October 3, 1996

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NOS. 185 AND 176 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M95211 AND M95212)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment Nos. 185 and 176 to Facility Operating License Nos. DPR-51 and NPF-6 for the Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2). These amendments consist of changes to the Technical Specifications (TSs) in response to your applications dated April 11, 1996, as supplemented August 23, 1996.

The amendments revise the TSs to permit implementation of the new 10 CFR Part 50, Appendix J, Option B.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:
Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures:
1. Amendment No. 185 to DPR-51
2. Amendment No. 176 to NPF-6
3. Safety Evaluation

cc w/encls: See next page

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Document Name: AR95211.AMD

OFC	PM/PD4-1	LA/PD4-1	OGC	PM/PD4-1
NAME	JKennedy/sp JL	PTressler PT	EHolcer	TAlexion TWA
DATE	9/18/96	9/18/96	9/25/96	10/3/96
COPY	YES/NO	YES/NO	YES/NO	YES/NO

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Mr. C. Randy Hutchinson
Entergy Operations, Inc.

Arkansas Nuclear One, Units 1 & 2

cc:

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

October 3, 1996

Mr. C. Randy Hutchinson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ISSUANCE OF AMENDMENT NOS. 185 AND 176 TO FACILITY OPERATING LICENSE NOS. DPR-51 AND NPF-6 - ARKANSAS NUCLEAR ONE, UNITS 1 AND 2 (TAC NOS. M95211 AND M95212)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment Nos. 185 and 176 to Facility Operating License Nos. DPR-51 and NPF-6 for the Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2). These amendments consist of changes to the Technical Specifications (TSs) in response to your applications dated April 11, 1996, as supplemented August 23, 1996.

The amendments revise the TSs to permit implementation of the new 10 CFR Part 50, Appendix J, Option B.

A copy of our related Safety Evaluation is also enclosed. A Notice of Issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures: 1. Amendment No. 185 to DPR-51
2. Amendment No. 176 to NPF-6
3. Safety Evaluation

cc w/encls: See next page

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-313

ARKANSAS NUCLEAR ONE, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 185
License No. DPR-51

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated April 11, 1996, as supplemented August 23, 1996, 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 license amendment will not be inimical to the common defense and security or to the health and safety of the public; and

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and Paragraph 2.C.(2) of Facility Operating License No. NPF-6 is hereby amended to read as follows:

   (2) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 185, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 185

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

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

REMOVE PAGES	INSERT PAGES
79	79
80	80
81	81
82	82
83	83
84	84
127	127

4.4 REACTOR BUILDING

4.4.1 Reactor Building Leakage Tests

Applicability

Applies to the reactor building.

Objective

To verify that leakage from the reactor building is maintained within allowable limits.

Specification

4.4.1.1 Integrated leakage rate tests shall be conducted and visual inspections performed in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.1.1 Deleted

4.4.1.1.2 Deleted

4.4.1.1.3 Deleted

4.4.1.1.4 Integrated leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.1.5 Deleted

4.4.1.1.6 Deleted

4.4.1.1.7 Deleted

4.4.1.2 Local leakage rate tests shall be conducted in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.2.1 Deleted

4.4.1.2.2 Deleted

4.4.1.2.3 Deleted

4.4.1.2.4 Deleted

4.4.1.2.5 Local leakage rate testing frequencies shall be in accordance with the Reactor Building Leakage Rate Testing Program.

4.4.1.3 Deleted

4.4.1.4 Isolation Valve Functional Tests

Every three months, remotely operated reactor building isolation valves shall be stroked to the position required to fulfill their safety function unless such operation is not practical during plant operation. The latter valves shall be tested once every 18 months.

4.4.1.5 Deleted

Bases (1)

The reactor building is designed for an internal pressure of 59 psig and a steam-air mixture temperature of 285°F.

The peak calculated reactor building pressure for the design basis loss of coolant accident, P_a, is 54 psig. The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

The reactor building will be periodically leakage tested in accordance with the Reactor Building Leakage Rate Testing Program. These periodic testing requirements verify the reactor building leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

REFERENCE

(1) FSAR, Sections 5 and 13.

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

6.8.3 Changes to procedures of 6.8.1 above may be made and implemented prior to obtaining the review and approval required in 6.8.2 above provided:

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

b. The change is approved by two members of the plant management staff, at least one of whom holds a Senior Reactor Operator's license on Unit 1.

c. The change is documented, reviewed and approved as required by the QAMO, within 14 days of implementation.

6.8.4 The Reactor Building Leakage Rate Testing Program shall be established, implemented, and maintained:

A program shall be established to implement the leakage rate testing of the reactor building as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated reactor building internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable reactor building leakage rate, L_a, shall be 0.20% of containment air weight per day at P_a.

Reactor building leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Reactor Building Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Reactor Building Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENTERGY OPERATIONS, INC.

DOCKET NO. 50-368

ARKANSAS NUCLEAR ONE, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 176
License No. NPF-6

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

   A. The application for amendment by Entergy Operations, Inc. (the licensee) dated April 11, 1996, as supplemented August 23, 1996, 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 license 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:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 176, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of its date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas W. Alexion

Thomas W. Alexion, Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: October 3, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 176

FACILITY OPERATING LICENSE NO. NPF-6

DOCKET NO. 50-368

Revise the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

REMOVE PAGES	INSERT PAGES
XVII	XVII
3/4 6-1	3/4 6-1
3/4 6-2	3/4 6-2
3/4 6-3	3/4 6-3
3/4 6-5	3/4 6-5
3/4 6-9	3/4 6-9
B 3/4 6-1	B 3/4 6-1
--	B 3/4 6-1a
B 3/4 6-2	B 3/4 6-2
--	6-26

ADMINISTRATIVE CONTROLS

3/4.6 CONTAINMENT SYSTEM

3/4.6.1 PRIMARY CONTAINMENT

CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.1.1 Primary CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: MODES 1, 2, 3, and 4.

ACTION:

Without primary CONTAINMENT INTEGRITY, restore CONTAINMENT INTEGRITY within one hour or be in at least HOT STANDBY within the next 6 hours and in COLD SHUTDOWN within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1 Primary CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all penetrations* not capable of being closed by OPERABLE containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in their positions, except for valves that are open under administrative control as permitted by Specification 3.6.3.1.

b. By verifying that each containment air lock is OPERABLE per Specification 3.6.1.3.

c. After each closing of the equipment hatch, by leak rate testing the equipment hatch seals in accordance with the Containment Leakage Rate Testing Program.

---

*Except valves, blind flanges, and deactivated automatic valves which are located inside the containment and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except that such verification need not be performed more often than once per 92 days.

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SURVEILLANCE REQUIREMENTS

4.6.1.3.1 Each containment air lock shall be demonstrated OPERABLE as specified in the Containment Leakage Rate Testing Program[5,6].

4.6.1.3.2 Each containment air lock interlock shall be demonstrated OPERABLE by testing the air lock interlock mechanism at least once per 184 days[7].

---

[5] Leakrate results shall also be evaluated against the acceptance criteria of specification 3.6.1.2.

[6] An inoperable air lock door does not invalidate the previous successful performance of the overall air lock leakage test.

[7] This surveillance requirement is only required to be performed upon entry or exit through the associated containment air lock.

4.6.1.5.2 End Anchorages and Adjacent Concrete Surfaces The structural integrity of the end anchorages of all tendons inspected pursuant to Specification 4.6.1.5.1 and the adjacent concrete surfaces shall be demonstrated by determining through inspection that no apparent changes have occurred in the visual appearance of the end anchorage or the concrete crack patterns adjacent to the end anchorages. Inspections of the concrete shall be performed during the Type A containment leakage rate tests (reference Specification 4.6.1.2) while the containment is at its maximum test pressure.

4.6.1.5.3 Containment Surfaces The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation has occurred in accordance with the Containment Leakage Rate Testing Program.

3/4.6.1 PRIMARY CONTAINMENT

3/4.6.1.1 CONTAINMENT INTEGRITY

Primary CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR 100 during accident conditions.

3/4.6.1.2 CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak design basis loss of coolant accident pressure, P_a, of 54 psig. As an added conservatism, the measured overall integrated leakage rate is further limited to $\leq 0.75\ L_a$ during the performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

The surveillance testing for measuring leakage rates are consistent with the requirements of Option B of Appendix "J" of 10 CFR 50.

The containment will be periodically leakage tested in accordance with the Containment Leakage Rate Testing Program. These periodic testing requirements verify the containment leakage rate does not exceed the assumptions used in the safety analysis. At $\leq 1.0\ L_a$ the offsite dose consequences are bounded by the assumptions of the safety analysis. During the first unit startup following testing in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the combined Type B and Type C leakage, and $\leq 0.75\ L_a$ for overall Type A leakage. At all other times between required leakage tests, the acceptance criteria is based on an overall Type A leakage limit of $\leq 1.0\ L_a$.

3/4.6.1.3 CONTAINMENT AIR LOCKS

Each containment air lock forms part of the containment pressure boundary. As part of the containment, the air lock safety function is related to control of the containment leakage rate resulting from a DBA. Thus, each air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event. For the purposes of this specification, the vertical end plates of the air lock barrel, on which the doors themselves are mounted, shall be considered part of the door.

Each air lock is required to be OPERABLE. For the air lock to be considered OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock allows only one air lock door of an air lock to be opened at one time. This provision ensures that a gross breach of containment does not exist when containment is required to be OPERABLE. Closure of a single door in each air lock is sufficient to provide a leak tight barrier following postulated events. Nevertheless, both doors are kept closed when the air lock is not being used for normal entry into and exit from containment.

3/4.6.1.4 INTERNAL PRESSURE, AIR TEMPERATURE AND RELATIVE HUMIDITY

The limitations on containment internal pressure, average air temperature and relative humidity ensure that 1) the containment structure is prevented from exceeding its design negative pressure differential with respect to the outside atmosphere of 5.0 psig, 2) the containment peak pressure does not exceed the design pressure of 54 psig during design basis conditions, and 3) the ECCS analysis assumptions are maintained.

The limitation on containment average air temperature ensures that the containment liner plate temperature does not exceed the design temperature of 300°F during LOCA conditions. The containment temperature limit is consistent with the accident analyses. Figure 3.6-1 represents analysis limits and does not account for instrument error.

3/4.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the facility. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 54 psig in the event of a LOCA. The visual examination of tendons, anchorages and containment surfaces and the Type A leakage tests of the Unit 2 containment in conjunction with the required surveillance activities of the Unit 1 containment are sufficient to demonstrate this capability.

The surveillance requirements for demonstrating the containment's structural integrity are in compliance with the recommendations of Regulatory Guide 1.35 "Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures", January 1976.

3/4.6.1.6 CONTAINMENT VENTILATION SYSTEM

The containment purge supply and exhaust isolation valves are required to be closed during plant operation since these valves have not been demonstrated capable of closing during a LOCA. Maintaining these valves closed during plant operations ensures that excessive quantities of radioactive materials will not be released via the containment purge system.

6.15 CONTAINMENT LEAKAGE RATE TESTING PROGRAM

A program shall be established to implement the leakage rate testing of the containment as required by 10 CFR 50.54(o) and 10 CFR 50, Appendix J, Option B, as modified by approved exemptions. This program shall be in accordance with the guidelines contained in Regulatory Guide 1.163, "Performance-Based Containment Leak-Test Program," dated September 1995.

The peak calculated containment internal pressure for the design basis loss of coolant accident, P_a, is 54 psig.

The maximum allowable containment leakage rate, L_a, shall be 0.1% of containment air weight per day at P_a.

Leakage rate acceptance criteria are:

a. Containment leakage rate acceptance criteria is $\leq 1.0\ L_a$. During the first unit startup following each test performed in accordance with this program, the leakage rate acceptance criteria are $\leq 0.60\ L_a$ for the Type B and Type C tests and $\leq 0.75\ L_a$ for Type A tests.

b. Air lock acceptance criteria are:

   1. Overall air lock leakage rate is $\leq 0.05\ L_a$ when tested at $\geq P_a$.

   2. Leakage rate for each door is $\leq 0.01\ L_a$ when pressurized to ≥ 10 psig.

The provisions of Specification 4.0.2 do not apply to the test frequencies specified in the Containment Leakage Rate Testing Program.

The provisions of Specification 4.0.3 are applicable to the Containment Leakage Rate Testing Program.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 185 AND 176 TO

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

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NOS. 1 AND 2

DOCKET NOS. 50-313 AND 50-368

1.0 INTRODUCTION

By letter dated April 11, 1996, as supplemented August 23, 1996, Entergy Operations, Inc. (the licensee) submitted a request for changes to the Arkansas Nuclear One, Unit Nos. 1 and 2 (ANO-1&2), Technical Specifications (TSs). The requested changes would allow the implementation of the recently approved Option B to 10 CFR Part 50, Appendix J, which allows for a performance-based option for determining the frequency for containment leakage rate testing.

The August 23, 1996, letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

Compliance with 10 CFR Part 50, Appendix J, "Primary Containment Leakage Testing for Water-Cooled Power Reactors," provides assurance that the primary containment, including those systems and components which penetrate the primary containment, do not exceed the allowable leakage rate specified in the TSs and Bases. The allowable leakage rate is determined so that the leakage rate assumed in the safety analyses is not exceeded.

On February 4, 1992, the NRC published a notice in the Federal Register (57 FR 4166) discussing a planned initiative to begin eliminating requirements marginal to safety which impose a significant regulatory burden. Title 10 of the Code of Federal Regulations, Part 50, Appendix J was considered for this initiative and the staff undertook a study of possible changes to this regulation. The study examined the previous performance history of domestic containments and examined the effect on risk of a revision to the requirements of Appendix J. The results of this study are reported in NUREG-1493, "Performance-Based Leak-Test Program."

Based on the results of this study, the staff developed a performance-based approach to containment leakage rate testing. On September 12, 1995, the NRC approved issuance of a revision to 10 CFR Part 50, Appendix J, which was subsequently published in the Federal Register on September 26, 1995, and became effective on October 26, 1995. The revision added Option B, "Performance-Based Requirements," to Appendix J to allow licensees to voluntarily replace the prescriptive testing requirements of Appendix J with testing requirements based on both overall and individual component leakage rate performance.

Regulatory Guide (RG) 1.163, "Performance-Based Containment Leak Test Program," dated September 1995, was developed as a method acceptable to the staff for implementing Option B. This RG states that the Nuclear Energy Institute (NEI) guidance document, NEI 94-01, Rev. 0, "Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J," provides methods acceptable to the staff for complying with Option B, with the four exceptions described herein.

Option B requires that the RG or other implementation document used by a licensee to develop a performance-based leakage testing program must be included by general reference in the plant TSs. The licensee has referenced RG 1.163 dated September 1995 in the proposed ANO-1&2 TSs.

RG 1.163 specifies an extension in Type A test frequency to at least one test in 10 years based upon two consecutive successful tests. Type B tests may be extended up to a maximum interval of 10 years based upon completion of two consecutive successful tests and Type C tests may be extended up to 5 years based on two consecutive successful tests.

By letter dated October 20, 1995, NEI proposed TSs to implement Option B. After some discussion, the staff and NEI agreed on the final TSs which were transmitted to NEI in a letter dated November 2, 1995. These TSs serve as a model for licensees to develop plant-specific TSs in preparing amendment requests to implement Option B.

In order for a licensee to determine the performance of each component, factors that are indicative of or affect performance, such as an administrative leakage limit, must be established. The administrative limit is selected to be indicative of the potential onset of component degradation. Although these limits are subject to NRC inspection to assure that they are selected in a reasonable manner, they are not TS requirements. Failure to meet an administrative limit requires the licensee to return to the minimum value of the test interval.

Option B requires that the licensee maintain records to show that the criteria for Type A, B, and C tests have been met. In addition, the licensee must maintain comparisons of the performance of the overall containment system and the individual components to show that the test intervals are adequate. These records are subject to NRC inspection.

3.0 EVALUATION

3.1 ANO-1 Proposed TSs

Option B permits a licensee to choose Type A; Type B and C; or Type A, B and C testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis for ANO-1.

The licensee's application proposes to establish a "Reactor Building Leakage Rate Testing Program," which references RG 1.163, and adds this program as new TS 6.8.4. The addition of this program requires a change to existing TSs 4.4.1.1, 4.4.1.1.4, 4.4.1.2, 4.4.1.2.5, the TS index, and the associated Bases sections.

The changes proposed by the licensee are in compliance with the requirements of 10 CFR Part 50, Appendix J and consistent with the guidance in RG 1.163. Despite the different format of the licensee's current TSs, all of the important elements of the guidance provided in the staff's letter to NEI dated November 2, 1995, are included. However, the licensee has proposed several changes that are in addition to the model TSs or that warrant further discussion.

The action statement of current ANO-1 TS 3.6.1, "Reactor Building," is entered if Type A or Type B and C leakage rates from TSs 4.4.1.1 and 4.4.1.2, respectively, are not within limits. Current TS 3.6.1 allows 6 hours to reach hot standby from power operation if containment integrity is lost, versus 12 hours in the model TSs. Although the licensee has chosen not to adopt the model TSs, the noted deviation is conservative and is therefore acceptable.

The licensee has proposed not to adopt the individual model TSs for the air lock leakage rates, and has proposed not to include the individual air lock leakage criteria as part of the Reactor Building Leakage Rate Testing Program referenced in new TS 6.8.4. Instead, air lock leakage rate will be included in the overall Type B and C leakage rate, consistent with the current TSs. The acceptance criteria located in TS 4.4.1.2.3 states, "the total leakage from all tested penetrations and isolation valves shall not exceed 60% L_a." Section 6.8.4 of the proposed change maintains the requirement for the air locks to be Type B tested with the same acceptance criteria of $\leq$ 60% L_a for the total leakage from all Type B and C tests. Because the proposed combined Type B and C leakage rates are the same as the current and model TSs, the proposed change is acceptable.

TS 4.4.1.1.1, 4.4.1.1.2, 4.4.1.1.3, 4.4.1.1.5, 4.4.1.1.6, 4.4.1.1.7, 4.4.1.2.1, 4.4.1.2.2, 4.4.1.2.3, 4.4.1.2.4, 4.4.1.3, and 4.4.1.5 will be deleted and the information they contain, where applicable, will be added to the Reactor Building Leakage Rate Testing Program. The deleted TSs contain information concerning the specific conduct of tests, acceptance criteria, reporting of results, corrective actions, and visual examinations. Removal of explicit test details and reporting requirements is consistent with the model TSs. Furthermore, visual inspections are now required by proposed TS 4.4.1.1, and corrective actions are given in current TS 3.6.1. Since the proposed changes are consistent with the model TSs and do not constitute a failure to adopt or a relaxation of Option B requirements, the staff finds the proposed changes acceptable.

The licensee has proposed deleting parts of the Bases to TS 4.4.1. These portions contain information regarding the frequency of testing and testing details. This information is now superseded by Option B and therefore no longer applicable or is contained in the Reactor Building Leakage Rate Testing Program. In its place, text consistent with the model TSs has been added. Because the proposed changes remove inapplicable information and are consistent with the model TSs, the proposed changes are acceptable.

The Reactor Building Leakage Testing Program will be added as new TS 6.8.4. With the exception of not adopting the specific leakage criteria for air locks, the acceptability of which has been discussed in a preceding paragraph, the adopted TSs are consistent with the model. The proposed addition of TS 6.8.4 is therefore acceptable.

3.2 ANO-2 Proposed TSs

Option B permits a licensee to choose Type A; Type B and C; or Type A, B and C testing to be done on a performance basis. The licensee has elected to perform Type A, B and C testing on a performance basis for ANO-2.

The licensee's application proposes to establish new TS Section 6.15, "Containment Leakage Rate Testing Program," which references RG 1.163. The addition of this program requires a change to existing TSs 3/4.6.1.1, 3/4.6.1.2, 3/4.6.1.3.1, 3/4.6.1.3.2, 3/4.6.1.5.3, the TS index, and associated Bases.

The TS changes proposed by the licensee are in compliance with the requirements of 10 CFR Part 50, Appendix J, Option B, and consistent with the guidance in RG 1.163. Despite the different format of the licensee's current TSs, all of the important elements of the guidance provided in the staff's letter to NEI dated November 2, 1995, are included in the proposed TSs. However, the licensee has proposed several changes that are in addition to the model TSs or that warrant further discussion.

The action statement for current TS 3.6.1.1, "Containment Integrity," allows 6 hours to reach hot standby from power operation if containment integrity is lost. The model TSs would allow 12 hours. This is a conservative deviation and is therefore acceptable.

The action statement for current TS 3.6.1.2, "Containment Leakage," requires that with containment leakage rates not within limits, restore the leakage rates to within the limits "prior to increasing the reactor coolant temperature above 200°F." The licensee intends to maintain this wording. Model TS 3.6.1.1 requires returning containment to operable within 1 hour, or placing the unit in hot shutdown within 12 hours and cold shutdown within

36 hours. While the model TSs correct a deficiency in the current TS which do not recognize that containment leakage rates can be determined during plant operation (Modes 1 through 4), keeping the current TSs is still adequately restrictive. This is because limiting condition for operation (LCO) 3.0.3 of the current TSs, which is entered when an action of a particular specification cannot be entered because of circumstances in excess of those addressed in the specification, would apply if leakage were determined to be exceeded during plant operations. LCO 3.0.3 requires initiating action within 1 hour to place the unit in hot standby in 6 hours, in hot shutdown in the next 6 hours, and in cold shutdown within the next 24 hours. Because the required actions in the ANO-2 TSs are equivalent to the model TSs, the staff finds this deviation acceptable.

ANO-2 TS 4.6.1.5.3, which requires visual examination of the accessible interior and exterior surfaces of containment, including the liner plate, is being revised. The current TSs state:

> "The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined during shutdown for each Type A containment leak rate test (reference Specification 4.6.1.2) by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation."

Proposed TS 4.6.1.5.3 would state:

> "The structural integrity of the exposed accessible interior and exterior surfaces of the containment, including the liner plate, shall be determined by a visual inspection of these surfaces and verifying no apparent changes in appearance or other abnormal degradation has occurred in accordance with the Containment Leakage Rate Testing Program."

While the proposed format is somewhat different than the model TSs, it preserves the structure of the current TSs and is consistent with RG 1.163. The staff, therefore, finds the proposed change acceptable.

The Bases for TS 3/4.6.1.5 were changed to reflect the most current maximum containment pressure in the event of a loss of coolant accident. The Bases for TS 3/4.6.1.2 were modified to explain the leakage acceptance criteria and eliminate information regarding low pressure testing of the containment because it is no longer being allowed by Option B. In addition, a reference to Option B of 10 CFR Part 50, Appendix J, was added for clarity. The Bases for TS 3/4.6.1.3 were expanded by adding clarifying information and removing the old Bases information that is repetitive. The staff finds these changes acceptable.

3.2 Conclusion

The staff has reviewed the changes to both the ANO-1&2 TSs and associated Bases proposed by the licensee and finds that they are in compliance with the requirements of 10 CFR Part 50, Appendix J, Option B, and consistent with the guidance of RG 1.163. The staff finds the proposed changes acceptable.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Arkansas State official was notified of the proposed issuance of the amendment. The State official had no comment.

5.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration and there has been no public comment on such finding (61 FR 20846). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

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Date: October 3, 1996