April 18, 1983

DISTRIBUTION:
Docket File
NRC PDR
L PDR
ORB#4 Rdg
Gray File+4
DEisenhut
RIngram
GVissing
OELD
AEOD
LHarmon-2
EJordan
ACRS-10
HOrnstein
EBlackwood
RDiggs
CMiles
LSchneider
TBarnhart-4
ASLAB
DBrinkman

Docket No. 50-313

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

Dear Mr. Griffin:

The Commission has issued the enclosed Amendment No. 77 to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1). The amendment consists of changes to the Technical Specifications in partial response to your application submitted by letter dated October 19, 1977, as supplemented by letter dated December 15, 1978.

This amendment revises the language for the Technical Specifications relating to inservice inspection requirements of safety class components to conform with the Codes and Standards Rule, 10 CFR 50.55a. This rule requires in part that inservice inspection of ASME Code Class 1, 2 and 3 components be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda except where specific written relief is granted by the Nuclear Regulatory Commission.

By letter dated October 19, 1977, supplemented by letter dated December 15, 1978, you also submitted a proposed inservice inspection program description and request for relief from ASME Code requirements pursuant to 10 CFR 50.55a(g) for ANO-1. This letter is also to inform you of the results of our review of your relief requests and to grant relief in part from the requirements of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (the Code) or impose other requirements, as appropriate.

Section 50.55a(g) of 10 CFR Part 50 requires that your program be revised at 120-month intervals with the start of commercial operation being the reference date. The start of the next interval for your facility is December 19, 1984, and your inservice inspection and testing program must be based on the edition and addenda of the Code incorporated by 10 CFR 50.55a(g) 12 months prior to that date. Any changes to your Technical Specifications are required to be submitted at least six months prior to the beginning of a 120-month interval, and it is requested that any requests for relief from Code requirements be provided on the same schedule. Our review of your relief requests for your next interval will be conducted on a schedule based on the program-revision requirements for your facility. Until that time you should follow the inservice inspection program proposed by your letter dated October 19, 1977, as supplemented by letter dated December 15, 1978.

OFFICE							
SURNAME	8305090147 830418 PDR ADOCK 05000313 P PDR						
DATE							

modified as described herein and by any further relief granted or additional testing imposed during the remainder of the period. Any relief from Code requirements granted herein expires on December 19, 1984.

The enclosed Safety Evaluation supporting Amendment No. 77 and the Granting of Relief delineates those items for which relief has been granted and alternate schedules and procedures defined. We have determined that where stated the Code requirements are impractical, the granting of this relief is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest considering the burden that could result if they were imposed on your facility.

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

Sincerely,

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

Enclosures:
1. Amendment No. 77
2. Safety Evaluation w/attached TER
3. Notice

cc w/enclosures:
See next page

amendment and Fed. Reg. notice only

OFFICE	ORB#4:DL	ORB#4:DL	C-ORB#4:DL	ORB#1:DL	MTEB:DE	OELD	AD-OR:DL
SURNAME	RIngram	GVissing/cb	JStolz	PTam	BDLiaw	4/5/83 McGurren	GLainas
DATE	3/28/83	3/24/83	3/28/83	3/31/83	3/31/83	3/ /83	3/ /83

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

April 18, 1983

DISTRIBUTION
Docket File
ORB#4 Rdg
RIngram

Docket No. 50-313

Docketing and Service Section
Office of the Secretary of the Commission

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1

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

- ☐ Notice of Receipt of Application for Construction Permit(s) and Operating License(s).
- ☐ Notice of Receipt of Partial Application for Construction Permit(s) and Facility License(s): Time for Submission of Views on Antitrust Matters.
- ☐ Notice of Availability of Applicant's Environmental Report.
- ☐ Notice of Proposed Issuance of Amendment to Facility Operating License.
- ☐ Notice of Receipt of Application for Facility License(s); Notice of Availability of Applicant's Environmental Report; and Notice of Consideration of Issuance of Facility License(s) and Notice of Opportunity for Hearing.
- ☐ Notice of Availability of NRC Draft/Final Environmental Statement.
- ☐ Notice of Limited Work Authorization.
- ☐ Notice of Availability of Safety Evaluation Report.
- ☐ Notice of Issuance of Construction Permit(s).
- ☒ Notice of Issuance of Facility Operating License(s) or Amendment(s).
- ☒ Other: Amendment No. 77.
  Referenced documents have been provided PDR.

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

Enclosure:
As Stated

OFFICE →	ORB#4:DL					
SURNAME →	RIngram;cf					
DATE →	4/19/83					

NRC FORM 102 (1-76)

cc w/enclosure(s):

Mr. John R. Marshall
Manager, Licensing
Arkansas Power & Light Company
P. O. Box 551
Little Rock, Arkansas 72203

Mr. James M. Levine
General Manager
Arkansas Nuclear One
P. O. Box 608
Russellville, Arkansas 72801

Mr. Leonard Joe Callan
U.S. Nuclear Regulatory Commission
P. O. Box 2090
Russellville, Arkansas 72801

Mr. Robert B. Borsum
Babcock & Wilcox
Nuclear Power Generation Division
Suite 220, 7910 Woodmont Avenue
Bethesda, Maryland 20814

Mr. Nicholas S. Reynolds
Debevoise & Liberman
1200 17th Street, NW
Washington, DC 20036

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

Regional Radiation Representative
EPA Region VI
1201 Elm Street
Dallas, Texas 75270

Mr. John T. Collins, Regional Administrator
U. S. Nuclear Regulatory Commission, Region IV
611 Ryan Plaza Drive, Suite 1000
Arlington, Texas 76011

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

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

ARKANSAS POWER & LIGHT COMPANY

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO.1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 77
License No. DPR-51

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

A. The application for amendment by Arkansas Power and Light Company (the licensee) dated October 19, 1977, as supplemented December 15, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

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

D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.c.(2) of Facility Operating License No. DPR-51 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No.77 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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

Attachment:
Changes to the Technical
Specifications

Date of Issuance: April 18, 1983

ATTACHMENT TO LICENSE AMENDMENT NO. 77

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

Remove	Insert
67	67
76	76
77	77
77b	77b

4 SURVEILLANCE REQUIREMENTS

Specified surveillance intervals may be adjusted plus or minus 25 percent to accommodate normal test and surveillance schedules. The maximum combined interval for any 3 consecutive tests shall not exceed 3.25 times the specified surveillance interval. Surveillance requirements are not applicable when the plant operating conditions are below those requiring operability of the designated component. However, the required surveillance must be performed prior to reaching the operating conditions requiring operability. For example, instrumentation requiring twice per week surveillance when the reactor is critical need not have the required surveillance when the reactor is shutdown.

Inservice inspection of ASME Code Class 1, 2, 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Failure to perform a Surveillance Requirement within the specified time interval shall constitute a failure to meet the OPERABILITY requirements for a Limiting Condition for Operation. Exceptions to these requirements are stated in the individual specifications. Surveillance Requirements do not have to be performed on inoperable equipment.

4.1 OPERATIONAL SAFETY ITEMS

Applicability

Applies to items directly related to safety limits and limiting conditions for operation.

Objective

To specify the minimum frequency and type of surveillance to be applied to unit equipment and conditions.

Specification

a. The minimum frequency and type of surveillance required for reactor protective system and engineered safeguards system instrumentation when the reactor is critical shall be as stated in Table 4.1-1.

b. Equipment and sampling test shall be performed as detailed in Tables 4.1-2 and 4.1-3.

c. Discrepancies noted during surveillance testing will be corrected and recorded.

d. A power distribution map shall be made to verify the expected power distribution at periodic intervals at least every 10 effective full power days using the incore instrumentation detector system.

4.2 REACTOR COOLANT SYSTEM SURVEILLANCE

Applicability

Applies to the surveillance of the reactor coolant system pressure boundary.

Objective

To assure the continued integrity of the reactor coolant system pressure boundary.

Specification

4.2.1 Prior to initial unit operation, an ultrasonic test survey shall be made of reactor coolant system pressure boundary welds as required to establish preoperational integrity and baseline data for future inspections.

4.2.2 Post-operational inspections of components shall be made in accordance with the methods and intervals indicated in Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the NRC.

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

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

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

4.2.6 Surface and volumetric examination of the reactor coolant pump flywheels will be conducted coincident with refueling or maintenance shutdowns such that during 10 year intervals all four reactor coolant pump flywheels will be examined. Such examinations will be performed to the extent possible through the access ports, i.e., those areas of the flywheel accessible without motor disassembly. The surface and volumetric examination may be accomplished by Acoustic Emission Examination as an initial examination method. Should the results of the Acoustic Emission Examination indicate that additional examination is necessary to ensure the structural integrity of the flywheel, then other appropriate NDE methods will be performed on the area of concern.

4.2.7 The reactor vessel material irradiation surveillance specimens removed from the reactor vessel in 1976 shall be installed, irradiated in and withdrawn from the Davis-Besse Unit No. 1 reactor vessel in accordance with the schedule shown in Table 4.2-1. Following withdrawal of each capsule listed in Table 4.2-1, Arkansas Power & Light Company shall be responsible for testing the specimens and submitting a report of test results in accordance with 10CFR50, Appendix H.

Table 4.2-1

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

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

Bases

The surveillance program has been developed to comply with the applicable edition of Section XI and addenda of the ASME Boiler and Pressure Vessel Code, Inservice Inspection of Nuclear Reactor Coolant Systems, as required by 10 CFR 50.55a, to the extent practicable within limitations of design, geometry and materials of construction.

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO REQUESTS FOR RELIEF FROM INSERVICE INSPECTION REQUIREMENTS

AND

SUPPORTING AMENDMENT NO. 77 TO

FACILITY OPERATING LICENSE NO. DPR-51

ARKANSAS POWER AND LIGHT COMPANY

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

Introduction

By letter dated October 19, 1977, as supplemented by letter dated December 15, 1978, Arkansas Power & Light Company (the licensee or AP&L) requested amendment to the Technical Specifications (TSs) appended to Facility Operating License No. DPR-51 for Arkansas Nuclear One, Unit No. 1 (ANO-1).

The amendment would revise the language for the TSs relating to inservice inspection requirements of safety class components to conform with the Codes and Standards Rule, 10 CFR 50.55a. This rule requires in part that inservice inspection of ASME Code Class 1, 2 and 3 components be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda except where specific written relief is granted by the Nuclear Regulatory Commission (the Commission).

The licensee also submitted a proposed inservice inspection program description and requested relief from certain Code requirements, determined to be impractical to perform on ANO-1 during the inspection interval.

Discussion

The proposed TS 4.2.2 conforms to the Codes and Standards Rule 10 CFR 50.55a(g). The proposed TS 4.2.2 for ANO-1 states that inservice examination of ASME Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g) except where specific written relief has been granted by the Commission. Certain requirements of later editions and addenda of Section XI are impractical to perform on older plants because of the plants' design, component geometry, and materials of construction. Thus, 10 CFR 50.55a(g)(6)(i) authorizes the Commission to grant relief from those requirements upon making the necessary findings.

By letters dated October 19, 1977 and December 15, 1978, AP&L submitted its inservice inspection program revisions, or additional information related to requests for relief from certain Code requirements, determined to be impractical to perform on ANO-1 during the inspection interval. The program is based on the 1974 Edition through Summer 1975 Addenda of Section XI of the ASME Code and covers the remainder of the 120-month inspection interval which ends December 19, 1984.

Evaluation

On the basis that the proposed TS 4.2.2 does conform to 10 CFR 50.55a(g), we find it acceptable.

Requests for relief from the requirements of Section XI which have been determined to be impractical to perform have been reviewed by our contractor, Science Applications, Inc. The contractor's evaluations of the licensee's requests for relief and his recommendations are presented in the attached Technical Evaluation Report (TER).

We have reviewed the TER and agree with the evaluations and recommendations. A summary of our determinations is presented in the following tables:

TABLE 1

CLASS 1 COMPONENTS

IWB-2600 ITEM NO.	IWB-2500 EXAM. CAT.	SYSTEM OR COMPONENT	AREA TO BE EXAMINED	REQUIRED METHOD	LICENSEE PROPOSED ALTERNATIVE EXAMINATION	RELIEF REQUEST STATUS
B1.4	B-D	Reactor Vessel	Nozzle Inside Radiused Sections	Volumetric: 25% During 1st 40-month period, 50% by end of 2nd 40-month Period, 100% by End of Interval	Volumetric Near end of of interval	Granted
B1.6	B-F	Piping, Core Flood Nozzles	Nozzle-To-Safe End Welds	Volumetric and Surface at Scheduled Intervals	Volumetric and Surface at end of Interval	Note 1
B1.11	B-G-2	Control Rod Drive Mechanism	Pressure - Retaining Bolting	Visual	Visual 10% Peripheral CRDM's	Granted
B4.5	B-J	High Pressure Injection and Core Flood	Circumferential Butt Welds: A1-8A, B1-10A, B2-10A, W-1 & Y-1	Volumetric	None	Note 1

Note 1: Components may be subjected to an augmented examination as a result of generic thermal sleeve failures. If not, the proposed alternative examination may be performed.

TABLE 1

CLASS 1 COMPONENTS
(Continued)

IWB-2600 ITEM NO.	IWB-2500 EXAM. CAT.	SYSTEM OR COMPONENT	AREA TO BE EXAMINED	REQUIRED METHOD	LICENSEE PROPOSED ALTERNATIVE EXAMINATION	RELIEF REQUEST STATUS
B4.9	B-K-1	Piping	Integrally - Welded Support Fillet Welds	Volumetric	Surface	Granted

TABLE 2

CLASS 2 COMPONENTS

IWC-2600 ITEM NO.	IWC-2520 EXAM. CAT.	SYSTEM OR COMPONENT	AREA TO BE EXAMINED	REQUIRED METHOD	LICENSEE PROPOSED ALTERNATIVE EXAMINATION	RELIEF REQUEST STATUS

[NO RELIEF REQUESTS]

TABLE 3

CLASS 3 COMPONENTS

IWC-2600 ITEM NO.	IWC-2520 EXAM. CAT.	SYSTEM OR COMPONENT	AREA TO BE EXAMINED	REQUIRED METHOD	LICENSEE PROPOSED ALTERNATIVE EXAMINATION	RELIEF REQUEST STATUS

[NO RELIEF REQUESTS]

TABLE 4

PRESSURE TESTS

SYSTEM OR COMPONENT	IWC-5000 & IWD-5000 TEST PRESSURE REQUIREMENT	LICENSEE PROPOSED ALTERNATE TEST PRESSURE	RELIEF REQUEST STATUS

[NO RELIEF REQUESTS]

TABLE 5

ULTRASONIC EXAMINATION TECHNIQUE

SYSTEM OR COMPONENT	REQUIREMENT	LICENSEE PROPOSED ALTERNATIVE EXAMINATION METHOD	RELIEF REQUEST STATUS

No Relief Requests

TABLE 6

GENERAL RELIEF REQUESTS

ALL CLASSES/COMPONENTS

SYSTEM OR COMPONENT	REQUIREMENT	LICENSEE ALTERNATE	RELIEF REQUEST STATUS

[NO RELIEF REQUESTS]

Based on the review summarized, we conclude that relief granted from the examination requirements and alternate methods imposed through this document give reasonable assurance of the piping and component pressure boundary and support structural integrity, that granting relief where the Code requirements are impractical is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest considering the burden that could result if they were imposed on the facility.

Environmental Consideration

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

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because this action does not involve a significant increase in the probability or consequences of an accident previously evaluated, does not create the possibility of an accident of a type different from any evaluated previously, and does not involve a significant reduction in a margin of safety, this action does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this action will not be inimical to the common defense and security or to the health and safety of the public.

Dated: April 18, 1983

The following NRC personnel have contributed to this Safety Evaluation: Guy S. Vissing, George Johnson.

ATTACHMENT 1

SCIENCE APPLICATIONS, INC.

SAI Report No. 186-028-39

ARKANSAS NUCLEAR ONE, UNIT 1

INSERVICE INSPECTION PROGRAM

TECHNICAL EVALUATION REPORT

Submitted to:

U.S. Nuclear Regulatory Commission
Contract No. 03-82-096

Science Applications, Inc.
McLean, Virginia 22102

September 17, 1982

Science Applications, Inc.

CONTENTS

TECHNICAL EVALUATION REPORT
ARKANSAS NUCLEAR ONE, UNIT 1
INSERVICE INSPECTION PROGRAM

INTRODUCTION

The revision to 10 CFR 50.55a, published in February 1976, required that Inservice Inspection (ISI) Programs be updated to meet the requirements (to the extent practical) of the Edition and Addenda of Section XI of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code* incorporated in the Regulation by reference in paragraph (b). This updating of the programs was required to be done every 40 months to reflect the new requirements of the later edition of Section XI.

As specified in the February 1976 revision, for plants with Operating Licenses issued prior to March 1, 1976, the Regulations became effective after September 1, 1976, at the start of the next regular 40-month inspection period. The initial inservice examinations conducted during the first 40-month period were to comply with the requirements in editions of Section XI and addenda in effect no more than six months prior to the date of start of facility commercial operation.

The Regulation recognized that the requirements of the later editions and addenda of the Section XI might not be practical to implement at facilities because of limitations of design, geometry, and materials of construction of components and systems. It therefore permitted determinations of impractical examination or testing requirements to be evaluated. Relief from these requirements could be granted provided health and safety of the public were not endangered giving due consideration to the burden placed on the licensee if the requirements were imposed. This report provides evaluations of the various requests for relief by the licensee, Arkansas Power and Light (APL), of the Arkansas Nuclear One, Unit 1 Plant. It deals only with inservice examinations of components and with system pressure tests. Inservice tests of pumps and valves (IST programs) are being evaluated separately.

---

* Hereinafter referred to as Section XI or Code.

The revision to 10 CFR 50.55a, effective November 1, 1979, modified the time interval for updating ISI programs and incorporated by reference a later edition and addenda of Section XI. The updating intervals were extended from 40 months to 120 months to be consistent with intervals as defined in Section XI.

For plants with Operating Licenses issued prior to March 1, 1976, the provisions of the November 1, 1979, revision are effective after September 1, 1976, at the start of the next one-third of the 120-month interval. During the one-third of an interval and throughout the remainder of the interval, inservice examinations shall comply with the latest edition and addenda of Section XI, incorporated by reference in the Regulation, on the date 12 months prior to the start of that one-third of an interval. For Arkansas One, the ISI program and the relief requests evaluated in this report cover the second and third 40-month inspection period of the first 10-year interval, i.e., from April 19, 1978, through December 19, 1984. <u>This program was based upon the 1974 Edition of Section XI of the ASME Boiler and Pressure Vessel Code with Addenda through the Summer of 1975.</u>

The November 1979 revision of the Regulation also provides that the ISI programs may meet the requirements of subsequent code editions and addenda, incorporated by reference in Paragraph (b) and subject to Nuclear Regulatory Commission (NRC) approval. Portions of such editions or addenda may be used provided that all related requirements of the respective editions or addenda are met. These instances are addressed on a case-by-case basis in the body of this report.

Finally, Section XI of the Code provides for certain components and systems to be exempted from its requirements. In some instances, these exemptions are not acceptable to NRC or are only acceptable with restrictions.

References (1) to (9) listed at the end of this report pertain to information transmittals on the Inservice Inspection (ISI) Program between the licensee and the NRC. By letters of April 28 and November 24, 1976,[1,3] the Commission provided general ISI guidance to all licensees. Submittals in response to that guidance were made by the licensee on June 9, 1976,[2] and October 19, 1977.[4] The October 19, 1977, submittal also contained proposed changes to the Technical Specifications. The Commission granted interim relief on April 20, 1978.[5] By letters of September 13, 1978,[6] and

March 2, 1982,[10] the NRC requested additional information to complete this review. Some responses were furnished by the licensee on December 15, 1978.[7] As of the date of this report, the licensee has not responded to the March 2, 1982, request. In addition, a one-time relief request was made on January 10, 1979,[8] for reactor welds whose examination had not been in compliance with Code. Relief was granted on March 8, 1979,[9] and so this request is not covered in this report.

From these submittals, a total of five requests for relief from Code requirements or updating to a later code were identified. These requests are evaluated in the following sections of this report. The failure of the licensee to respond to the March 2, 1982 request for information does not affect SAI's evaluation of these relief requests. However, the licensee apparently still has a number of pending items that have not been formalized into relief requests. In addition, he has not committed to a program to inspect the Emergency Core Cooling, Residual Heat Removal, and Containment Heat Removal systems as required by 10 CFR 50.55a(b)(2)(iv)(A).

I. CLASS 1 COMPONENTS

A. Reactor Vessel

1. Nozzle-to-Shell Welds and Nozzle Inside Radiused Sections Category B-D, Item B1.4

Code Requirement

A volumetric analysis of these welds shall be made according to the schedule given in paragraph IWB-2411, which states, "at least 25% of the required examinations shall have been completed by the expiration of one-third of the inspection interval (with credit for no more than 33-1/3% if additional examinations are completed) and at least 50% shall have been completed by the expiration of two-thirds of the inspection interval (with credit for no more than 66-2/3%). The remaining required examinations shall be completed by the end of the inspection interval."

Code Relief Request

Relief is requested from the schedule given in IWB-2411.

Proposed Alternative Examination

All nozzles will be examined once every 10 years near the end of the interval when the core barrel is removed. In addition, both outlet nozzles would be examined from the inside during the first third of the interval.

Licensee's Basis for Requesting Relief

This request involves the four inlet nozzles and two core flood nozzles. Access to these nozzles would require defueling and removal of the core barrel.

Evaluation

Removing the core barrel more than once during an interval merely to comply with schedule given in IWB-2411 is not practical from the standpoint of keeping personnel exposures as low as reasonably achievable (ALARA). This is recognized for Category B-B pressure retaining welds where Code permits examination at or near the end of each inspection interval. The licensee's proposed alternative examination exceeds Code requirements in one respect: the outlet nozzles are examined twice--once during the first third of the interval and once at the end of the interval.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief from IWB-2411 should be granted and the proposed alternative of examining all the nozzles at one time when the core barrel is removed (with the outlet nozzles already examined during the first inspection period) should be approved.

References

Reference 4.

2. Nozzle to Safe-End Welds, Category B-F, Item B1.6

Code Requirement

Volumetric and surface examinations shall be made of 100% of each circumferential weld of dissimilar metals. Examinations in each 40-month period shall be in accordance with paragraph IWB-2411.

Code Relief Request

Relief is requested from the schedule given in IWB-2411 for the two core flood nozzle safe-end welds.

Proposed Alternative Examination

Both core flood nozzle safe-end welds would be examined at the end of the interval from the vessel ID using a remote examination device.

Licensee's Basis for Requesting Relief

Access would require defueling and removal of core barrel. Although these welds were examined from the OD during baseline, ultrasonic examination from the vessel ID using a remote examination device is preferred to limit high radiation exposures to personnel.

Evaluation

Examination from the inside using the remote examination device is the only practical way to examine these welds and keep personnel exposure as low as reasonably achievable. The core barrel needs to be removed to make these examinations and this should only need doing once per interval. This is recognized in the Code requirements for Category B-B examinations, which only require that the examination be done at or near the end of the inspection interval.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, the following is recommended:

Code relief from IWB-2411 should be granted and the proposed alternative of examining both core flood nozzles at one time when the core barrel is removed should be approved.

References

Reference 4.

3. Pressure Retaining Bolting, Category B-G-2, Item B1.11

Code Requirement

Visual examination of 100% of the bolts, studs and nuts each interval is required. Bolting may be examined either in place under tension, when the connection is disassembled, or when bolting is removed.

Code Relief Request

Relief is requested from examining 100% of the Control Rod Drive Mechanism (CRDM) bolts and housing flange rings.

Proposed Alternative Examination

The licensee proposes to examine bolts and nuts on 10% of peripheral CRDM's to coincide with the extent of CRDM's examination as required by Category B-O, Pressure Retaining Welds in CRD Housing.

Licensee's Basis for Requesting Relief

It is impractical to visually examine the light flange bolts on each of the 69 CRDMs from the platform of the head service structure, approximately 20 feet above the flange surface. Most of the bolts on the 24 peripheral CRDM can be observed through the twelve 12" dia. ports in the service structure cylinder. The remainder of the CRDMs accessible for examination only when removed.

Evaluation

Due to the design of the reactor, the pressure retaining bolting, except for the peripheral CRDM is not accessible for visual inspection except when the CRDM is removed. Visual inspection of the bolting in place provides only limited information about the condition of the bolting. Furthermore, unbolting to examine the bolting may compromise the system more than it provides assurance of integrity. Evidence of leakage during pressure tests provides better information. The cost and personnel exposure encountered in removing all the CRDMs to make a visual inspection is not warranted by the increase in safety.

The licensee proposes to examine 10% of the 24 peripheral CRDM bolting each interval. In addition, the licensee should examine the bolting of any CRDM when removed and should conduct the visual inspection for leakage during pressure tests.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the bolting discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, it is recommended that relief should be granted from 100% visual examination of the CRDM bolting, provided that:

(a) the bolting of 10% of the peripheral CRDMs is examined each interval,

(b) the bolting of all the removed CRDMs is examined, and

(c) visual examinations for evidence of leakage are made during pressure tests performed according to IWB-5000.

References

Reference 4.

B. Pressurizer

No relief requests.

C. Heat Exchangers and Steam Generators

No relief requests.

D. Piping Pressure Boundary

1. Circumferential Butt Welds, Category B-J, Item B4.5

Code Requirement

The volumetric examinations performed during each inspection interval shall cover all of the area of 25% of the circumferential joints, including the adjoining 1-foot sections of longitudinal joints, as scheduled according to paragraph IWB-2411. Examinations in each interval shall cover a different 25% until all welds have been examined.

Code Relief Request

Relief is requested from making examinations of inaccessible circumferential welds which are as follows:

(1) High Pressure Injection Lines - Welds A1-8A, A2-4A, B1-10A, and B2-10A.

(2) Core Flood Lines - Welds W-1 and Y-1.

Proposed Alternative Examination

None.

Licensee's Basis for Requesting Relief

High Pressure Injection welds A1-8A, A2-4A, B1-10A, and B2-10A are inside penetrations in the shield wall and are not accessible for examination. Core Flood welds W-1 and Y-1 are inaccessible for examination due to pipe supports. These welds were not examined during baseline and this fact is documented in the preoperational inspection report.

Evaluation

Because these six welds are completely inaccessible, examination is not practical. However, the number of inaccessible welds is sufficiently small and random, compared with the total number of welds in Category B-J (or in either of the two affected systems) that none of these welds needs to be included in the 25% sample to be examined during this inspection interval.

For subsequent inspection intervals, the licensee has the option of updating to subsequent code versions or of staying with the 1974 Edition and addenda through the Summer 1975 Addenda, pursuant to 10 CFR 50.55a(b)(2)(ii). Updating would allow the licensee to examine the same 25% sample, if the provisions of the Summer 1978 Addenda of the 1977 Edition continue to prevail (see Footnote (2) of Category B-J in Table IWB-2500-1). By adopting 10 CFR 50.55a(b)(2)(ii) the Commission was offering an option whereby "operating facilities with ongoing inservice inspection programs would have continuity in the extent and frequency of examinations for pipe welds" (see 44 FR 57913).

Based on these considerations, relief from these requirements is not required at this time for these welds. It is preferable to defer a decision until the next inspection interval after the licensee has determined which of the above options he wishes to exercise.

In addition, visual examination of the welds for which code relief is requested could be performed in the interim. Those welds covered by the pipe supports could also be examined if the pipe supports can and need to be disassembled for maintenance.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for these welds in the core flood and high pressure injection lines, relief from the impractical Code requirements is not needed. Therefore, the following is recommended:

(a) Relief from volumetric examination should not be granted for this inspection interval.

(b) In the event that the pipe supports are disassembled for maintenance and the welds W-1 and Y-1 in the core flood lines are accessible for examination, the Code-required examination should be performed.

References

Reference 4.

2. Integrally Welded Supports, Category B-K-1, Item B4.9

Code Requirement

The areas shall include the integrally welded external support attachments. This includes the welds to the pressure retaining boundary and the base metal beneath the weld zone and along the support attachment member for a distance of two support thicknesses. The examination performed during each inspection interval shall cover 25% of the integrally welded supports and shall be scheduled within the interval per IWB-2411.

Code Relief Request

Relief is requested from the volumetric examination.

Proposed Alternative Examination

Surface examination will be performed on integrally welded attachments.

Licensee's Basis for Requesting Relief

The welds are not designed for ultrasonic examination. Most of the welded attachment for supports are fillet welds (as opposed to full penetration) and are comprised of components with geometric configurations that prohibit ultrasonic examination of the examination area.

Evaluation

The geometry of fillet welds for piping supports generally cannot be examined to the extent required by Section XI by ultrasonic examination. Ultrasonic examination of the base metal would detect piping flaws in the heat affected zone but would provide little or no information on weld penetration. Any penetration flaws would most likely generate at the surface and be detectable by surface examination.

Conclusions and Recommendations

Based on the above evaluation, it is concluded that for the welds discussed above, the code requirements are impractical. It is further concluded that the alternative examination discussed above will provide necessary added assurance of structural reliability. Therefore, it is recommended that Code relief from volumetric examination be granted provided the alternative surface examination is performed.

References

Reference 4.

E. Pump Pressure Boundary

No relief requests.

F. Valve Pressure Boundary

No relief requests.

II. CLASS 2 COMPONENTS

No relief requests.

III. CLASS 3 COMPONENTS

No relief requests.

IV. PRESSURE TESTS

No relief requests.

V. GENERAL

No relief requests.

REFERENCES

1. D. L. Ziemann (NRC) to J. D. Phillips (APL), Arkansas Nuclear One-Unit No. 1, April 28, 1976.

2. W. Cavanaugh, III (APL) to D. L. Ziemann (NRC), Inservice Inspection Testing Requirements, June 9, 1976.

3. D. L. Ziemann (NRC) to J. D. Phillips (APL), Arkansas Nuclear One-Unit No. 1, License No. DPR-51, November 24, 1976.

4. W. Cavanaugh III (APL) to D. K. Davis (NRC), Arkansas Nuclear One-Unit No. 1, Docket No. 50-313, License No. DPR-51, Proposed Technical Specifications, October 19, 1977.

5. R. W. Reid (NRC) to W. Cavanaugh III (APL), April 20, 1978.

6. R. W. Reid (NRC) to W. Cavanaugh III (APL), September 13, 1978.

7. D. H. Williams (APL) to R. W. Reid (NRC), Arkansas Nuclear One-Unit No. 1, Docket No. 50-313, License No. DPR-51, Inservice Inspection Program, December 15, 1978.

8. D. H. Williams (APL) to R. W. Reid (NRC), Arkansas Nuclear One - Unit No. 1 Docket 50-313, License No. DPR-51, Inservice Inspection, January 10, 1979.

9. R. W. Reid (NRC) to W. Cavanaugh III (APL), March 8, 1979.

10. J. F. Stolz (NRC) to W. Cavanaugh III (APL), March 2, 1982.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-313

ARKANSAS POWER & LIGHT COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE AND GRANTING OF RELIEF FROM ASME SECTION XI INSERVICE INSPECTION REQUIREMENTS

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 77 to Facility Operating License No. DPR-51, issued to Arkansas Power and Light Company (the licensee), which revised the Technical Specifications for operation of Arkansas Nuclear One, Unit No. 1 (ANO-1) located in Pope County, Arkansas. The amendment is effective as of the date of issuance.

The amendment modifies the Technical Specifications relating to inservice inspection requirements of safety class components.

By letter dated April 18, 1983, as supported by the related Safety Evaluation, the Commission has also granted to the licensee relief from certain requirements of the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components". The relief relates to the inservice inspection program for ANO-1. The ASME Code requirements are incorporated by reference into the Commission's rules and regulations in 10 CFR Part 50. The relief is effective as of its date of issuance.

The application for the amendment and request for relief comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment and letter granting relief. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

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

For further details with respect to this action, see (1) the application for amendment and request for relief dated October 19, 1977, as supplemented December 15, 1978, (2) Amendment No. 77 to License No. DPR-51, (3) the Commission's related Safety Evaluation and (4) the Commission's letter to the licensee dated April 18, 1983. These items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Arkansas Tech University, Russellville, Arkansas. A copy of items (2), (3) and (4) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland this 18th day of April 1983.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

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