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

SEP 1 1978

Docket No. 50-368

Arkansas Power and Light Company
ATTN: Mr. William Cavanaugh III
Executive Director of Generation
and Construction
P. O. Box 551
Little Rock, Arkansas 72203

Gentlemen:

SUBJECT: ISSUANCE OF AMENDMENT NO. 1 TO FACILITY OPERATING LICENSE NO. NPF-6 (ARKANSAS NUCLEAR ONE, UNIT 2)

The Nuclear Regulatory Commission has issued the enclosed Amendment No. 1 to Facility Operating License No. NPF-6 (enclosed) to the Arkansas Power and Light Company for the Arkansas Nuclear One - Unit 2 plant authorizing operation at 2815 megawatts thermal subject to the conditions delineated therein. However, the operation of the facility is temporarily restricted to the sequence of operational modes described in Attachment 1 to the license until the preoperational tests, startup tests and other items noted in Attachment 1 are completed to the written satisfaction of the Commission. A copy of the Notice of Issuance is also enclosed.

The Commission's Office of Nuclear Reactor Regulation has also issued Supplement No. 2 to the Safety Evaluation Report. Two copies are enclosed for your information and use.

The Commission has determined that the facility requires exemptions from certain requirements of (1) Sections 50.55a(g)(2) and 50.55a(g)(4) of 10 CFR Part 50, (2) Appendices G and H to 10 CFR Part 50, and (3) Appendix J to 10 CFR Part 50. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. These exemptions have, therefore, been

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granted. The Office of Nuclear Reactor Regulation safety evaluations supporting the granting of these exemptions are enclosed.

Sincerely,

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 1 to Facility Operating License No. NPF-6
2. Federal Register Notice
3. Safety Evaluation Supporting Amendment No. 1 to NPF-6
4. Safety Evaluations Supporting Exemptions
5. Supplement No. 2 to SER (NUREG-0308)

ccs w/encls:
See page 2

cc: Mr. Daniel H. Williams
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ARKANSAS POWER AND LIGHT COMPANY

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT 2

FACILITY OPERATING LICENSE

Amendment No. 1
License No. NPF-6

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

   A. The issuance of this license amendment to Arkansas Power and Light Company (the licensee) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations set forth in 10 CFR Chapter I;

   B. Construction of Arkansas Nuclear One, Unit 2 (the facility) has been substantially completed in conformity with Construction Permit No. CPPR-89 and the application, as amended, the provisions of the Act and the regulations of the Commission;

   C. The facility requires exemptions from certain requirements of (1) Sections 50.55a(g)(2) and 50.55a(g)(4) of 10 CFR Part 50, (2) Appendices G and H to 10 CFR Part 50 and (3) Appendix J to 10 CFR Part 50 for a period of three years. These exemptions are described in the Office of Nuclear Reactor Regulation's safety evaluations supporting the granting of these exemptions which are enclosed in the letter transmitting this license amendment. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. The exemptions are, therefore, hereby granted. With the granting of these exemptions, the facility will operate in conformity with the application, as amended, the provisions of the Act, and the regulations of the Commission;

   D. There is reasonable assurance: (i) that the activities authorized by this operating license can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the regulations of the Commission;

   E. The licensee is technically and financially qualified to engage in the activities authorized by this operating license in accordance with the regulations of the Commission;

   F. The licensee has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements," of the Commission's regulations;

G. The issuance of this amended operating license will not be inimical to the common defense and security or to the health and safety of the public;

H. After weighing the environmental, economic, technical and other benefits of the facility against environmental and other costs and considering available alternatives, the issuance of Amendment No 1 to Facility Operating License No. NPF-6 subject to the conditions for protection of the environment set forth herein, is in accordance with 10 CFR Part 51 (formerly Appendix D to 10 CFR Part 50) of the Commission's regulations and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct and special nuclear material as authorized by this license will be in accordance with the Commission's regulations in 10 CFR Parts 30, 40 and 70, including 10 CFR Sections 30.33, 40.32, 70.23 and 70.31.

2. Amendment No. 1 hereby amends Facility Operating License No NPF-6 to Arkansas Power and Light Company in its entirety to read as follows:

A. This amended license applies to Arkansas Nuclear One, Unit 2, a pressurized water reactor and associated equipment (the facility) owned by the Arkansas Power and Light Company. The facility is located in Pope County, Arkansas and is described in the Final Safety Analysis Report as supplemented and amended (Amendments 20 through 47) and the Environmental Report as supplemented and amended (Amendments 1 through 7).

B. Subject to the Conditions and requirements incorporated herein, the Commission hereby licenses Arkansas Power and Light Company:

(1) Pursuant to Section 103 of the Act and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility at the designated location in Pope County, Arkansas in accordance with the procedures and limitations set forth in this amended license;

(2) Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Final Safety Analysis Report, as supplemented and amended;

(3) Pursuant to the Act and 10 CFR Parts 30, 40 and 70 to receive, possess and use at any time any byproduct, source and special nuclear material as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts as required;

(4) Pursuant to the Act and 10 CFR Parts 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source or special nuclear material without restriction to chemical or physical form, for sample analysis or instrument calibration or associated with radioactive apparatus or components; and

(5) Pursuant to the Act and 10 CFR Parts 30 and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of the facility.

C. This amended license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I; Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; and is subject to all applicable provisions of the Act and to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) Maximum Power Level

The licensee is authorized to operate the facility at steady state reactor core power levels not in excess of 2815 megawatts thermal. Prior to attaining the power level Arkansas Power and Light Company shall comply with the applicable conditions identified in Paragraph (3) below and complete the preoperational tests, startup tests and other items identified in Attachment 2 to this license in the sequence specified. Attachment 2 is an integral part of this license.

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B to the original NPF-6 Arkansas Nuclear One - Unit 2 license are hereby incorporated in this license. Arkansas Power and Light Company shall operate the facility in accordance with the Technical Specifications except for the following specific exemptions:

The licensee shall be exempted from compliance with the following Appendix A Technical Specification related to the steam generator low water level trip setpoint while conducting the steam generator feedwater system waterhammer testing during the initial startup and power ascension testing program. The value of the steam generator low water level trip setpoint in Item 8(b) of Technical Specification Table 3.3-4 may be reduced, during this testing only, from a value of greater than or equal to 49.4 percent to greater than or equal to 10.0 percent. The licensee shall be exempted from compliance with Appendix A Technical Specification 3.3.3.6 for the Containment Radiation Monitors during Mode 3 operations.

(3) Additional Conditions

The matters specified in the following conditions shall be completed to the satisfaction of the Commission within the stated time periods following the issuance of the license or within the operational restrictions indicated. The removal of these conditions shall be made by an amendment to the license supported by a favorable evaluation by the Commission.

(a) Fuel Performance

Arkansas Power and Light Company shall, prior to startup for that cycle of operation in which burnups greater than 20,000 megawatt days per ton of uranium are expected to be attained, provide for Commission review and obtain Commission approval of revised fission gas release calculations and other affected analyses utilizing fission gas release calculational methodology approved for burnups greater than 20,000 megawatt days per ton of uranium.

(b) Fire Protection

Arkansas Power and Light Company shall maintain in effect and fully implement all provisions of the approved fire protection program. The approved fire protection program consists of the licensee's documents as follows:

Date	Document
-	Final Safety Analysis Report Section 9.5.1 through Amendment 47 dated June 16, 1978
May 17, 1977	Letter submitting a comparison of the ANO-2 Fire Protection Program to Branch Technical Position 9.5-1.
August 30, 1977	Letter transmitting the Fire Hazards Analysis and responses to staff questions.

Date	Document
September 21, 1977	Letter transmitting responses to staff questions and positions.
October 26, 1977	Letter transmitting Fire Hazards Analysis Information and responses to staff questions and positions.
February 3, 1978	Letter transmitting Fire Hazards Analysis Information.
February 28, 1978	Letter transmitting Administrative Controls information.
March 31, 1978	Letter transmitting responses to staff questions.
April 12, 1978	Letter transmitting responses to staff questions and positions.
April 17, 1978	Letter transmitting responses to staff questions and positions.
April 26, 1978	Letter transmitting responses to staff questions and positions.
June 8, 1978	Letter transmitting affidavit for previously docketed letters.
June 13, 1978	Letter transmitting Administrative Controls information.
June 15, 1978	Letter transmitting Administrative Controls information.
June 29, 1978	Letter transmitting Administrative Control information.
July 7, 1978	Letter transmitting affidavit for previously docketed letters.
July 7, 1978	Letter transmitting Administrative Controls information.
July 13, 1978	Letter transmitting affidavit for previously docketed letters.

(c) Less Than Four Reactor Coolant Pump Operation

The licensee shall not operate the reactor in operational Modes 1 and 2 with fewer than four reactor coolant pumps in operation, except as allowed by Special Test Exception 3.10.3 of the facility Technical Specifications.

(d) Instrument Trip Setpoints Drift Allowance

Prior to February 28, 1979 the licensee shall submit for Commission review the following values for each Reactor Protection System and Engineered Safety Features instrumentation channel for incorporation in the Technical Specifications:

(1) the instrument drift assumed to occur during the interval between technical specification surveillance tests;

(2) the components of the cumulative instrument bias; and

(3) the minimum margin between the technical specification trip setpoint and the trip value assumed in the accident analysis.

(e) Fire Protection

The licensee shall complete the following modifications by the indicated dates in accordance with the staff's findings as set forth in the fire protection evaluation report, NUREG-0223 "Fire Protection Safety Evaluation Report."

Implementation Dates for Proposed Modifications

Applicable Section of NUREG-0223		Date
3.1	Portable Radio Communication Equipment	March 31, 1979
3.2	Separation of Power Cables in Manholes	*
3.3	Protection from Water Spray	September 1, 1978
3.4	Protection of Redundant Cables in the MCC Room (2096-M)	December 30, 1978
3.5	Protection of Redundant Cables in the Hallway - Elevation 372 (2109-U)	*, **
3.6	Protection of Redundant Cables in the Cable Spreading Room (2098-L)	*
3.7	Protection of Redundant Cables in the Switchgear Room (2100-Z)	*
3.8	Protection of Redundant Cables in the Electrical Equipment Room (2091-BB)	September 30, 1978
3.9	Protection of Redundant Cables in the Lower South Electrical Penetration Room (2111-T)	September 30, 1978
3.10	Protection of Safe Shutdown Cables in the Upper South Piping Penetration Room (2084-DD)	September 30, 1978
3.11	Protection of Redundant Reactor Protection System Cables (2136-I)	*, **
3.12	Fire Dampers	September 30, 1978
3.13	Portable Extinguisher for the Control Room (2199-J)	November 15, 1978
3.14	Smoke Detectors	*, **
3.15	Manual Hose Stations (2055-JJ, 2084-DD, Containment, Elev. 317' of Auxiliary Building	*, **
3.16	Portable Smoke Exhaust Equipment	December 1, 1978
3.17	Emergency Lighting	December 1, 1978
3.18	Reactor Coolant Pump Oil Collection System	*
3.19	Control of Fire Doors	March 31, 1979
3.20	Administrative Control Changes	December 1, 1978

(Numbers in parentheses refer to fire zone designations in the AP&L fire hazards analysis.)

*Prior to startup following the first regularly scheduled refueling outage.

**Technical Specifications covering these items should be proposed not later than 90 days prior to implementation.

(f) Overpressure Mitigating System

The licensee shall achieve full implementation of its proposed overpressure mitigating systems described in the licensee's letter dated October 11, 1977 prior to startup following the first regularly scheduled refueling outage. The system utilizes relief valves with a low pressure setpoint that will be lined up for use at low temperature and low pressure conditions

(g) Verification of Transient Analysis Code

The licensee shall complete tests to verify the use of the CESEC Code during the initial startup and power ascension testing program and submit the results for Commission review and approval.

The CESEC verification test results shall include an analysis of the uncertainties associated with the test instrumentation and a demonstration that the test instrumentation was adequate for the intended purpose.

(h) Main Feedwater System Modification

The licensee shall achieve full implementation of its proposed modifications to the main feedwater system to preclude unacceptable mass and energy blowdown into the containment in the event of a main steamline break accident prior to startup following the first regularly scheduled refueling outage. The modifications shall consist of the installation of one additional main feedwater isolation valve in each of the main feedwater lines which shall be designed to close upon receipt of a containment isolation signal. The required modifications are described in Section 6.2.1.1.2.6 of the Final Safety Analysis Report as updated through Amendment No. 45.

(i) Containment Radiation Monitor

The licensee shall, prior to Mode 2 operation, submit for the Commission's review and approval a description and analysis of the use of portable radiation monitors located outside of containment. The licensee shall, prior to the completion of the startup and power ascension testing program, submit for Commission review and subsequent approval documentation which establishes the adequacy of the qualifications of the containment radiation monitors located inside the containment to perform their intended functions.

(j) Environmental Qualifications of Safety Related Instrumentation

(1) The licensee shall, prior to Mode 2 operation submit for Commission review and approval, the results of sequential testing on the Foxboro pressurizer high pressure trip and the Foxboro high containment pressure trip transmitters which exposes the same piece of equipment to radiation, seismic and environmental effects that are calculated for the loss-of-coolant and main steamline break accident conditions at the plant and demonstrate that the equipment maintains its functional operability under these conditions.

If the Foxboro equipment cannot meet the applicable performance requirements during these environmental qualification tests, the licensee shall, prior to Mode 2 operation, replace the Foxboro equipment with other transmitters that have been demonstrated to be qualified to the the loss-of-coolant and main steamline break accident conditions to the satisfaction of the Commission.

(2) The licensee shall, prior to Mode 2 operation, submit for the Commission's review and approval the results of sequential testing on the Fischer Porter equipment for the pressurizer low pressure trip, the number one and number two steam generator low pressure trip, and the number one and number two steam generator high and low level trips. The sequential testing shall expose the same piece of equipment to radiation, seismic and environmental effects that are calculated for the loss-of-coolant and main steamline break accident conditions at the plant and shall demonstrate that the equipment maintains its functional operability under these conditions.

If the Fischer Porter equipment cannot meet the applicable performance requirements during these environmental qualification tests, the licensee shall, prior to Mode 2 operation, replace the Fischer Porter equipment with other transmitters that have been demonstrated to be qualified to the loss-of-coolant and main steamline break accident conditions to the satisfaction of the Commission.

(k) Core Protection Calculator System (CPCS)

(1) CPCS Position No. 1, Power Distribution Algorithm

The licensee shall submit for Commission review, prior to February 28, 1979, and subsequent approval, the results of startup verification testing which demonstrate the conservatism of the calculation of the power distribution uncertainty factors. The startup testing shall be performed in accordance with information previously submitted, as identified in Section D.3.5 of the Staff's Safety Evaluation Report and Supplements Number 1 and 2 thereto, by the licensee in support of the resolution of CPCS Position No. 1.

(2) CPCS Position No. 5, Cable Separation

The licensee shall submit for Commission review, prior to February 28, 1979, and subsequent approval, the results of measurements from the startup testing program which demonstrates that noise or electromagnetic interference effects from non-Class IE circuits which are in close proximity to Class IE circuits are within previously established acceptable ranges. These measurements shall be performed in accordance with information previously submitted by the licensee, as identified in Section D.4.1.2 of the Safety Evaluation Report and Supplements Number 1 and 2 thereto, in support of the resolution of CPCS Position No. 5.

(3) CPCS Position No. 12, Electrical Noise and Isolation

The licensee shall submit for Commission review, prior to February 28, 1979 and subsequent approval the results of measurements from the startup testing program which demonstrates that noise or electromagnetic interference effects upon the operation of the optical isolators are within previously established acceptable ranges. These measurements shall be performed in accordance with information previously submitted by the licensee, as identified in Sections D.4.1.4 and D.4.4.4 of Supplements Number 1 and 2 to the Safety Evaluation Report, in support of the resolution of CPCS Position No. 12.

(4) CPCS Position No. 19, Software Change Procedure Qualification

The licensee shall submit for Commission review and approval prior to Mode 2 operation additional information demonstrating that an acceptable procedure has been developed for the execution of changes to the CPCS software. The information shall consist of responses to items (1), (3) and (4) as identified in Supplement No. 2 to the Safety Evaluation Report, Section D.4.4.6, Summary Subsection.

The licensee shall submit for Commission review and approval, prior to Mode 1 operation, additional information in response to item (2) as identified in Supplement No. 2 to the Safety Evaluation Report, Section D.4.4.6, Summary Subsection.

(1) CEA Guide Tube Surveillance Program

The licensee shall, prior to startup following the first regularly scheduled refueling outage, submit for Commission review and approval the results of a surveillance program conducted on the design modifications to the control element assembly (CEA) guide tubes. The program, as discussed in Section 4.2.4 of SER Supplement No. 2, shall be directed toward determining whether unacceptable degradation of the guide tube components has occurred.

(m) Redundant Valve Position Indication

The licensee shall within six months of the issuance of this amendment to the operating license complete the implementation of modifications required to provide redundant indication in the control room of the position of the valve (valve number 2CV-5628-2) in the recirculation line from the engineered safety feature system pumps to the refueling water storage tank. The modifications shall be completed in accordance with the licensee's letter dated March 30, 1978.

(n) Fire Barrier Testing

Prior to October 31, 1978 the licensee shall submit for the Commission's review and subsequent approval a report on the results of fire testing conducted on a fire barrier containing steel conduit loaded with cables and sealed at the ends of the conduit. The test results must demonstrate the acceptability of the licensee's criteria identified in a letter to the staff dated July 26, 1978 for sealing conduit penetrations in fire barriers.

(o) Offsite Power System

Prior to February 28, 1979 the licensee shall achieve full implementation of all design modifications proposed in the letter dated March 30, 1978 in response to the matter of protection from degraded offsite power systems.

D. Arkansas Power and Light Company shall maintain in effect and fully implement all provisions of the approved physical security plan, including amendments and changes made pursuant to the authority of Section 50.54(p) of 10 CFR Part 50. The approved security plan consists of the licensee's proprietary documents, collectively entitled "Arkansas Nuclear One Industrial Security Plan," as follows: Revision 9 dated May 23, 1975 (This revision replaced the previous plan in its entirety), Revision 10 dated October 31, 1975 and Revision 12 dated June 9, 1978.

E. This license is subject to the following additional condition for the protection of the environment:

Before engaging in additional construction or operational activities which may result in an environmental impact that was not evaluated by the Commission, the licensee will prepare and record an environmental evaluation or such activity. When the evaluation indicates that such activity may result in a significant adverse environmental impact that was not evaluated, or that is significantly greater than that evaluated, in the Final Environmental Statement (NUREG-0254) or any addendum thereto, the licensee shall provide a written evaluation of such activities and obtain prior approval from the Director, Office of Nuclear Reactor Regulation.

F. This license is effective as of the date of issuance and shall expire at midnight, December 6, 2012.

FOR THE NUCLEAR REGULATORY COMMISSION

D B Vassallo / for

Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

Attachments:
Preoperational Tests, Startup Tests
and Other Items Which Must Be Completed
By the Indicated Operational Mode

Date of Issuance: SEP 1 1978

ATTACHMENT 2 TO LICENSE NPF-6

Preoperational Tests, Startup Tests, and Other Items Which Must be Completed Prior to Proceeding To Succeeding Operational Modes

This attachment identifies certain preoperational tests, startup tests, and other items which must be completed to the Commission's satisfaction prior to proceeding to certain specified Operational Modes. Arkansas Power & Light Company shall not proceed beyond the authorized Operational Modes without prior written authorization from the Commission.

A. The following items must be completed prior to proceeding to Operational Mode 2 (Initial Criticality).

1. Completion of significant startup punchlist items which affect the operability of the following:

   - Sampling System
   - Auxiliary Building H&V (1)
   - Emergency Feedwater System (2)
   - Plant Protective System (4)
   - Reactor Coolant System (3)
   - Waste Gas System (1)
   - Area Radiation Monitors (2)
   - Air & Gas Radiation Monitors (6)
   - Safety Injection System (2)
   - Liquid Radwaste System (4)

2. Completion of the following Preoperational Tests:

   2.083.01 Main Steam Supply and Safety Relief Valves

3. Closeout of outstanding Startup Program Test Deficiencies.

4. Approval and issuance of the following procedure:

   2.800.01 App. U Unit Load Transient Test

5. Resolution of main feedwater line break potential within the containment piping penetration room.

6. Resolution of the following items relating to radiation protection.

   a. Complete installation and calibration of health physics monitoring equipment.

   b. Complete calibration of area radiation monitors.

   c. Complete calibration of process radiation monitors.

7. Complete hanger installation.

8. Complete installation of independent DC power supplies to the series containment penetration breakers.

9. Resolution of discrepancies identified in the Facility Operating Procedures.

10. Resolution of test deficiencies relating to the failure of the Hydrogen Purge System to meet FSAR acceptance criterion. These deficiencies include:

    - Failure of the filters to pass the Freon-112 test.

    - Failure of the system to meet specified flow rate.

11. Resolution of LPSI Pump Motor Failure.

12. Resolution of loose part in safety injection system.

B. The following items must be completed prior to proceeding to Operational Mode 1 (Power Operation).

   1. Completion of significant startup punchlist items which affect the operability of the following:

      - Control Room H&V (1)
      - Miscellaneous H&V (1)
      - Feedwater System (1)
      - Steam Generators (2)
      - Fuel Pool and Auxiliaries (8)
      - Waste Gas System (1)
      - Solid Radiation Waste System (4)
      - Main Steam System (2)

   2. Resolution of the following outstanding operations punchlist items:

      - Instrumentation in place for CECEC Code verification

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-368

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT 2

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE
AND
GRANTING OF EXEMPTIONS FROM CERTAIN REQUIREMENTS OF SECTION 55.55a
OF 10 CFR PART 50 AND APPENDICES G, H AND J TO 10 CFR PART 50

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 1 to Facility Operating License No. NPF-6 to Arkansas Power and Light Company for operation of Arkansas Nuclear One, Unit 2 (the facility) at steady state reactor core power levels not in excess of 2815 megawatts thermal, in accordance with the provisions of the amended license and the Technical Specifications. However, the facility is temporarily restricted from operating at full rated power until certain tests and other items noted in license conditions are completed to the written satisfaction of the Commission. The facility is located at the licensee's site in Pope County, Arkansas. The amended license is effective as of its date of issuance and shall expire at midnight on December 6, 2012.

The Commission has made appropriate findings as required by the Atomic Energy Act of 1954, as amended (the Act), and the Commission's regulations in 10 CFR Chapter I, which are set forth in the amended license. The application for the license complies with the standards and requirements of the Act and the Commission's regulations.

This action is in furtherance of the licensing action encompassed in the combined Notice of Receipt of Application for Facility Operating License; Notice of Availability of Applicant's Environmental Report; and Notice of Opportunity for Hearing published in the FEDERAL REGISTER on April 23, 1974 (39 F.R. 14371).

The Commission has determined that the facility requires exemptions from certain requirements of (1) Section 50.55a of 10 CFR Part 50, (2) Appendices G and H to 10 CFR Part 50, and (3) Appendix J to 10 CFR Part 50 for a period of three years. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. In making this determination, the Commission has given due consideration to the burden that could result if these requirements were imposed on the facility.

The exemption from certain requirements of Section 50.55a of 10 CFR Part 50 relates to the preservice and inservice examination requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the ASME Code) for pump scroll welds, nozzle welds, longitudinal and circumferential vessel welds, integrally welded supports, pump studs, vessel cladding, valve bodies, circumferential and longitudinal piping welds, calibration blocks and testing requirements.

The Commission has evaluated the preservice examination program for Arkansas Nuclear One, Unit 2 and has determined that a large portion of the ASME Code required preservice examinations were performed, and has concluded that failure to perform 100 percent preservice examination of the welds identified in the exemption will not significantly affect the assurance of the initial system integrity or the ability to subsequently detect and correct service-induced defects.

The exemptions from certain requirements of Appendices G and H to 10 CFR Part 50 relates to the fracture toughness requirements for the ferritic materials of the pressure-retaining components of the reactor coolant pressure boundary and the reactor vessel material surveillance program.

Although the Arkansas Nuclear One, Unit 2 reactor vessel was ordered and fabricated and its testing program was developed well before the requirements of Appendices G and H to 10 CFR Part 50 became effective, alternative methods for providing adequate margins of safety against brittle fracture and an alternative material surveillance program were proposed.

The Commission has evaluated the alternative methods for providing adequate margins of safety against brittle fracture and the alternative material surveillance program and has concluded that they are acceptable.

The exemption, which shall be granted for a period of three years, from certain requirements of Appendix J to 10 CFR Part 50 relates to the leakage testing requirements for the containment airlocks. Implementation of the specific Appendix J requirements would result in the testing of the entire containment airlock volume after each entry. The Commission has determined that the testing of the airlock door seals at least once each seventy-two hours for multiple entries occurring during that interval will provide adequate assurance of the leak tightness of the containment airlock including the containment airlock door seals.

The Commission has determined that the granting of these exemptions and the issuance of this amendment will not result in any significant environmental impact and that pursuant to Section 51.5(d)(4) of 10 CFR Part 51, an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared.

For further details with respect to this action, see (1) Amendment No. 1 to Facility Operating License No. NPF-6 complete with Preoperational Tests and Other Items Which Must Be Completed Prior to Loading Fuel (Attachment 1); and (2) the Commission's related Safety Evaluation supporting Amendment No. 1 to License No. NPF-6; and (3) the Office of Nuclear Reactor Regulation's Safety Evaluation Report (NUREG-0308) Supplement No. 2 dated September 1978. All of these items are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. 20555 and the Arkansas Polytechnic College, Russellville, Arkansas 72801.

A copy of Items (1) and (2) may be obtained upon request addressed to the United States Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Divison of Project Management. Copies of the Safety Evaluation Report Supplement No. 2 (NUREG-0308) may be purchased at current rates from the National Technical Information Service, Department of Commerce, 5285 Port Royal Road, Springfield, Virginia 22151.

Dated at Bethesda, Maryland, this 1st day of September 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

John Angelo

John Angelo, Acting Chief
Light Water Reactors Branch No. 1
Division of Project Management

SEP 1 1978

SAFETY EVALUATION
BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO. 1

(Arkansas Power and Light Company)

A. Evaluation Concerning Environmental Testing of Foxboro and Fischer Porter Transmitters

In Supplement No. 2 to the Arkansas Nuclear One - Unit 2 (ANO-2) Safety Evaluation Report, we provided our evaluation of previously identified outstanding review items. The resolution of or the basis for authorization of plant operation in operational Modes 2 and 1 is provided in Supplement No. 2 for each of these items with one exception. As noted in Section 3.11 of Supplement No. 2 we conclude that certain instruments were not adequately qualified for the loss-of-coolant accident environment. The following evaluation is in support of issuance of authorization for operation of the ANO-2 plant in operational modes 4 (Hot Shutdown) and 3 (Hot Standby) while the resolution to this matter is being achieved. We have informed the licensee that as a condition of the license, they will be required to complete a properly conducted test which demonstrates that these instruments are acceptably qualified. The instruments involved are the:

1. Foxboro E11 series utilized for containment high pressure and the pressurizer high pressure trip transmitters.

2. Fischer Porter 50EP1041 which is utilized for the following: (1) pressurizer low pressure trip, and (2) steam generator low pressure trip.

3. Fischer Porter 13D2495 for steam generator high and low water level trip and pressurizer level control.

The licensee has informed the staff by letters dated August 8, 1978 and August 13, 1978 that he has entered into an arrangement to have the necessary verification testing performed for the Foxboro and Fischer Porter Transmitters.

The additional verification testing of the Foxboro and Fischer Porter Transmitters is now anticipated to provide the required test data by the middle of September 1978. We understand that at that time sufficient information will be available to determine whether the required tests have been successfully completed and that a final report can be provided by October 31, 1978.

Since our review of the results of these verification tests will not be completed at the time the plant is currently scheduled to be prepared for entry into Modes 4 (Hot Shutdown) and 3 (Hot Standby) we have evaluated the information provided by the licensee to determine whether operation in Mode 4 and 3 conditions during the time interval required to perform the tests, report the results, and conduct our evaluation would be acceptable. The purpose of the required verification test is to confirm by sequential testing that the Foxboro and Fischer Porter Transmitters presently installed in the ANO-2 plant can conservatively perform their design requirements with margin. The staff has evaluated the information presented in the licensee's letters to the staff dated May 17, 1978, August 8, 1978 and August 15, 1978. The Foxboro and Fischer Porter Transmitters previously tested, which were identical to those installed in the plant, demonstrated acceptable results when exposed to pressure, and temperature environments more severe than those that would result from any design basis event. The radiation testing of the electronics performed for these transmitters was completed in other separate tests on different instruments of the same type which demonstrated acceptable results at integrated radiation exposures higher than those that would result from forty-year integrated dose levels plus the radiation from any design basis event.

The present core loading of ANO-2 consists entirely of new fuel. Operational Mode 3 of the Technical Specifications requires a reactivity condition K_{eff} less than 0.99, zero percent of rated power, and an average coolant temperature greater than or equal to 300 degrees Fahrenheit.

In the highly unlikely event of a postulated loss-of-coolant accident, an inconsequential amount of decay heat would result solely from the spontaneous natural decay of the fuel. No forced cooling of the fuel would be necessary to prevent exceeding the fuel clad temperature and other requirements of Section 50.46 and Appendix K to 10 CFR Part 50.

Until the core is made critical and operated at power, there will be no significant increase in the decay heat above that generated in the new fuel. The Technical Specifications contain restrictions during Hot Shutdown and Hot Standby conditions which will prevent achieving criticality even in the event of an operator error or equipment malfunction. Therefore, any radioactive release into the containment would be insignificant for a postulated loss-of-coolant accident.

We have evaluated the issue of environmental qualifications for the subject components and for the reasons stated above have concluded that the operation of the ANO-2 plant in Modes 4 (Hot Shutdown) and 3 (Hot Standby), while completing the sequential verification testing, will not result in a significant risk to the health and safety of the public.

We shall require that the licensee provide the preliminary results of the confirmatory tests for our review and approval prior to Mode 2 operation and a final report by October 31, 1978. The final acceptance of these transmitters will be addressed by the staff after completion of the qualification testing and the review of the final test program report which is to be supplied by October 31, 1978.

B. Evaluation Concerning Environmental Testing of Containment Radiation Monitors

In a letter dated June 23, 1978, the licensee informed us that the radiation monitors located inside containment for post accident monitoring (PAM) purposes had not yet been qualified to the environmental qualification requirements of IEEE Standard 323-1971. The licensee further stated that they expected to have the qualification test completed by August 1978.

Subsequently, the licensee advised us in a letter dated August 31, 1978, that previously unforseen delays had occurred and the testing program would not now be expected to be complete until November 1978. This would mean that the staff's requirement, as included in the facility Technical Specifications, for the assured capability to monitor the post-accident radiation levels inside the containment could not be met by reliance on the presently installed Victoreen radiation monitors inside the containment. In lieu of reliance on these monitors the licensee proposed the monitoring of post accident radiation levels inside the containment by using a portable radiation detector located outside of the containment. Analyses have been performed by the licensee to enable the correlation of these detection readings with the post-accident radiation levels inside the containment.

We have evaluated this matter and conclude that an exception to Technical Specification Section 3.3.3.6 for Mode 3 is required and justified to enable the procession of the ANO-2 plant into Mode 3 (Hot Standby).

Our basis for the granting of this exemption is as follows:

The present core loading of ANO-2 consists entirely of new fuel. Operational Mode 3 of the Technical Specifications requires a reactivity condition K_{eff} less than 0.99, zero percent of rated power, and an average coolant temperature greater than or equal to 300 degrees Fahrenheit. In the highly unlikly event of a postulated loss-of-coolant accident, an inconsequential amount of decay heat would result solely from the spontaneous natural decay of the fuel. No forced cooling of the fuel would be necessary to prevent exceeding the fuel clad temperature and other requirements of

Section 50.46 and Appendix K to 10 CFR Part 50. Until the core is made critical and operated at power, there will be no significant increase in the decay heat above that generated in the new fuel. The Technical Specifications contain restrictions during Hot Shutdown and Hot Standby conditions which will prevent achieving criticality even in the event of an operator error or equipment malfunction. Therefore, any radioactive release into the containment would be insignificant for a postulated loss-of-coolant accident.

Therefore, in consideration of the insignificant amount of radiation which could be released into the containment from a loss-of-coolant accident and subject to the availability of the portable radiation monitor and the accompanying procedures for its use we conclude that an exception from the operability requirements of Technical Specification Section 3.3.3.6 for the containment radiation monitors during operational Mode 3 is required and justified and accordingly is granted as stated in Section 2.C.2 of Amendment No. 1 to Facility Operating License NPF-6.

We shall require that the license provide further details regarding the equipment, the analysis performed to determine the correlation of the portable detector to the in-containment radiation source and the plant operating procedure for the use of the detector from our review and approval prior to Mode 2 operations. We shall also require, prior to completion of the initial startup and power ascension test program, the licensee to submit the documentation which establishes the adequacy of the in-containment radiation monitors to perform their design requirements with margin.

SEPTEMBER 1978

SAFETY EVALUATIONS IN SUPPORT OF EXEMPTIONS FROM CERTAIN REQUIREMENTS OF THE COMMISSION'S RULES AND REGULATIONS

BY THE

OFFICE OF NUCLEAR REACTOR REGULATION

U. S. NUCLEAR REGULATORY COMMISSION

IN THE MATTER OF

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE - UNIT 2

DOCKET NO. 50-368

SEP 1 1978

SAFETY EVALUATIONS IN SUPPORT OF EXEMPTIONS FROM CERTAIN REQUIREMENTS OF THE COMMISSION'S RULES AND REGULATIONS

We have determined that the Arkansas Nuclear One - Unit No. 2 plant requires exemptions from certain requirements of (1) Section 50.55a and (2) Appendices G and H to 10 CFR Part 50 and Appendix J to 10 CFR Part 50. These exemptions are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest. Our safety evaluations supporting the granting of these exemptions are contained herein.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF APPENDIX J TO 10 CFR PART 50

I. INTRODUCTION

We have determined that the radiological safety technical specifications for the Arkansas Nuclear One - Unit No. 2 plant, Appendix A to Facility Operating License No. NPF-6, contain certain provisions concerning the primary containment air locks which do not meet certain explicit requirements of Appendix J to 10 CFR Part 50. We have also determined, however, that the ANO-2 technical specifications contain acceptable alternatives to those provisions which do not meet the explicit requirements of Appendix J to 10 CFR Part 50. We, therefore, conclude that an exemption from certain explicit requirements of Appendix J to 10 CFR Part 50 is required and justified and, accordingly, shall be granted for a three year period. The bases for our conclusions are discussed in the following sections.

II. EVALUATION

Primary Containment Air Locks

Paragraph III.D.2 of Appendix J to 10 CFR Part 50 requires in part that primary containment air locks be tested at six-month intervals and after each opening. Paragraph III.B.2 of Appendix J to 10 CFR Part 50 requires that these tests be performed at a pressure not less than the calculated peak containment internal pressure related to the design basis accident (52.8 pounds per square inch guage for ANO-2). Specification 4.6.1.3 of the ANO-2 technical specifications, however, provides that (1) the primary containment air lock door seals be leak tested after each opening except when the air lock is being used for multiple entries, then at least once every 72 hours, and (2) these tests verify door seal leakage no greater than 175 cubic centimeters per minute when the gap between the door seals is pressurized to at least fifty-four pounds per square inch gauge for at least 15 minutes.

Based on plant operating experience, the leakage testing of primary containment air locks after each opening as required by Paragraph III.D.2 of Appendix J to 10 CFR Part 50, when frequent air lock usage is necessitated over a short period of time is, in our judgment, impractical and unnecessary to assure the maintenance of the leaktight integrity of the air locks. It is our judgment that the verification of less than 175 cubic centimeters per minute door seal leakage when the gap between the door seals is pressurized to at least fifty-four pounds per square inch guage for at least 15 minutes at least once every 72 hours when the air locks are being used for multiple entries, as provided in the ANO-2 technical specifications, provides the required assurance that the leaktight integrity of the air locks are maintained. The ANO-2 technical specifications provide that the air locks be tested at six month intervals at the test pressure of 54.0 pounds per square inch guage as required by Paragraphs III.D.2 and III.B.2 of Appendix J to 10 CFR Part 50. These latter tests involve pressurization of the entire air lock instead of just the gap between the door seals.

We, therefore, conclude that the methods for leakage testing the primary containment air locks provided by the ANO-2 technical specifications represent acceptable alternatives to those required by Appendix J to 10 CFR Part 50.

III. PUBLIC INTEREST CONSIDERATION

To require specific conformance with the applicable requirements of Appendix J to 10 CFR Part 50 for the leakage testing of the primary containment air locks would necessitate after each opening (1) the installation of retainer clips on the interior air lock door (since the interior air lock door is designed to resist internal containment pressure), (2) the pressurization of the entire air lock to a pressure of 54.0 pounds per square inch guage, (3) the determination of the leakage rate of the air lock, (4) the depressurization of the air lock, and (5) the removal of the retainer clips from the interior air lock door. The licensee estimates, based on previous experience, that approximately 8 hours would be required to perform this operation as opposed to approximately 15 minutes to pressurize the air gap between the door seals to a pressure of at least fifty-four pounds per square inch guage for at least 15 minutes at least once every 72 hours when the air locks are being used for multiple entries. It is our judgment that to require that the entire air lock be leakage tested at a pressure of 54.0 pounds per square inch guage after each opening when frequent air lock usage is necessitated over a short period of time would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

We, therefore, conclude that the public interest is served by not imposing the applicable requirements of Appendix J to 10 CFR Part 50 since such an imposition would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

IV. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to Section 50.12 of 10 CFR Part 50, a specific exemption for a period of three years as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption 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. We have concluded that this exemption would be insignificant from the standpoint of environmental impact and pursuant to Paragraph (d)(4) of Section 51.5 of 10 CFR Part 51 that an environmental impact statement, or negative declaration and environmental impact appraisal, need not be prepared in connection with this action.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF APPENDICES G AND H TO 10 CFR PART 50

I. INTRODUCTION

Arkansas Power and Light Company provided information in support of the staff's evaluation of their method of compliance with 10 CFR Part 50, Appendices G and H, in a letter dated June 14, 1978. As a result of our review of this information, we have determined that exemptions to 10 CFR Part 50, Appendices G and H are required and have also determined that exemptions regarding this matter are justified. Our bases for this conclusion are discussed in the subsequent paragraphs of this report.

II. TECHNICAL EVALUATION CONSIDERATIONS

A. The objective of Appendix G is to specify minimum fracture toughness requirements for ferritic materials of pressure-retaining components of the reactor coolant pressure boundary of water cooled power reactors to provide adequate margins of safety during any condition of normal operation, including anticipated operational occurrences and system hydrostatic tests to which the pressure boundary may be subjected over its service lifetime. Specimens of the material of fabrication are required to be tested and the data used to develop safe operating condition limits for the reactor pressure vessel.

The objective of Appendix H is to monitor the change in fracture toughness properties of ferritic materials in the reactor vessel beltline region of water cooled power reactors resulting from exposure to neutron irradiation and the thermal environment. Under this program, fracture toughness test data are obtained from material specimens placed in the vessel before operation and withdrawn periodically during operation and tested to obtain fracture toughness data. These data permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life.

The bulk of the detailed procedures and practices to be followed are given by way of reference to the ASME Code and ASTM Standards.

B In the following evaluation the staff has considered each area of possible variance with the regulations of Appendices G and H, as listed by Arkansas Power and Light Company, and have assessed the importance of those variances on the fulfillment of the safety objective of the regulations, as well as the feasibility of requiring absolute compliance with the regulations.

III. EXEMPTIONS REQUIRED

We have reviewed the information submitted by the Arkansas Power and Light Company related to their method of compliance with 10 CFR Part 50, Appendices G and H. Based on this information and our review of the design, geometry, and materials of construction of the components, the requirement to comply with certain provisions of 10 CFR Part 50, Appendices G and H, has been determined to result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Therefore, pursuant to 10 CFR Section 50.12 specific exemption for those requirements is justified as follows:

A. 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements."

Exemption Required: Arkansas Power and Light Company has addressed the areas in which the Arkansas Nuclear One - Unit 2 (ANO-2) plant is in non-compliance with certain requirements of 10 CFR Part 50, Appendix G. Based on our evaluation of this information we have determined that an exemption is required to enable the substitution of an alternative method of compliance with the requirements of 10 CFR Part 50, Appendix G.

Reason for Request: Based on our evaluation of the information provided by the Arkansas Power and Light Company we have determined that the requirements of Appendix G of 10 CFR Part 50 have been met except for the following:

Item 1: Section III.C of Appendix G is not complied with to the extent that only base plate and representative welds in the beltline region were considered for the required testing of beltline materials. The reactor vessel beltline as used in Section III.C is defined by Section II.H of Appendix G to include the base plate, representative welds and weld heat affected zone (HAZ) material.

Item 2: Section III.C.2 of Appendix G is not complied with to the extent that excess material for test specimen weldment is not necessarily from the actual production plate, although it is from the same P-number.

Bases and Conclusions:

Item 1: The ANO-2 material testing program was formulated in accordance with Section III the 1968 Edition of the Code through the Summer 1970 Addenda, Paragraph N-330, which did not require the inclusion of samples from the Heat Affected Zone (HAZ) of the beltline weldments in the subject testing program.

Paragraph III.C of Appendix G requires that test specimens are to be taken from the HAZ material in the vessel beltline. This would require a consideration of testing HAZ specimens representative of each plate/weld combination in the vessel beltline. This was not done for the ANO-2 reactor vessel since, as stated above, Appendix G was not effective at the time of fabrication of the vessel. However, Arkansas Power and Light Company has stated that the ANO-2 baseline surveillance program, as outlined in Table 5.2-16 of the FSAR, did include HAZ material from the plate in the beltline with the most limiting properties. The results from this program indicate that, while the precise requirements of Appendix G have not been complied with, the safety objective of Appendix G has been met.

We have evaluated the data presented in Table 5.2-16 of the FSAR and based on the results of our evaluation we have determined that the objective of Appendix G, as cited above, has been met.

Item 2: The ANO-2 reactor vessel was designed and constructed in accordance with the requirements of Section III of the 1968 Edition of the Code, through the Summer 1970 Addenda. Paragraph N-521 of Section III of this Edition and Addenda of the Code requires that welding processes used in the construction of pressure-containing welds be restricted to processes which are capable of producing welds in accordance with the welding procedure qualification requirements of Section IX of the Code.

Section IX of the Code permitted all base metals with the same P-Number (a grouping of base metals that have similar characteristics, i.e., composition, mechanical properties and weldability) that are joined by the same welding methods, using the same filler metal and flux, to be qualified by a single weld qualification procedure. Part of this procedure was an acceptance standard based on impact testing, used to characterize the fracture toughness properties of the weldment. It was from this welding qualification procedure, as required by the applicable Code, that the ANO-2 weldment test specimens, to satisfy the requirments of paragraph III.C.2 of Appendix G, were fabricated.

Paragraph III.C.2 of Appendix G, which came into effect after the testing program for ANO-2 was formulated, requires that the weldment test specimen be taken directly from excess weld material of the reactor vessel or from a weld fabricated from excess material of the reactor vessel shell course. The intent of this requirement is that impact testing used to characterize the fracture toughness properties of the beltline region of the reactor vessel be as representative of the actual properties as possible.

For similar base metals, the results of impact testing performed on weldment test specimens are more a function of the welding process and weld metal-flux compositions than of the toughness properties of the base metal. Therefore, the use of the Section IX weldment test specimens satisfy the intent of the requirements of Appendix G.

In addition to the testing program required by Section IX, the Arkansas Power and Light Company has stated that the ANO-2 baseline surveillance program, required by Appendix H of 10 CFR Part 50, did include some weld material fabricated from excess material of the reactor vessel shell course (weld 2-203A, plates C-8009-1/C-8009-2) as outlined in Table 5.2-6 of the FSAR. While tests for all beltline welds were not performed, this weld material was obtained, fabricated into test specimens, tested and met the fracture toughness requirements of Appendix G, including the specific requirements of paragraph III.C.2.

The results of both the Section IX and the baseline surveillance testing programs indicate that, while the precise requirements of Appendix G have not been complied with, the safety objective of Appendix G has been met.

We have evaluated the information supplied by Arkansas Power and Light Company and based on the results of our evaluation we have determined that the objectives of Appendix G, as cited above, has been met.

B. 10 CFR Part 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements"

Exemption Required: Arkansas Power and Light Company has addressed the areas in which the ANO-2 plant is in non-compliance with a requirement of 10 CFR Part 50, Appendix H. Based on our evaluation of this information we have determined that an exemption is required to enable the substitution of an alternative method of compliance with the requirements of 10 CFR Part 50, Appendix H.

Reason for Request: Arkansas Power and Light Company has stated that the reactor vessel material surveillance program meets the requirements of 10 CFR Part 50 Appendix H with the exception of Section II.C.2 in that surveillance specimen capsules are attached to the cladding on the inside of the reactor vessel in the beltline region. The Combustion Engineering, Inc. (C-E) Topical Report CENPD-155P, "C-E Procedure for Design, Fabrication, Installation and Inspection of Surveillance Specimen Holder Assemblies," dated September 1974, presents C-E's position and a discussion of the bases for the attachment of the capsules to the vessel cladding.

Bases and Conclusions: Paragraph II.C.2 of Appendix H to 10 CFR Part 50 states that surveillance capsules containing the surveillance specimens shall be located near but not attached to the inside vessel wall in the beltline region, 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, and the thermal environment is as close as practical to that of the vessel inner surface. By attaching the surveillance holder assemblies to the reactor vessel cladding, C-E does not strictly comply with Appendix H.

The intent of this section of Appendix H is to avoid any vessel material degradation resulting from the attachment welds, avoid interference with inservice inspections, and eliminate the possibility that the attachment brackets might cause unacceptable local stresses in the vessel wall. Based on these considerations, we have evaluated capsule attachment and conclude that attachment to the vessel cladding is acceptable provided the following precautions are taken:

1. Weld procedures are stringently controlled to ensure that the HAZ does not encompass any vessel base metal.

2. The capsule holder assembly will be designed so that it will not interfere with inservice inspections required by Section XI of the ASME Code.

3. A stress analysis is made to demonstrate that the attachment fixture does not impair the structural integrity of the vessel.

In a letter from the staff to Combustion Engineering, Inc. (O. D. Parr to F. M. Stern, dated May 15, 1975) the staff stated that the proposed attachment method, described in CENPD-155P, was carefully reviewed and concluded that it meets the above stated precautions and does not cause any degradation of the base material, prevent inservice inspections nor produce any unacceptable loads in the reactor vessel.

We have reviewed the attachment of the surveillance specimen holder assemblies for ANO-2 and have determined that the recommended precautions were followed and have been found to be acceptable, thus, fulfilling the objective of Appendix H.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH 10 CFR PART 50, APPENDICES G AND H

Our technical evaluation has not identified any practical method by which the existing ANO-2 reactor vessel can meet the specific requirements of 10 CFR Part 50, Appendices G and H. Requiring specific compliance with these Appendices would delay the startup of the plant due to the need to complete the following actions: (1) obtain, if possible, sufficient material from the actual ANO-2 beltline plates to fabricate heat affected zone specimens for the remaining plate/weld combinations, (2) fabricate and test the weldment test specimens, and (3) remove and relocate the installed material surveillance capsules.

We believe the public interest is served by not imposing certain provisions of 10 CFR Part 50, Appendices G and H, that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

V. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to 10 CFR Section 50.12, a specific exemption as discussed above is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility.

Furthermore, we have determined that the granting of this exemption 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. We have concluded that this exemption would be 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 this action.

SAFETY EVALUATION IN SUPPORT OF AN EXEMPTION FROM CERTAIN REQUIREMENTS OF SECTION 50.55a(g)(2) OF 10 CFR PART 50 AND IN SUPPORT OF RELIEF FROM CERTAIN REQUIREMENTS OF SECTION 50.55a(g)(4) OF 10 CFR PART 50

I. INTRODUCTION

Arkansas Power and Light Company provided information in support of the staff's evaluation of their method of compliance with 10 CFR Part 50.55a "Codes and Standards" in letters dated March 24, 1978 and June 13, 1978 for the Arkansas Nuclear One-Unit 2 Plant. As a result of our review of this information, we have determined that an exemption from certain preservice inspection requirements of 10 CFR 50.55a "Codes and Standards" is required and have also determined that an exemption regarding this matter is justified. Also as a result of our review of this information, we have determined that, pursuant to 10 CFR 50.55a(g)(6), relief from certain inservice inspection requirements of 10 CFR 50.55a is required and have also determined that relief regarding this matter is justified. Our bases for these conclusions are discussed in the subsequent paragraphs of this report.

PRESERVICE

For nuclear power facilities whose construction permits were issued on or after January 1, 1971, but before July 1, 1974, 10 CFR 50.55a(g)(2) specifies that components shall meet the preservice examination requirements set forth in editions of Section XI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code and Addenda in effect six months prior to the date of the issuance of the construction permit. The provisions of 10 CFR 50.55a(g)(2) also state that components (including supports) may meet the requirements set forth in subsequent editions of this code and addenda which become effective.

Therefore, our evaluation consisted of determining the areas where ANO-2 met 10 CFR 50.55a(g)(2) requirements and the areas where exemptions to the regulation were necessary and the basis for these exemptions.

INSERVICE

For nuclear power facilities whose construction permits were issued on or after January 1, 1971, but before July 1, 1974, 10 CFR 50.55a(g)(4) specifies that throughout the service life of a facility components shall meet the requirements set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code and Addenda that become effective subsequent to editions specified in Paragraph 50.55a(g)(2) which are incorporated by reference in Paragraph 50.55a(b). Paragraph 50.55a(g)(4) further states that the initial inservice examinations conducted during the first 40 month period shall comply with the editions of the Code and Addenda in effect no more than six months prior to the date of start of facility commercial operation.

Therefore, our evaluation consisted of determining the areas where ANO-2 met 10 CFR 50.55a(g)(4) requirements and the areas where, pursuant to 10 CFR 50.55a(g)(6), relief from these requirements is necessary and the basis for the relief granted.

II. TECHNICAL EVALUATION CONSIDERATIONS

A. The Arkansas Nuclear One, Unit No. 2 Plant, received a Construction Permit on December 6, 1972. In accordance with 10 CFR 50.55a, the preservice inspection must conform with the ASME Code, Section XI, 1971 Edition, including Addenda through Summer 1971 and may, if the licensee elects, meet later Editions and Addenda which become effective. In accordance with 50.55a, the inservice inspection must conform with the ASME Code, Section XI, 1974 Edition, including Addenda through Summer 1975. The ASME first published rules for inservice inspection in the 1970 Edition of Section XI. No preservice or inservice inspection requirements existed prior to that date. Since the ANO-2 plant system design and ordering of long lead time components were well underway by the time the Section XI rules became effective, full compliance with the access and inspectability requirements was difficult to achieve. As can be seen in Section III below, which discusses individual welds or examination categories, a large portion of the required volumetric examinations were performed.

B. Verification of as-built structural integrity of the primary pressure boundary is not dependent on the Section XI preservice examination. The applicable construction codes to which the ANO-2 primary pressure boundary was fabricated, contain examination and testing requirements which by themselves provide the necessary assurance that the pressure boundary components are capable of performing safely under all operating conditions and postulated accidents reviewed in the FSAR and described in the plant design specification. As a part of these examinations the primary pressure boundary full penetration welds were volumetrically inspected (radiographed) and the system was subjected to hydrostatic pressure tests.

C. The intent of a preservice examination is to establish a reference or base line prior to the initial operation of the facility. The results of subsequent inservice examination can then be compared to the original condition to determine if changes have occurred. If review of the inservice inspection results show no change from the original condition no action is required. In the case where base line data are not available, all indications must be treated as new indications and disposed of accordingly. Section XI of the

ASME Code contains acceptance standards which are used as the basis for evaluating the acceptability of such indications. Therefore, conservative disposition of defects found during inservice inspection can be accomplished even though preservice information is not available.

D. Other benefits of preservice examination include providing redundant or alternate volumetric inspection of the primary pressure boundary using a test method different from that employed during the component fabrication thereby increasing the overall probability of finding all significant fabrication flaws. Successful performance of a preservice examination also demonstrates that the welds so examined are capable of subsequent inservice examination using a similar test method.

In the case of ANO-2, a large portion of the code required preservice examinations were performed. We have concluded that failure to perform 100% preservice examination of the welds specifically identified below will not significantly affect the assurance of the initial system integrity or the ability to subsequently detect and correct service induced defects.

E. In some instance where the required preservice examinations were not performed to the full extent specified by the applicable ASME Code, we will require that these or supplemental examinations be conducted as a part of the inservice inspection program. We have concluded that requiring these supplemental examinations to be performed at this time (before plant startup) would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Acceptable preoperational integrity has already been established by similar Section III fabrication examinations and the probability of system degradation between these examinations and initial plant startup is small.

In cases where parts of the required examination areas cannot be effectively examined because of a combination of component design/ current inspection technique limitations, we will continue to evaluate the development of new or improved volumetric examination techniques. As improvements in these areas are achieved, we will require, pursuant to 10 CFR 50.55a(g)(4), that these new techniques be made a part of the inservice examination requirements of those components or welds which received a limited preservice examniation.

F. PRESERVICE

The FSAR and the referenced letters from the licensee contain information on the preservice examination of Code Class 1 and 2 components. For ANO-2, 10 CFR 50.55a(g)(2) requires that the preservice examination conform with Section XI, 1971 Edition through Summer 1971 Addenda and provides that the components may meet the requirements set forth in later editions of this code and addenda which become effective. Accordingly, the licensee has chosen to meet the 1971 Edition of the Code through the 1973 Summer Addenda. While not all the specific examinations have been conducted, for the reasons set forth above, those examinations performed provide an adequate level of assurance of the preservice structural integrity and the ability to subsequently detect and correct service-induced defects.

INSERVICE

The FSAR and the referenced letters from the licensee contain information on the inservice examination of Code Class 1 and 2 components. For ANO-2 10 CFR 50.55a(g)(4) requires that the inservice examination conform with Section XI, 1974 Edition through Summer 1975 Addenda. While not all the specific examinations are to be conducted for the reasons set forth above, those examinations to be performed will provide an adequate level of assurance of the inservice structural integrity and the ability to detect and correct service-induced defects.

III. EXEMPTIONS REQUIRED

We have reviewed the information submitted by the Arkansas Power & Light Company related to the preservice examination of the ANO-2 Plant. Based on this information and our review of the design, geometry, and materials of construction of the components, certain preservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI, have been determined to be either impractical or would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety as provided in 10 CFR 50.55a.

Therefore, pursuant to 10 CFR Section 50.12 specific exemption for those preservice requirements is justified as follows:

We have also reviewed the information provided by the Arkansas Power & Light Company related to the inservice examination of the ANO-2 plant. Based on this information and our review of the design, geometry, and materials of construction of the components, certain inservice requirements of the ASME Boiler and Pressure Vessel Code, Section XI, have been determined to be either impractical or would result in hardships or unusual difficulties without a compensating increase in the level of quality and safety as provided in 10 CFR 50.55a.

Therefore, pursuant to 10 CFR Section 50.55a(g)(6) specific relief for those inservice requirements is justified as follows:

1. COMPONENT: Reactor Coolant Pump

   a. CODE CLASS: I

   b. PRESERVICE CODE SECTION: IS-251-L-1, 1973 Summer Addenda, 1971 Code

   c. INSERVICE CODE SECTION: IWB-2500 B-L-I 1975 Summer Addenda, 1974 Code

   d. CODE REQUIREMENT: 100 percent of the pressure-containing welds in the reactor coolant pumps are to be volumetrically examined.

   e. RELIEF REQUEST: Preservice and Inservice Program. An exemption is requested on the preservice and relief is requested on the inservice from performing 100 percent of the code volumetric examination requirements on the reactor coolant pump scroll welds.

   f. REASON FOR REQUEST: Present day volumetric techniques, ultrasonic testing or radiographic testing, (UT or RT) and procedures are not amenable to field preservice volumetric examination of the scroll welds.

   g. BASES AND CONCLUSION: AP&L Co. provides the following bases: (A) Original fabrication examination: 100% radiographic examination.* (B) Extent of preservice examination: 100% based on NRC acceptance of shop fabrication records.* (C) Measures for accessibility: None. (D) Conclusion: Techniques for conducting inservice (field) volumetric examination of the subject welds are under development. Examination of the scroll weld is planned at or near the end of the first inspection interval as required by Section XI based on developing a technique by then.

*The original fabrication examination and preservice examination referred to in this item and later items in this evaluation refers to the examinations made pursuant to the manufacturing requirements for these components as set forth in the Code and not the preservice examinations performed pursuant to the requirements of Section XI of the Code.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that the capability to perform a meaningful preservice volumetric examination is currently unavailable. We conclude that for the reactor coolant pump scroll welds that (1) the construction code radiographic examination, and (2) the shop fabrication examinations, provides an adequate level of assurance of structural integrity. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

We conclude that the present day volumetric techniques are not amenable to inservice volumetric examination. In lieu of this inspection requirement we require that surface and visual inspections be performed on these welds. We conclude that this supplemental surface and visual inspection will provide an adequate level of assurance of structural integity. Therefore, the requested relief from the specific inservice requirement of Section XI is granted.

2. COMPONENT: Reactor Coolant Piping Branch Nozzles

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-251-J-1, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWB-2500-B-J 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Piping branch nozzle connections are to be volumetrically examined on both sides of the weld, nozzle side and base metal on the main coolant piping (MCP) side

e. EXEMPTION REQUEST: An exemption is requested on the preservice and relief is requested on the inservice volumetric examinations of the nozzle side of all branch piping nozzles welded to the main coolant piping

f. REASON FOR REQUEST: Access from the nozzle side of these welds is restricted by weld and nozzle configurations and is not amenable to UT.

g. BASES AND CONCLUSION: AP&LCo proivides the following bases: (A) Original fabrication examination: 100% radiographic examination. (B) Extent of preservice examination: essentially 100% based on shop and field examinations. (C) Measures for accessibility: None. (D) Conclusion: Examinations in the field during inservice will include all of the weld and base material on the MCP side.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that these nozzles are not readily accessible for the purposes of ultrasonically examining them from the nozzle side of the weld. We conclude that (1) the construction code radiographic examination, (2) the shop fabrication examinations, and (3) examination of these nozzle connections on the main coolant piping side of the weld provides an adequate level of assurance of structural integrity. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

Inservice

We conclude that these nozzles are not readily accessible for the purposes of ultrasonically examining them from the nozzle side of the weld. In lieu of this inspection requirement we require that surface and visual inspections be performed on these welds. We conclude that (1) the examination of these welds from the main coolant piping side in accordance with Section XI, and (2) the supplemental surface and visual inspection will provide an adequate level of assurance of structural integrity. Therefore, the requested relief from the specific inservice requirement of Section XI is granted.

3. COMPONENT: Pressurizer and Steam Generator Nozzle-to-Vessel Welds

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-251-D, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWB-2500-B-D, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Nozzle-to-Vessel welds are to be volumetrically examined on both sides of the weld, nozzle side and vessel side.

e. EXEMPTION REQUEST: An exemption is requested on the preservice and relief is requested on the inservice examinations of the nozzle side of the pressurizer and steam generator nozzles.

f. REASON FOR REQUEST: Access from the nozzle side of these welds is restricted by weld and nozzle configurations and is not amenable to UT.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% radiographic examination. (B) Extent of preservice examination: essentially 100% based on ship and field examinations. (C) Measures for accessibility: None. (D) Conclusion: Examinations in the field will include all of the weld and base material on the vessel side of the weld.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that these nozzles are not readily accessible for the purposes of ultrasonically examining them from the nozzle side of the weld. We conclude that (1) the construction code radiographic examination, (2) the shop fabrication examinations and (3) examination of these nozzle connections on the vessel side of the weld provides an adequate level of assurance of structural integrity. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

We conclude that these nozzles are not readily accessible for the purposes of ultrasonically examining them from the nozzle side of the weld. In lieu of this inspection requirement we require that surface and visual inspections be performed on these welds. We conclude that (1) the examination of these welds from the vessel side in accordance with Section XI, and (2) the supplemental surface and visual inspection will provide an adequate level of assurance of structural integrity. Therefore, the requested relief from the specific inservice requirement of Section XI is granted.

4. COMPONENT: Reactor Vessel

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-2510B, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWB-2500-B-B, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Volumetric examination of longitudinal welds in the upper shell course section is required.

e. EXEMPTION REQUEST: An exemption is requested on the preservice and relief is requested on the inservice volumetric examinations of the longitudinal welds in the area where the reactor vessel outlet nozzle lip interferes with the weld.

f. REASON FOR REQUEST: The ultrasonic scanning equipment cannot reach this area of the reactor vessel longitudinal weld.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% radiographic examination. (B) Extent of preservice examination: 100% based on shop and field examination. (C) Measures for accessibility: None. (D) Conclusion: Examinations in the field will include all of the longitudianl weld required by the Code except for the portion near the outlet nozzle lip. Complete coverage may be attainable in the tenth year by augmeting the ID examinations with OD examinations of the specific area, if radiation levels permit OD access.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that these longitudinal welds are not accessible for volumetric examination due to the geometric configuration of the reactor vessel outlet nozzle lip interface with the weld. We conclude that (1) the construction code radiographic examination, and (2) the shop fabrication examinations provides an adequate level of assurance of structural integrity. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

We conclude that these longitudinal welds are not accessible for volumetric examination due to the geometric configuration of the reactor vessel outlet nozzle lip interface with the weld. In lieu of this inspection requirement we require that the augmenting outer diameter examinations discussed above in AP&LCo's conclusions shall be performed to the extent practicable as permitted by personnel radiation exposures limits. Therefore, the requested relief from the specific inservice requirements of Section XI is granted.

5. COMPONENT: Integrally Welded Supports

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-251-K-1, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWB-2500-B-K-1, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Volumetric examination of integrally welded supports to piping, valves, and pumps is required.

e. EXEMPTION REQUEST: An exemption is requested on the Preservice and relief is requested on the inservice volumetric examination of integrally welded supports to Class I components. Surface examination is proposed in lieu of the volumetric examination.

f. REASON FOR REQUEST: The integrally welded supports are of a weld configuration (fillet) that is not amenable to UT.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% visual and surface examination. (B) Extent of preservice examination: 100% visual and surface examination and 0% volumetric examination. (C) Measures for accessibility: None. (D) Conclusion: Inservice examinations are planned to be visual and surface in lieu of visual and volumetric since a volumetric examination cannot be done. [Later Code Addenda (Winter 1976) allows the option of either volumetric or surface examinations.]

h. STAFF CONCLUSION:
PRESERVICE

We conclude that the visual and surface inspections will provide an adequate level of assurance of structural integrity of the integrally welded supports. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

We conclude that the proposed visual and surface inspections will provide an adequate level of assurance of the integrity of the integrally welded supports. Therefore, the requested relief from the specific inservice requirements of Section XI is granted.

6. COMPONENT: Reactor Coolant Pump Studs

   a. CODE CLASS: I

   b. PRESERVICE CODE SECTION: IS-251-G-1, 1973 Summer Addenda, 1971 Code

   c. INSERVICE CODE SECTION: IWB-2500-B-G-1, 1975 Summer Addenda, 1974 Code

   d. CODE REQUIREMENT: Volumetric examination of the reactor coolant pump studs is required.

   e. EXEMPTION REQUEST: An exemption is requested to the preservice volumetric examination of the subject studs.

   f. REASON FOR REQUEST: The end preparation on the exposed end of the studs is not amenable to UT. The pumps would have to be disassembled to meet this examination requirement.

   g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% ultrasonic examination of the bar stock during fabrication. (B) Extent of preservice examination: 100% based on NRC acceptance of shop fabrication records. (C) Measures for accessibility: Disassembly of the reactor coolant pumps. (D) Conclusion: Since fabrication volumetric inspection was done, it is planned that the studs will not be examined in the field until required by the inservice portion of Section XI or until the pumps are disassembled for maintenance reasons, whichever comes first.

   h. STAFF CONCLUSION:
      PRESERVICE

      We conclude that the fabrication volumetric inspection will serve as satisfactory preservice assurance of the integrity of the studs. Therefore, the requested exemption freom the specific preservice requirements of Section XI is granted.

      INSERVICE

      No relief is requested or required since AP&LCo states the studs will be examined as required by the inservice requirement of Section XI.

7. COMPONENT: Interior Clad Surfaces of Reactor Vessel

   a. CODE CLASS: I

   b. PRESERVICE CODE SECTION: IS-251-I-1, 1973 Summer Addenda, 1971 Code

   c. INSERVICE CODE SECTION: IWB-2500-B-I-1, 1975 Summer Addenda, 1974 Code

   d. CODE REQUIREMENT: A remote visual examination of six (6) clad patches in the reactor vessel is required.

   e. EXEMPTION REQUEST: An exemption is requested to the visual preservice examination of the clad patches.

   f. REASON FOR REQUEST: The clad patches are not to be examined during inservice inspection since later Code Addenda has deleted this requirement. Therefore, no useful data can be obtained from this preservice examination.

   g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: Not applicable. (B) Extent of preservice examination: 0%. (C) Measures for accessibility: Is presently accessible. (D) Conclusion: This examination has been deleted from Section XI in the Summer 1976 Addenda and therefore will not be part of the end of the ten year interval inspection plan.

   h. STAFF CONCLUSION:
      PRESERVICE

      We conclude that no useful data can be obtained from the preservice inspection. Therefore, the requested exemption from the specific preservice requirement of Section XI is granted.

      INSERVICE

      We conclude that no useful data could be obtained from inservice inspection of cladding patches. The 1975 Summer Addenda, 1974 Edition of the Code is currently applicable to ANO-2 and does include a requirement for the inspection of cladding patches.

We conclude that since the cladding is acessible for examination and could be inspected without imposing an undue hardship on the licensee it is not impractical to conduct such inspections. On this basis, relief from this requirement pursuant to Paragraph 50.55a(g)(6) cannot be granted. However, based on our conclusion that no useful information related to the structural integrity of the reactor coolant pressure boundary can be obtained from the inservice inspection of the cladding we have determined that an exemption is justified and should be granted. Therefore, an exemption from the specific inservice requirement is granted.

8. COMPONENT: Pressurizer

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-251-B, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: None

d. CODE REQUIREMENT: Visual examination of longitudinal and circumferential vessel welds is required in addition to the volumetric examination for which no relief was requested.

e. EXEMPTION REQUEST: An exemption is requested to the preservice visual examination.

f. REASON FOR REQUEST: The subject welds are not to be visually examined during inservice inspection since later Code Addenda has deleted this requirement. Therefore, no useful data can be obtained from a preservice inspection.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: Not applicable. (B) Extent of preservice examination: 0% visual examination. (C) Measures for accessibility: Remove insulation. (D) Conclusion: This examination has been deleted from Section XI in the 1974 Edition.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that no useful data can be obtained from the preservice examination. Therefore, the requested exemption from the specific preservice requirement of Section XI is granted.

INSERVICE

No relief is requested or required on the basis that the Section XI inservice requirements do not include a requirement for visual examination of these welds.

9. COMPONENT: Valve Bodies

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: IS-251-M-2, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWB-2500-B-M-2, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Visual examination of internal surfaces of one valve in each of those groups of valves of the same construction design, manufacturing method, manufacturer and performing similar functions in the system is required.

e. EXEMPTION REQUEST: An exemption is requested to the preservice visual examination.

f. REASON FOR REQUEST: The subject visual examinations would require disassembly of the valves.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: Not applicable. (B) Extent of preservice examination: 0% visual examination. (C) Measures for accessibility: Complete disassembly of the valves. (D) Conclusion: Inspection is considered impractical. The valves will be visually examined when they are disassembled for maintenance purposes.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that disassembly of the valves for preservice visual examination can be omitted with little decrease in assurance of adequate structural integrity of the valves. We note that these valves have been subjected to various surface, volumetric and visual examinations during their manufacture. Therefore, the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

No relief is requested or required since AP&LCo did not request any relief from the Section XI requirements pertaining to inservice examination of the valves such as during maintenance outages.

10. COMPONENT: Shutdown Cooling Heat Exchangers (2 units)

a. CODE CLASS: II

b. PRESERVICE CODE SECTION: ISC-251-C-B, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWC-2520-C-B, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: 100% volumetric examination of the nozzle-to-vessel attachment welds is required.

e. EXEMPTION REQUEST: An exemption is requested from the preservice and relief is requested to the inservice volumetric examination inspections. A surface examination of the collar fillet welds is proposed in lieu of the volumetric requirement.

f. REASON FOR REQUEST: The manufacturer of the heat exchangers welded a steel plate collar support over the subject welds and then welded the collar to the body of the heat exchanger and to the nozzle making the subject welds inaccessible for UT.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% surface examination. (B) Extent of preservice examination: 0% volumetric examination. (C) Measures for accessibility: Redesign heat exchanger nozzle configurations. (D) Conclusion: A field preservice surface examination of the collar fillet welds was performed. Inservice inspection is planned to be done by surface examination of the collar support fillet welds.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that the examination during fabrication and the field examination of the collar weld will provide adequate assurance of preservice integrity of the welds. Therefore the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

The staff believes that AP&LCo should evaluate further the feasibility of the stated redesign to provide accessibility for the inservice inspection of the collar fillet welds. Therefore the requested relief from the specific inservice requirements of Section XI is not granted.

11. COMPONENT: Steam Generator Nozzle-to-Vessel Welds

a. CODE CLASS: II

b. PRESERVICE CODE SECTION: ISC-251-C-B, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWC-2520-C-B, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Nozzle-to-vessel welds are required to be volumetrically examined on both sides of the weld, nozzle side and vessel side.

e. EXEMPTION REQUEST: An exemption is requested from the preservice and relief is requested on the inservice volumetric examination of the nozzle side of the main steam and main feedwater nozzles on the steam generators.

f. REASON FOR REQUEST: Access from the nozzle side of these welds is restricted by weld and nozzle configurations and is not amenable to UT.

g. BASES AND CONCLUSION: AP&LCo provides the following bases: (A) Original fabrication examination: 100% radiographic examination. (B) Extent of preservice examination: 100% based on shop and field examinations. (C) Measures for accessibility: None. (D) Conclusion: Examinations in the field during inservice inspection will include all of the weld and base material on the vessel side of the weld.

h. STAFF CONCLUSION:
PRESERVICE

We conclude that inspection of the nozzle side of these welds is precluded due to the geometric configuration and the accessibility restrictions. We conclude that (1) the construction code radiographic examination, (2) the shop and field examinations, and (3) examination of these welds on the vessel side of the weld provide an adequate level of preservice structural integrity. Therefore the requested exemption from the specific preservice requirement of Section XI is granted.

INSERVICE

We conclude that inspection of the nozzle side of these welds is precluded due to the geometric configuration and the accessibility restrictions. In lieu of performing these examinations we require that surface and visual inspection be performed within the constraints imposed by personel radiation exposure limits. We conclude that (1) the volumetric examination of the vessel side of the weld in accordance with the requirements of Section XI and (2) the supplemental surface and visual inspections will provide an adequate level of assurance of inservice structural integrity. Therefore the requested relief from the specific inservice requirement of Section XI is granted.

12. COMPONENT: Main Steam and Main Feedwater Piping

a. CODE CLASS: II

b. PRESERVICE CODE SECTION: IS-251-C-G, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: IWC-2520-C-G, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: The volumetric examination is to include 50% of the total number of circumferential butt welds in primary containment penetration anchors.

e. EXEMPTION REQUEST: An exemption is requested from the preservice and relief is requested to the inservice UT of the welds located inside the flued containment penetrations.

f. REASON FOR REQUEST: Due to the design configuration of these penetrations, the subject welds are completely inaccessible for UT.

g. BASES AND CONCLUSION: AP&L Co. provides the following bases: (A) Original fabrication examination: 100% radiographic examination. (B) Extent of preservice examination: 100% based on shop inspection records. (C) Measures for accessibility: Redesign of the flued heads.

h. STAFF CONCLUSION:
PRESERVICE:

We conclude that the welds inside the fluid containment penetrations are inaccessible for the purposes of ultrasonic testing (UT) examination. We conclude that the fabrication inspection will provide adequate assurance of the preservice integrity of the main feedwater and main steam pipe welds in containment penetration. Therefore the requested exemption from the specific preservice requirements of Section XI is granted.

INSERVICE

We conclude that the welds on the inside of the flued containment penetrations are inaccessible for the purposes of ultrasonic testing examination. In lieu of testing these welds we require that the adjacent welds be inspected on an augmented schedule. We conclude that the examination of adjacent welds will be indicative of the condition of the inaccessible welds. Therefore, the requested relief from the specific inservice requirements of Section XI is granted.

13. COMPONENT: Main Steam Piping Calibration Blocks

a. CODE CLASS: II

b. PRESERVICE CODE SECTION: Appendix III, 1975 Winter Addenda, 1974 Code

c. INSERVICE CODE SECTION: Appendix III, 1975 Winter Addenda, 1974 Code

d. CODE REQUIREMENT: The basic calibration blocks shall be made from material (pipe) of the same nominal diameter and wall thickness or pipe schedule as those to be examined. However, alternate design and layout may be utilized provided beam paths are the same.

e. EXEMPTION REQUEST: An exemption is requested from the preservice and relief is requested on the inservice requirement for using material of the same nominal diameter as the 38" diameter main steam pipe.

f. REASON FOR REQUEST: 38" diameter main steam piping is not readily available for use as a calibration standard and flat plate material is readily available.

g. BASES AND CONCLUSION: AP&L Co. provides the following bases: Since Appendix III allows use of an alternate design and layout if the beam paths are the same (Section III-3430), an evaluation was done by C-E comparing contact area on the flat plate vs the rounded pipe contour and it was determined that the beam paths (surface contact area) was essentially the same. Therefore, the main steam piping calibration blocks meet the requirements of Appendix III Section III-3430 and do not strictly meet the requirements of Appendix III Section III-3410. However, the calibration blocks are good representatives of the pipe and they do allow accurate calibration.

h. STAFF CONCLUSION
Preservice and Inservice

We conclude that the use of flat plate of the same P-number as the pipe for calibration will not measurably reduce the assurance of the integrity of the main steam piping. Therefore the requested exemption from the specific preservice requirement and the requested relief from the inservice requirements of Section XI is granted.

14. COMPONENT: Ultrasonic Examination Requirements

a. CODE CLASS: I

b. PRESERVICE CODE SECTION: Appendix I, 1973 Summer Addenda, 1971 Code

c. INSERVICE CODE SECTION: Appendix I, 1975 Summer Addenda, 1974 Code

d. CODE REQUIREMENT: Instrument calibration for linearity and angle beam calibration shall be done in accordance with Sections I-2120, I-4100, I-4230, I-4440, I-4450, and I-4460 for the shop preservice inspection of Class I components.

e. RELIEF REQUEST: Relief is requested to the shop preservice requirement for instrument linearity calibration per I-2120, I-4100, and I-4230. An exemption is also requested to the shop preservice requirement for position calibration per I-4440, planar reflector calibration per I-4450, and beam spread measurement in the vertical plane per I-4460.

f. REASON FOR REQUEST: The shop preservice examination of Class I components was done in 1974 prior to the replacement or modification of UT equipment to meet the recently published version of Section XI. Therefore, the above sections of Appendix I could not be met with the existing UT equipment.

g. BASES AND CONCLUSION: AP&L Co. provides the following bases: The shop preservice inspection was done to the standards of the previous addenda of Section XI and was upgraded to meet as much of the Summer 1973 Addenda as was possible without replacing the existing calibration blocks and UT equipment. All field preservice inspections by UT have been with improved instruments and calibration blocks to meet the more stringent requirements of the 1973 Summer Addenda.

h. STAFF CONCLUSION:
PRESERVICE AND INSERVICE

We conclude that the use of then-existing UT calibration requirements for shop preservice inspection will not measurably reduce the assurance of preservice integrity of the components thus inspected. Therefore the requested exemption from the specific preservice requirement of Section XI is granted.

15. COMPONENT: Class I Stainless Steel Piping and Class II Stainless Steel and Ferritic Piping Examinations

a. CODE CLASS: I and II

b. PRESERVICE CODE SECTION: Appendix III, 1975 Winter Addenda 1974 Code in lieu of 1973 Summer Addenda 1971 Code.

c. INSERVICE CODE SECTION: Appendix III, 1975 Winter Addenda, 1974 Code in lieu of 1975 Summer Addenda, 1974 Code.

d. CODE REQUIREMENT: Appendix III, which defines the ultrasonic (UT) examination methods, equipment, and requirements applicable to Class 1 and 2 ferritic steel piping systems, was first added to Section XI of the Code in the 1975 Winter Addenda of the 1974 Code. Prior to that time Section XI of the Code referred one to applicable sections of Section III of the Code for a definition of the UT examination methods, equipment and requirements.

e. EXEMPTION REQUEST: Although the licensee has not specifically requested an exemption from the preservice requirements of the Summer 1973 Addenda, 1971 Edition or from the inservice requirements of the Summer 1975 Addenda of the 1974 Edition we have determined that an exemption is required and have also determined that an exemption regarding the definition of the UT examination methods, equipment and requirements is justified.

f. REASON FOR REQUEST: The licensee states that Class I stainless steel piping and Class II stainless steel and ferritic piping examinations were conducted in accordance with Appendix III criteria of the 1975 Winter Addenda of the 1974 Edition of Section XI. However, the licensee has elected to meet the 1973 Summer Addenda 1971 Edition for other major elements of the preservice requirements and Paragraph 50.55a(b) stipulates that the 1975 Summer Addenda, 1974 Edition is applicable for inservice requirements. Therefore, the criteria used for the Class I stainless steel piping and Class II stainless steel and ferritic piping is not consistent with that elected for the remainder of the Class I and II preservice examinations and does not strictly meet the requirements of Paragraphs 50.55a(g)(4) and 50.55a(b) regarding the inservice examinations.

g. BASES AND CONCLUSION: We have determined that since the licensee has completed the preservice examinations it would pose an undue hardship with no compensating increase in the level of quality and safety to require the reperformance of the preservice examination program in accordance with the same Addenda and Edition of the Code as were utilized for the remainder of the Class I and II components. Based on our position that the same techniques should be used wherever possible for the inservice examinations as were used for the preservice examinations we have determined that no compensating increase in the level of quality and safety would result from requiring the licensee to meet the specific inservice requirements of the 1973 Summer Addenda, 1971 Edition.

h. STAFF CONCLUSION:
PRESERVICE AND INSERVICE

We conclude that the conduct of the Class I stainless steel piping and the Class II stainless steel and ferritic piping examinations in accordance with the Appendix III criteria of the 1975 Winter Addenda of the 1974 Edition of Section XI is acceptable. Therefore, an exemption from the specific preservice requirement of the 1973 Summer Addenda, 1971 Edition and an exemption from the specific inservice requirements of the 1975 Summer Addenda, 1974 Edition of Section XI is granted.

IV. PUBLIC INTEREST REGARDING COMPLIANCE WITH SECTIONS 50.55a(g)(2) AND 50.55a(g)(4) OF 10 CFR PART 50

Our technical evaluation has not identified practical methods by which the Arkansas Nuclear One - Unit No. 2 preservice and inservice inspection programs can meet certain ASME Code Section XI requirements of 10 CFR, Part 50, Paragraph 50.55a(g)(2) and Paragraph 50.55a(g)(4). Requiring specific compliance with this paragraph would include the following actions: delay the startup of the plant and remove significant portions of the involved systems as discussed in Part III review, redesign and fabricate, if possille, new sections for the involved systems within the available space; reinstall the new sections for the involved systems; and repeat the system hydrostatic pressure test. The as built structural integrity of the primary pressure boundary piping is not dependent on the required Section XI preservice examination since the applicable construction codes certain examination and testing requirements which by themselves provide the necessary assurance of structural integrity. Adequate assurance of inservice structural integrity is provided by the examination of the subject welds to the extent practicable and by supplemental surface and visual examinations of the welds. We believe that the public interest is served by not imposing the certain provisions of 10 CFR, Part 50, Paragraph 50.55a(g)(2) and Paragraph 50.55a(g)(4), that have been determined to be either impractical or would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

V. CONCLUSIONS

Based on the foregoing, we have determined that, pursuant to 10 CFR Section 50.12 and pursuant to 10 CFR Section 50.55a(g)(6), an exemption and relief, respectively, is authorized by law and can be granted without endangering life or property or the common defense and security and is otherwise in the public interest. In making this determination we have given due consideration to the burden that could result if these requirements were imposed on the facility. We have also determined that the granting of this exemption and relief would not involve a significant increase in the probablity of consequences of accidents previously considered nor a significant decrease in safety margin; and thus, would not involve a significant hazards consideration.

Furthermore, we have determined that the granting of this exemption and relief 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. We have concluded that this exemption and relief would be 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 this action.

AMENDMENT NO. 1
ARKANSAS NUCLEAR ONE, UNIT 2 OPERATING LICENSE NO. NPF-6 DATED SEPTEMBER 1, 1978

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