and Technical Specifications 3.1.2.1.b and 3.1.2.2 shall be updated for the next service period in accordance with 10 CFR 50 Appendix G, Section V.B. The service period shall be of sufficient duration to permit the scheduled evaluation of a portion of the surveillance data scheduled in accordance with Specification 4.2.7. The highest predicted adjusted reference temperature of all the beltline region materials shall be used to determine the adjusted reference temperature at the end of the service period. The basis for this prediction shall be submitted for NRC staff review in accordance with Specification 3.1.2.7.

3.1.2.7 The updated proposed technical specifications referred to in 3.1.2.6 shall be submitted for NRC review at least 90 days prior to the end of the service period. Appropriate additional NRC review time shall be allowed for proposed technical specifications submitted in accordance with 10 CFR Part 50, Appendix G, Section V.C.

BASES

All reactor coolant system components are designed to withstand the effects of cyclic loads due to system temperature and pressure changes.(1) These cyclic loads are introduced by unit load transients, reactor trips, and unit heatup and cooldown operations. The number of thermal and loading cycles used for design purposes are shown in Table 4-8 of the FSAR. The maximum unit heatup and cooldown rate of 100 F per hour satisfies stress limits for cyclic operation.(2) The 200 psig pressure limit for the secondary side of the steam generator at a temperature less than 100 F satisfies stress levels for temperatures below the DTT.(3) The plate material and welds in the core region of the reactor vessel have been tested to verify conformity to specified requirements and a maximum NDTT value of 10 F has been determined based on Charpy V-notch tests. The maximum NDTT value obtained for the steam generator shell material and welds was 40 F.

Figures 3.1.2-1 and 3.1.2-2 contain the limiting reactor coolant system pressure-temperature relationship for operation at DTT[4] and below to assure that stress levels are low enough to preclude brittle fracture. These stress levels and their bases are defined in Section 4.3.3 of the FSAR.

As a result of fast neutron irradiation in the region of the core, there will be an increase in the NDTT with accumulated nuclear operation. The predicted maximum NDTT increase for the 40-year exposure is shown on Figure 4-10.[4] The actual shift in NDTT will be determined periodically during plant operation by testing of irradiated vessel material samples located in this or a similar reactor vessel.[5] The results of the irradiated sample testing will be evaluated and compared to the design curve (Figure 4-11 of FSAR) being used to predict the increase in transition temperature.

The design value for fast neutron (E > 1 Mev) exposure of the reactor vessel is 3.0×10^{10} n/cm^{2}sec at 2568 MWt rated power and an integrated exposure of 3.0×10^{19} n/cm^2 for 40 years operation.[6] The calculated maximum values are 2.2×10^{10} n/cm^{2}sec and 2.2×10^{19} n/cm^2 integrated exposure for 40 years operation at 80 percent load.[4] Figure 3.1.2-1 is based on the design value which is considerably higher than the calculated value. The DTT value for Figure 3.1.2-1 is based on the projected NDTT at the end of the first two years of operation. During these two years, the energy output has been conservatively estimated to be 1.7×10^6 thermal megawatt days which is equivalent to 655 days at 2568 MWt core power. The projected fast neutron exposure of the reactor vessel for the two years is 1.7×10^{18} n/cm^2 which is based on the 1.7×10^6 thermal megawatt days and the design value for fast neutron exposure.

The actual shift in NDTT will be established periodically during plant operation by testing vessel material samples which are irradiated cumulatively by securing them near the inside wall of this or a similar vessel in the core area. To compensate for the increases in the NDTT caused by irradiation, the limits on the pressure-temperature relationship are periodically changed to stay within the established stress limits during heatup and cooldown.

The NDTT shift and the magnitude of the thermal and pressure stresses are sensitive to integrated reactor power and not to instantaneous power level. Figures 3.1.2-1 and 3.1.2-2 are applicable to reactor core thermal ratings up to 2568 MWt.

The pressure limit line on Figure 3.1.2-1 has been selected such that the reactor vessel stress resulting from internal pressure will not exceed 15 percent yield strength considering the following:

A. A 25 psi error in measured pressure.

B. System pressure is measured in either loop.

C. Maximum differential pressure between the point of system pressure measurement and reactor vessel inlet for all operating pump combinations.

For adequate conservatism, in lieu of portions of the operational requirements of Appendix G to 10 CFR 50, a maximum pressure of 550 psig and a maximum heatup rate of 50 F/hr (averaged over one hour) has been imposed below 275°F as shown on Figure 3.1.2-1.

The spray temperature difference restriction based on a stress analysis of the spray line nozzle is imposed to maintain the thermal stresses at the pressurizer spray line nozzle below the design limit. Temperature requirements for the steam generator correspond with the measured NDTT for the shell.

The heatup and cooldown rates stated in this specification are intended as the maximum changes in temperature in one direction in a one hour period. The actual temperature linear ramp rate may exceed the stated limits for a time period provided that the maximum total temperature difference does not exceed the limit and that a temperature hold is observed to prevent the total temperature difference from exceeding the limit for the one hour period.

REFERENCES

(1) FSAR, Section 4.1.2.4

(2) ASME Boiler and Pressure Code, Section III, N-415

(3) FSAR, Section 4.3.10.5

(4) FSAR, Section 4.3.3

(5) FSAR, Section 4.4.5

(6) FSAR, Sections 4.1.2.8 and 4.3.3

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 The reactor vessel material irradiation surveillance specimens removed from the reactor vessel in 1976 shall be installed, irradiated in and withdrawn from the Davis-Besse Unit No. 1 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2-1, Arkansas Power & Light Company shall be responsible for testing the specimens and submitting a report of test results in accordance with 10 CFR 50, Appendix H.

4.2.8 The licensee shall submit a report or application for license amendment to the NRC within 90 days after the occurrence of any of the following:

1. Failure of Davis-Besse Unit No. 1 to achieve commercial operation at 100% power by January 1, 1978, or

2. Beginning one year after attainment of commercial operation at 100% power, any time that Davis-Besse Unit No. 1 fails to maintain a cumulative reactor utilization factor of greater than 65%.

The report shall provide justification for continued operation of ANO-1 with the reactor vessel surveillance program conducted at Davis-Besse Unit No. 1 or the application for license amendment shall propose an alternative program for conduct of the ANO-1 reactor vessel surveillance program.

Table 4.2-1

ANO-1 CAPSULE ASSEMBLY WITHDRAWAL SCHEDULE AT DAVIS-BESSE 1

CAPSULE	INSERTION/WITHDRAWAL
ANI-E	Has been withdrawn for testing
ANI-B	Withdraw following 1st cycle at Davis-Besse 1
ANI-A	Withdraw following 3rd cycle at Davis-Besse 1
ANI-C	Withdraw following 7th Cycle at Davis-Besse 1
ANI-D	Insert in location WZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 12th cycle
ANI-F	Insert in location YZ (upper) prior to 4th cycle at Davis-Besse 1; withdraw following 11th cycle

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 1972 Summer 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.

For the purpose of Technical Specification 4.2.8, the definition of Regulatory Guide 1.16, Revision 4 (August 1975) applies for the term "commercial operation". Cumulative reactor utilization factor is defined as: [(Cumulative thermal megawatt hours since attainment of commerical operation at 100% power) x 100] ÷ [(licensed thermal power) x (cumulative hours since attainment of commercial operation at 100% power)].

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

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

ARKANSAS POWER & LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 1

DOCKET NO. 50-313

INTRODUCTION

By letter dated August 17, 1976, as supplemented by letters dated December 20 and 22, 1976, and January 13, 1977, Arkansas Power & Light Company (AP&L) requested that (1) the exemption to Appendix H of Title 10 of the Code of Federal Regulations Part 50 (10 CFR 50) granted for Arkansas Nuclear One - Unit No. 1 (ANO-1) by letter of June 11, 1976, be modified to allow indefinite operation of ANO-1 with the remainder of the reactor vessel surveillance capsules to be irradiated at Davis-Besse Unit No. 1 rather than in-situ, and (2) the ANO-1 Technical Specifications be revised to allow the remainder of ANO-1 reactor vessel surveillance capsules to be irradiated at Davis-Besse Unit No. 1. We have determined that several modifications to the AP&L proposal were necessary to meet regulatory requirements. These modifications have been discussed with and agreed to by the AP&L staff.

DISCUSSION AND EVALUATION

The original ANO-1 design included three reactor vessel surveillance specimen holder tubes (SSHTs) located near the reactor inside vessel wall. Each of these SSHTs housed two capsules containing reactor vessel surveillance specimens. When failures of the SSHTs occurred at other Babcock & Wilcox (B&W) designed plants, the licensee shut down the ANO-1 facility on March 19, 1976, to inspect the SSHTs. The inspection revealed that all of the SSHTs had suffered severe damage and that portions of two SSHTs had fallen to the bottom of the reactor vessel. To prevent further damage, all surveillance capsules and all parts of the SSHTs that had failed or were deemed likely to fail during the remainder of that operating cycle (Cycle 1) were removed from the vessel.

Since the discovery of the damage to the SSHTs, Babcock & Wilcox Company (B&W), the reactor supplier, has undertaken the design, manufacture and testing of an improved SSHT. SSHTs of this improved design are presently installed in Davis-Besse Unit No. 1 and Crystal River Unit No. 3 and are planned for installation in Three Mile Island Unit No. 2. All three of these plants have reactors supplied by B&W and all are in the process of beginning initial operation within the next few months. In addition, all of these reactors are of the same basic B&W 177 fuel assembly vessel design as ANO-1. The acceptability of the redesigned SSHTs has been demonstrated by a test program reviewed and approved by the staff and performed in conjunction with the Hot Functional Test performed at Davis-Besse Unit No. 1.

Installation of the redesigned SSHTs in the Davis-Besse Unit No. 1 and Crystal River Unit No. 3 reactor vessels did not present any unusual difficulties because it was performed prior to neutron activation of the reactor internals. This will also be the case at Three Mile Island Unit No. 2. Studies of methods to install the redesigned SSHTs in the irradiated B&W reactors indicate that substantial difficulties will be experienced, primarily because precision machining, alignment and inspection must be performed remotely and under water. Although such problems do not in themselves justify relief from a requirement to re-install the SSHTs in ANO-1, they would cause significant radiation to personnel. Based on their experience in removing the SSHTs at Three Mile Island Unit No. 1 and Rancho Seco Unit No. 1, B&W estimated that installing SSHTs in irradiated reactors would result in personnel exposure of about 100 man-rem per reactor. In the interest of maintaining the radiation exposure of plant personnel as low as reasonably achievable, the licensee, in cooperation with B&W and the owners of other B&W 177 fuel assembly plants, has proposed an alternative program that does not require re-installing the SSHTs in ANO-1 and the other irradiated B&W plants.

This program is complex and includes provisions to provide additional information, if required under Appendix G, 10 CFR 50, Paragraph V.C., in addition to the normal requirements of Appendix H.

The proposed plan involves integrating the interrupted surveillance programs at operating reactors which suffered damage to SSHTs into the programs for new plants in a manner generally similar to that covered in Appendix H, 10 CFR 50, Paragraph II.C.4, except that the surveillance program for reactors such as ANO-1 would be at different sites. There are three distinct features of these proposed programs:

1. A host-reactor feature, in which the original surveillance materials from one or more reactors that have been in service will now be irradiated in a new host reactor which has been fitted with the newly-designed capsule holders on the thermal shield in less time and without radiation exposure of the workmen;

2. An augmented surveillance feature in which more weld metal specimens and some larger fracture mechanics [compact tension (CT)] specimens will be included in the program; and

3. A data-sharing feature in which all available irradiation data for all of the beltline welds of a given reactor vessel will be considered by the licensee or his consultants in predicting the adjusted reference temperature and in making any fracture analyses for that vessel. Typically, several of the welds in any one vessel were made with the same weld wire and flux as those used on some other reactors. The data sharing feature is required because the welds in these reactors have high radiation sensitivity due to high copper content, large and random variation of copper from point to point in the weld, and low initial upper shelf energy.

The specific program proposed for ANO-1 involves installing the remaining original ANO-1 surveillance capsules (one has been removed and tested) in extra locations provided in the Davis-Besse Unit No. 1 vessel. This plan will accomplish the original purpose of obtaining information on the effect of radiation on material that is representative of (although not identical to) the material in the ANO-1 reactor vessel on a schedule that provides an appropriate lead time over the vessel irradiation rate. The overall integrated program also will provide information from surveillance programs in Crystal River Unit No. 3, Three Mile Island Unit No. 2, and Davis-Besse Unit No. 1 on material considered to be essentially identical to the actual welds in the ANO-1 vessel. It is also important to note that still more information relevant to the ANO-1 vessel materials will be obtained from the NRC sponsored Heavy Section Steel Technology (HSST) irradiation programs underway. Details are provided below.

Two weld materials are of primary interest for the ANO-1 vessel, Procedure Qualification (P.Q.) numbers* WF 112 and WF 182-1. These are used in

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*Weld materials are specifically identified by the ASME Code by the procedure qualification test number. A procedure qualification test is required on each combination of heat of weld wire and batch of flux.

the top and center circumferential welds. The end of life (EOL) fluence for both of these welds is estimated to be 1.1×10^{19} nvt, and both have compositions that are expected to make them relatively sensitive to radiation damage. Weld P.Q. No. WF 18, used for the longitudinal welds, has low copper, making it less sensitive to radiation. Further, the EOL fluence at the azimuthal locations of these longitudinal welds is low (7×10^{18}) so they will not become limiting during the service life. Another shell weld, the lower circumferential, is made of a material that is expected to be radiation sensitive (P.Q. No. SA 1788), but the EOL fluence at this location is estimated to be at least an order of magnitude lower than that of the other circumferential welds, so it is not expected to be limiting.

The orginal ANO-1 surveillance material, WF 193, used the same heat of filler wire as WF 112 but a different batch of flux. Metallurgical considerations suggest that the radiation behavior is affected more by the wire than the flux, thus it is expected to respond to radiation much like WF 112.

The following table shows where samples of these three pertinent weld materials will be irradiated in the proposed integrated program, what kinds of specimens will be used, and when information will be available under the present plan.

WELD	CAPSULE DESIGNATION*	IRRADIATION LOCATIONS	INFORMATION AVAILABLE	SPECIMEN TYPES **
WF 112	OCI-E	Oconee Unit No. 1	1977	Cv (already removed)
	OCI-A	Crystal River Unit No. 3	1985	Cv
	OCI-C	Crystal River Unit No. 3	1989	Cv
WF 182	TMI-2-B	Three Mile Island Unit No. 2	1985	Cv, CT
	TMI-2-D	Three Mile Island Unit No. 2	1995	Cv
	TMI-2-F	Three Mile Island Unit No. 2	1998	Cv, CT
	TE1-F	Davis-Besse Unit No. 1	1979	Cv, CT
	TE1-B	Davis-Besse Unit No. 1	1983	Cv, CT
	TE1-D	Davis-Besse Unit No. 1	1992	Cv, CT
WF 193	AN1-E	Arkansas Unit No. 1	1977	Cv (already removed)
	AN1-A	Davis-Besse Unit No. 1	(to be	Cv
	AN1-C	Davis-Besse Unit No. 1	determined)	Cv

The irradiation schedule and withdrawal dates shown will be modified as initial test results are obtained and evaluated to optimize the information obtained.

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*OCI-E - means capsule E from the Oconee Unit No. 1 reactor
TMI-2-B - means capsule B in the Three Mile Island Unit No. 2 reactor
TE1-F - means capsule F in Toledo Edison's Davis-Besse Unit No. 1 reactor
ANI-E - means capsule E from Arkansas Nuclear One - Unit No. 1 reactor
**Cv - means Charpy V-notch specimen
CT - means Compact Tension specimen

In addition to this integrated program, "research" capsules containing tensile, Charpy V-notch (Cv), and several sizes of CT specimens will be included in the overall B&W power reactor surveillance program. Samples of the weld most likely to be limiting in ANO-1, P.Q. WF 112, will be irradiated in Davis-Besse, and samples of a weld made of the same heat of weld wire as WF 182-1 will be irradiated in the Crystal River Unit No. 3 program. Details of withdrawal schedules will be determined later, and will depend on test results from the other programs.

Research programs being sponsored by the NRC will also provide useful information on the effect of radiation on these specific weld materials and on several additional B&W weld materials expected to respond to radiation in a similar manner. These programs, HSST-2 and HSST-3, consist of many tensile, Cv and CT specimens irradiated in a test reactor. Although information on shift in the reference temperature for nil-ductility (RT_{NDT}) will be obtained, the main emphasis of the HSST programs is to develop methods that can be used to better evaluate low shelf toughness using the rather small specimens used in the power reactor programs.

The staff has evaluated the effectiveness of this overall program plan, and has concluded that the information to be developed that is directly and indirectly relevant to the ANO-1 reactor vessel will be sufficient to provide assurance of safety margins against vessel failure that comply with Appendix G, 10 CFR 50.

Until data become available from the surveillance program, a conservative prediction of radiation damage can be made by using R.G. 1.99* for at least the next five years of operation. This Regulatory Guide is based on the NRC staff's analysis of all data available at the time it was written. New data, in particular the results of the augmented integrated surveillance program described above, will be used to periodically update the Regulatory Guide. Predictions of the adjustment of reference temperature and the drop in upper shelf energy are given graphically in R.G. 1.99 as functions of copper and phosphorus content and of fluence. In addition there is an "Upper Limit" line on each graph, which is to be used when information about the copper and phosphorus contents is inadequate. Because the chemical analyses of the B&W welds have shown considerable variation, the NRC staff intends to use the Upper Limit lines as the basis for any prediction required at this time.

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*Regulatory Guide 1.99, "Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials", July 1975. Revision 1 is to be published in April 1977.

We have also considered the uncertainties involved in applying radiation effects information obtained in other reactors to the ANO-1 vessel. The major uncertainties involved are:

1. Accuracy of neutron fluence calculations;

2. Magnitude and effect of variation in neutron spectra between reactors;

3. Magnitude and effect of variations in irradiation temperature between reactors;

4. Magnitude and effect of variations in rate of irradiation on material properties.

The effects of these variables have been studied for at least 20 years. Although some uncertainties still remain, the effects are fairly well established and understood as discussed below.

1. Calculational methods for estimating the neutron flux at the reactor vessel wall and at irradiation capsule locations have been developed over many years. The dosimetry used in irradiation capsules has furnished information that was used to check out and refine the calculational methods. As a result, the fast neutron flux and fluence in these locations can generally be calculated to an accuracy of $\pm$ 20%, particularly if some dosimetry checks are available. Dosimeters from the original ANO-1 surveillance program were removed and tested, so the fluence calculations for the vessel can be verified.

   In addition, it should be noted that the effect of neutron radiation on reactor vessel steel varies as the square root of the fluence; hence, uncertainties of 20 to 50% in fluence are not highly significant.

   We have also considered the fact that the design of the ANO-1 vessel, internals, and core is nominally identical to that of the other reactors which will be used to obtain radiation effects information.

   These considerations are the basis for our conclusion that uncertainties in the calculation of neutron fluence will be small, and the effect of such uncertainties on the assessment of the radiation effects on the vessel material will also be small.

2. Although differences in neutron energy spectra can cause uncertainties in the effects of radiation on material when this is evaluated without considering spectrum effects, only very large differences in spectra are significant. The variations from one B&W reactor to another are stated to be relatively minor, because they have similar geometry.

   We considered the possible differences in neutron spectra that could occur between the B&W power reactors involved in the integrated program. Such effects can be dealt with, if necessary, through methods that are being developed for that purpose. However, the worst expected differences are judged inconsequential based on present knowledge of irradiation effects. The neutron spectrum uncertainty will be kept under active scrutiny by the NRC staff and if additional developments (theoretical or experimental) suggest that the effect might be significant under some conditions, appropriate adjustments in reference temperature, drop in upper shelf energy or other suitable parameter can be made.

3. The effect of the temperature of irradiation has also been the subject of considerable research. It is well known that radiation damage is less severe at 600°F than at 500°F (the temperature range of concern). The differences in effect on the steel appear to be noticeable and should be taken into account if the irradiation temperature difference is over about 25°F. Enough information is known to permit conservative evaluations of the effect of temperature differences of at least 50°F, and probably even 100°F or more. The differences in the temperature of the surveillance capsules and vessel walls between the B&W power reactors involved in the integrated program are expected to be less than 50°F, and can be conservatively evaluated.

4. The effect of irradiation rate has also been evaluated by research programs at the Naval Research Laboratory (NRL) and other laboratories. Although the consensus of experts on this subject is that there will be no major differences in material property changes by irradiation rates varying over 2 to 3 orders of magnitude, more data from surveillance programs are needed to provide verification. However, the differences in the rates of irradiation of specimens in the integrated program and the limiting material in the walls of the affected vessels will be less than one order of magnitude. Therefore, we have concluded that there will be no significant uncertainties in this program associated with differences in rate of irradiation.

CONCLUSION

We have evaluated the adequacy of the proposed integrated, augmented reactor vessel material irradiation program for ANO-1 as an alternative to the original program that was interrupted by failure of the associated hardware. We conclude that the proposed program will provide the information required to comply with Appendix G, 10 CFR 50, and that the uncertainties involved in using data obtained from surveillance specimens irradiated in various other B&W power reactors to establish ANO-1 vessel operating limitations are small and can be accounted for by imposition of appropriate margins. We also conclude that the associated Technical Specification changes to implement the program are acceptable.

Additionally, the proposed integrated, augmented program (with possible minor modification yet to be finalized) should provide more useful information than could have been extracted from the original surveillance program. The proposed program will also give results of the kind required to meet Paragraph V.C of Appendix G, 10 CFR 50.

Until the results of the proposed surveillance program become available, our predictions of radiation damage in the B&W power reactors will be based on the current revision of Regulatory Guide 1.99. At present, this is Revision 1.

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 impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability of consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does 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:

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

This amendment authorized changes in the Technical Specifications to permit irradiation of the remaining Arkansas Nuclear One - Unit No. 1 (ANO-1) reactor vessel surveillance specimens at Davis-Besse Unit No. 1 (Docket No. 50-346). An exemption to that provision of Appendix H to Title 10 of the Code of Federal Regulations Part 50, which would have otherwise required irradiation of the capsules in the ANO-1 vessel, has been issued as a part of this action. Such action was in response to a generic failure of first-generation design Surveillance Specimen Holder Tubes (SSHTs) at ANO-1 and other operating Babcock & Wilcox 177 fuel assembly reactors.

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

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and regulations.in 10 CFR Chapter I, which are set forth in the license amendment. Notice of Proposed Issuance of Amendment to Facility Operating License was published in the Federal Register on October 21, 1976 (41 F.R. 46521). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

The Commission has determined that the issuance of the amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and 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 August 17, 1976, as supplemented by letters dated December 20 and 22, 1976, and January 13, 1977, (2) Amendment No. to Facility Operating 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 single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated in Bethesda, Maryland, this

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors