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

September 9, 1999

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REACTOR BUILDING STRUCTURAL INTEGRITY SURVEILLANCE REQUIREMENTS (TAC NO. MA5216)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 199 to Facility Operating License No. DPR-51 for the Arkansas Nuclear One, Unit No. 1. The amendment consists of changes to the Technical Specifications (TSs) in response to your application dated April 9, 1999, as supplemented by letter dated July 14, 1999.

The amendment revises the requirements associated with reactor building testing and inspection. The majority of the changes involved deletion of the existing requirements for tendon surveillance and substituting the inspection in accordance with the requirements of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section XI, Subsection IWL and Section 50.55a(g)(6)(ii)(B) of Title 10 of the *Code of Federal Regulations* (10 CFR). You also proposed a change to the reporting requirements.

The staff notes that approval of the proposed TS changes does not relieve you of responsibility to report, pursuant to 10 CFR 50.73(a)(2)(ii), any event or condition that results in the condition of the nuclear power plant being seriously degraded. In this case, these conditions may include serious degradation of the containment concrete structure, such as dome delamination, multi-wire or anchor-head failures, and widespread corrosion of the liner plate.

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

Sincerely,

ORIGINAL SIGNED BY

Nicholas D. Hilton, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosures: 1. Amendment No. 199 to DPR-51
2. Safety Evaluation

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Arkansas Nuclear One

cc:
Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Director, Division of Radiation
Control and Emergency Management
Arkansas Department of Health
4815 West Markham Street, Slot 30
Little Rock, AR 72205-3867

Winston & Strawn
1400 L Street, N.W.
Washington, DC 20005-3502

Manager, Rockville Nuclear Licensing
Framatone Technologies
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Senior Resident Inspector
U.S. Nuclear Regulatory Commission
P. O. Box 310
London, AR 72847

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 400
Arlington, TX 76011-8064

County Judge of Pope County
Pope County Courthouse
Russellville, AR 72801

Vice President, Operations Support
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

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. 199
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 9, 1999, as supplemented by letter dated July 14, 1999, 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. 199, 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 and shall be implemented within 30 days from the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A Gramm

Robert A. Gramm, Chief, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: September 9, 1999

ATTACHMENT TO LICENSE AMENDMENT NO. 199

FACILITY OPERATING LICENSE NO. DPR-51

DOCKET NO. 50-313

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
ii	ii
iv	iv
54	54
55	55
80	80
81	---
82	---
83	---
84	---
85	---
85a	---
85b	---
85c	---
85d	---
86	---
87	---
88	---
89	---
90	---
91	---
146	146
146a	146a

LIST OF FIGURES

3.6 REACTOR B' ING

Applicability

Applies to the operability of the reactor building.

Objective

To assure reactor building operability.

Specification

3.6.1 The reactor building shall be operable whenever all three (3) of the following conditions exist:

a. Reactor coolant pressure is 300 psig or greater.

b. Reactor coolant temperature is 200°F or greater.

c. Nuclear fuel is in the core.

With the reactor building inoperable, restore the reactor building to operable status within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.2 Reactor building integrity shall be maintained when the reactor coolant system is open to the reactor building atmosphere and the requirements for a refueling shutdown are not met. The provisions of Specification 3.0.3 are not applicable.

3.6.3 Positive reactivity insertions which would result in the reactor being subcritical by less than 1% Δk/k shall not be made by control rod motion or boron dilution whenever reactor building integrity is not in force. The provisions of Specification 3.0.3 are not applicable.

3.6.4 The reactor shall not be taken critical or remain critical if the reactor building internal pressure exceeds 3.0 psig or a vacuum of 5.5 inches Hg. With the reactor critical, restore the containment pressure to within its limits within one hour or be in at least Hot Standby within the next 6 hours and in Cold Shutdown within the following 30 hours.

3.6.5 Prior to criticality following a refueling shutdown, a check shall be made to confirm that all manual reactor building isolation valves which should be closed are closed and locked, as required. The provisions of Specification 3.0.3 are not applicable.

3.6.6 If, while the reactor is critical, a reactor building isolation valve is determined to be inoperable in a position other than the closed position, the other reactor building isolation valve (except for check valves) in the line shall be tested to insure operability. If the inoperable valve is not restored within 48 hours, the reactor shall be brought to the cold shutdown condition within an additional 24 hours or the operable valve will be closed.

Bases

Included in reactor building operability are both the reactor building integrity as defined in Specification 1.7 and the reactor building structural integrity. Structural integrity limitations as described in the ANO Containment Inspection Program ensure the reactor building will be maintained comparable to the original design standards throughout the facility life span. Visual and other required examinations of tendons, anchorages and surfaces are performed periodically in accordance with station procedures. These procedures embody applicable requirements of the 1992 Edition with the 1992 Addenda of Section XI, Subsection IWL of the ASME Boiler and Pressure Vessel Code as set forth in 10 CFR 50.55a(g)(6)(ii)(B). Any degradations exceeding the Containment Inspection Program acceptance criteria during inspection surveillances will be reviewed under an engineering evaluation within 60 days of the completion of the inspection to determine what impact the degradation has on overall containment operability, if any.

The reactor coolant system conditions of cold shutdown assure that no steam will be formed and hence there will be no pressure buildup in the reactor building if the reactor coolant system ruptures.

The selected shutdown conditions are based on the type of activities that are being carried out and will preclude criticality in any occurrence.

The reactor building is designed for an internal pressure of 59 psig and an external pressure 3.0 psi greater than the internal pressure. The design external pressure of 3.0 psi corresponds to a margin of 0.5 psi above the differential pressure that could be developed if the building is sealed with an internal temperature of 110°F and the building is subsequently cooled to an internal temperature of less than 50°F.

When reactor building integrity is established, the limits of 10 CFR 100 will not be exceeded should the maximum hypothetical accident occur.

REFERENCE

FSAR, Section 5.

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.

6.12.4 Reactor Building Inspection Report

6.12.4.1 Any degradation exceeding the acceptance criteria of the containment structure detected during the tests required by the ANO Containment Inspection Program shall undergo an engineering evaluation within 60 days of the completion of the inspection surveillance. The results of the engineering evaluation shall be reported to the NRC within an additional 30 days of the time the evaluation is completed. The report shall include the cause of the condition that does not meet the acceptance criteria, the applicability of the conditions to the other unit, the acceptability of the concrete containment without repair of the item, whether or not repair or replacement is required and, if required, the extent, method, and completion date of necessary repairs, and the extent, nature, and frequency of additional examinations.

6.12.5 Special Reports

Special reports shall be submitted to the Administrator of the appropriate Regional Office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable reference specification.

a. Deleted

b. Inoperable Containment Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

c. Deleted

d. Steam Generator Tubing Surveillance - Category C-3 Results, Specification 4.18.

e. Miscellaneous Radioactive Materials Source Leakage Tests, Specification 3.12.2.

f. Deleted

g. Deleted

h. Deleted

i. Deleted

j. Degraded Auxiliary Electrical Systems, Specification 3.7.2.H.

k. Inoperable Reactor Vessel Level Monitoring Systems, Table 3.5.1-1

l. Inoperable Hot Leg Level Measurement Systems, Table 3.5.1-1

m. Inoperable Main Steam Line Radiation Monitors, Specification 3.5.1, Table 3.5.1-1.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 199 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

On January 7, 1994, the Nuclear Regulatory Commission (NRC) published a proposed amendment to the regulations to incorporate by reference the 1992 Edition with the 1992 Addenda of Subsections IWE and IWL of Section XI, Division 1 of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code). The final rule, Section 50.55a(g)(6)(ii)(B) of Title 10 of the *Code of Federal Regulations* (10 CFR), became effective on September 9, 1996, and requires licensees to implement Subsections IWE and IWL, with specified modifications and limitations, by September 9, 2001.

In its letter of April 9, 1999, as supplemented by letter dated July 14, 1999, Entergy Operations, Inc. (the licensee), submitted an amendment to the Technical Specifications (TSs) for Arkansas Nuclear One, Unit 1 (ANO-1). The proposed changes revise the licensee's TS so that it conforms to the new regulatory requirements. The majority of the changes involve deletion of the existing requirements for tendon surveillance and substituting inspection in accordance with the requirements of the ASME Code, Section XI, Subsection IWL and 10 CFR 50.55a(g)(6)(ii)(B). The licensee has also proposed a change to the reporting requirements. The licensee's program incorporating Subsection IWE has not been completed and is not included in its submittal.

The July 14, 1999, letter provided clarifying information that did not change the scope of the April 9, 1999, application and the initial proposed no significant hazards consideration determination.

The licensee proposes the following changes to its current TS:

Change 1 The TS Table of Contents and Lists of Figures have been updated to reflect the changes made to ANO-1 specifications relevant to the submittals.

Change 2 Terminology used in TS 3.6.1, its action statement, and the Objective of Specification 3.6 have been revised to permit the inclusion of additional requirements beyond those defined in TS 1.7, "Reactor Building."

Enclosure

Change 3 The footer on Page 80 of the TSs has been modified to inform the user that several of the following pages have been deleted.

Change 4 The specific surveillance criteria of TS 4.4.2 has been deleted from the specifications and incorporated in ANO-1's containment inspection program.

Change 5 Figures 4.4.2-1, 4.4.2-2, and 4.4.2-3 have been deleted from the specifications and relocated to ANO-1's containment inspection program.

Change 6 The requirements of TS 6.12.5.a have been deleted. Reporting requirements associated with this program are incorporated in the containment inspection program.

Change 7 TS 6.12.4 has been added to the Administrative Controls section of the TSs. It states the reporting requirements associated with the ANO-1 containment inspection program.

Change 8 The Bases for TS 3.6 have been revised to clarify the reactor building inspection evaluation requirements.

The adequacy of the proposed changes are discussed below.

2.0 EVALUATION

Change 1

The TS Table of Contents and Lists of Figures have been updated to reflect the changes made to ANO-1 specifications relevant to the submittals. These changes are editorial, consistent with the technical changes, and therefore, acceptable.

Change 2

Terminology used in TS 3.6.1, its action statement, Objective of Specification 3.6, and its associated Bases, have been revised to permit the inclusion of additional requirements beyond those defined in TS 1.7, "Reactor Building." The terminology of this section has been changed from "maintaining reactor building integrity" to stating that the "reactor building shall be operable." By doing this, the requirements of structural integrity are included in addition to the requirements of maintaining the integrity of the reactor building. This is acceptable because it clarifies the requirements of this section.

Change 3

The footer on Page 80 of the TSs has been modified to inform the user that several of the following pages have been deleted. This editorial change is acceptable.

Changes 4 and 5

The specific surveillance criteria of TS 4.4.2, including Figures 4.4.2-1, 4.4.2-2, and 4.4.2-3, have been deleted from the specifications and incorporated into the ANO-1 containment inspection program. Included in TS 4.4.2 are:

- TS 4.4.2.1, Tendon Surveillance

  This specification requires testing of 21 tendons at 1-, 3-, and 5-year intervals. Successful completion of these tests allow decreasing the required number of tendons tested per interval to nine. The licensee has completed testing of the 21 tendons at the 1-, 3-, and 5-year intervals as evidenced by its letters of September 11, 1975, October 4, 1977, and August 22, 1979. The remaining requirements of this specification were modified to comply with the requirements of Subsection IWL of the ASME Code, 10 CFR 50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). These requirements are contained in ANO-1's containment inspection program.

  The requirements of TS 4.4.2.1 have either been fulfilled or are redundant to the requirements contained in Subsection IWL, 10 CFR 50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). Therefore, deletion of this specification is acceptable.

- TS 4.4.2.1.1, Lift Off

  This specification states that lift off readings shall be taken for all surveillance tendons. This requirement is redundant to the requirements contained in Subsection IWL, 10 CFR 50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). Therefore, deletion of this specification is acceptable.

- TS 4.4.2.1.2, Wire Inspection Testing

  This specification describes the required testing of wires from surveillance tendons and also the applicable anchor assemblies. This requirement is redundant to the requirements contained in Subsection IWL, 10 CFR 50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). Therefore, deletion of this specification is acceptable.

- TS 4.4.2.1.3, Acceptance Criteria, and Figures 4.4.2-1, 4.4.2-2, and 4.4.2-3

  In this specification, the acceptance criteria for reactor building post tensioning system is listed. Figures 4.4.2-1, 4.4.2-2, and 4.4.2-3 are directly incorporated into the containment inspection program. The requirements of Subsection IWL and 10 CFR 50.55a(g)(6)(ii)(B) encompass the intent of this specification and provide greater detail in stating the acceptance criteria. Therefore, deletion of this TS is acceptable.

- TS 4.4.2.2, Inspection Interval and Reports

  This specification describes the requirements for the inspection interval and states that a special report be submitted within 90 days from completion of the inspection. Both of these requirements are addressed in either Subsection IWL or IWA, 10 CFR

50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). Because of the redundancy, deletion of this specification is acceptable.

- TS 4.4.2.3, End Anchorage Concrete Surveillance

  This specification describes the interval and locations for inspection of the end anchorages of the surveillance tendons and adjacent concrete surface. The requirements associated with this specification have been fulfilled and no longer pertain to current testing regulations. Therefore, deletion of the TS is acceptable.

- TS 4.4.2.4, Liner Plate Surveillance

  This specification describes the examination of the liner plate prior to the initial pressure test. The requirement for liner plate surveillance is included in Subsection IWL, 10 CFR 50.55a(g)(6)(ii)(B), and 50.55a(b)(2)(ix). Therefore, deletion of the TS is acceptable.

Changes 6, 7, and 8

The licensee proposes to delete the requirements of TS 6.12.5.a, "Special Reports, Tendon Surveillance," and add the requirement, TS 6.12.4, "Reactor Building Inspection Report." TS 6.12.4 addresses the reporting requirements for the containment building inspection surveillance. The licensee proposes that degradations that exceed the acceptance criteria of 10 CFR 50.55a(b)(2)(ix) or Subsection IWL will be reported to the NRC within 30 days upon completion of the engineering evaluation. The Bases for TS 3.6 have been revised to require any resulting engineering evaluation be performed within 60 days of completion of the containment building inspection surveillance. The combination of TS 6.12.4 and the Bases to TS 3.6 will ensure that a report on the condition of the reactor building structure and any corrective actions that are required as a result of the inspection, is submitted to the NRC within 90 days. This is acceptable. However, approval of the proposed TS changes does not relieve the licensee of its responsibility to report, pursuant to 10 CFR 50.73(a)(2)(ii), any event or condition that results in the condition of the nuclear power plant being seriously degraded. These conditions include serious degradation of the containment concrete structure, such as dome delamination, multi-wire or anchor-head failures, and widespread corrosion of the liner plate.

3.0 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 comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendment involves 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 amendment involves no significant hazards consideration, and there has been no public comment on such finding (64 FR 27320 dated May 19, 1999). The amendment also changes recordkeeping, reporting, or administrative procedures or requirements. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 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 amendment will not be inimical to the common defense and security or to the health and safety of the public.

Approval of the proposed TS changes does not relieve the licensee of its responsibility to report, pursuant to 10 CFR 50.73(a)(2)(ii), any event or condition that results in the condition of the nuclear power plant being seriously degraded. These conditions include serious degradation of the containment concrete structure, such as dome delamination, multi-wire or anchor-head failures, and widespread corrosion of the liner plate.

Principal Contributor: M. Kotzalas

Date: September 9, 1999