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Subject: Arkansas Nuclear One - Unit 1
Docket No. 50-313
License No. DPR-51
Application for Exemption from 10CFR50 Appendix J and Technical Specification Change Request Concerning Type A Test Frequency

Gentlemen:

In accordance with 10CFR50.12 and 10CFR50.90, Entergy Operations hereby applies for an exemption from 10CFR50 Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors," regarding integrated leak rate testing intervals, and the technical specification change necessary to implement the exemption at Arkansas Nuclear One, Unit 1 (ANO-1). This exemption and change would provide a one-time interval extension for a Type A test from the currently scheduled 36 months to 54 months. This extension would result in the delay of the integrated leak rate test (ILRT) currently scheduled for the twelfth refueling outage (1R12) until the thirteenth refueling outage (1R13). The exemption would also have the effect of extending the 10-year ILRT interval to correspond to the end of the recently extended ANO-1 inservice inspection/inservice testing (ISI/IST) interval.

Proposed submittals from Entergy Operations concerning Appendix J testing intervals were discussed in a meeting with the NRC staff on June 8, 1993, as part of a list of proposed cost beneficial licensing actions (CBLAs). Since that time, significant progress has been made on a proposed rulemaking to revise Appendix J in a performance based manner similar to Entergy Operation's original intent. Because of the currently anticipated approval schedule for the proposed rulemaking, Entergy Operations no longer considers it necessary to request the entire scope of changes to Appendix J testing for ANO-1 discussed at the June 8, 1993, meeting or included in the related Grand Gulf submittal dated August 13, 1993. The proposed changes are limited to that aspect of compliance with Appendix J which will provide a significant hardship without a corresponding increase in safety in the 1R12 refueling outage, i.e., performance of the scheduled ILRT. By obtaining an exemption to the current Appendix J testing frequency requirements, the scheduled date for the next Type A test will occur after approval of the proposed revision to Appendix J, thereby allowing Entergy Operations to take advantage of the anticipated performance based criteria. On August 2, 1994, the NRC approved a request from Entergy Operations which had a similar basis to this request;

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approval to extend the current ISI/IST interval at ANO-1 to save the cost expenditures associated with a potentially unnecessary program revision, pending the completion of a proposed rulemaking. Entergy Operations estimates that approving this current CBLA request to extend the ILRT testing interval will save in excess of $1,350,000 at ANO-1.

The exemption request has been found to meet the requirements of sections (a)(2)(ii) and (a)(2)(vi) of 10CFR50.12. Additionally, the proposed technical specification change has been evaluated in accordance with 10CFR50.91(a)(1) using criteria in 10CFR50.92(c) and it has been determined that this change involves no significant hazards considerations. The bases for these determinations are included in the attached submittal.

Entergy Operations requests that the effective date for this technical specification change and exemption be the date of NRC approval, but not later than January 30, 1995, in order to facilitate scheduling for the 1R12 outage. Although this request is neither exigent nor emergency, your prompt review is requested.

Very truly yours,

Jerry W. Yelverton

JWY/jjd
Attachments

To the best of my knowledge and belief, the statements contained in this submittal are true.

SUBSCRIBED AND SWORN TO before me, a Notary Public in and for Johnson **County and the State of Arkansas, this** 30 **day of** August**, 1994.**

Juana M. Tapp
Notary Public
My Commission Expires 11-8-2000

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ATTACHMENT

TO

1CAN089405

PROPOSED TECHNICAL SPECIFICATION

AND

RESPECTIVE SAFETY ANALYSES

IN THE MATTER OF AMENDING

LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT ONE

DOCKET NO. 50-313

DESCRIPTION OF PROPOSED CHANGES

"Type A Tests" are defined in Appendix J Section II.F as "tests intended to measure the primary reactor containment overall integrated leakage rate."

Exemption is requested from the following portion of Appendix J, Section III.D.1.(a) for Type A test intervals:

> "Type A tests shall be performed, at approximately equal intervals during each 10-year service period. The third test of each set shall be conducted when the plant is shutdown for the 10-year plant inservice inspections."

The exemption will allow the interval between the second and third ILRT of the second period to be extended from 36 months to 54 months, while at the same time extending the end of the ILRT period to match that of the recently revised ANO-1 ISI/IST interval (1CNA089402, dated August 2, 1994).

ANO-1 Technical Specifications (TS) 4.4.1.1.4 is being revised to reference 10CFR50, Appendix J directly, rather than to paraphrase the regulation, and to allow approved exemptions to the ILRT frequency requirements.

BACKGROUND

The reactor building is a Seismic Class 1, fully continuous, reinforced prestressed concrete structure in the shape of a cylinder with a shallow domed roof and a flat foundation slab. The cylindrical portion is prestressed by a post tensioning system consisting of horizontal and vertical tendons. The dome has a 3-way post-tensioning system. Hoop tendons are placed in 3-240 degree systems using three buttresses as anchorages. The foundation slab is conventionally reinforced with high strength reinforcing steel and is founded on bedrock. A continuous access gallery is provided beneath the base slab for installation of vertical tendons. A welded steel liner is attached to the inside face of the concrete shell to insure a high degree of leak tightness. The base liner is installed on top of the structural slab and is covered with concrete. The structure provides shielding for both normal and accident conditions. The reactor building completely encloses the entire reactor and reactor coolant system (RCS). This insures that an acceptable upper limit for leakage of radioactive materials to the environment will not be exceeded even if gross failure of the RCS occurs. Further information on the reactor building design can be found in Section 5.2 of the ANO-1 Safety Analysis Report (SAR).

The intent of the Type A test (i.e., ILRT) is to determine that the total leakage from containment does not exceed the maximum allowable leakage rate (L_a) as specified in ANO-1 TS 4.4.1.1.1, and the ANO-1 SAR. The ANO-1 reactor building design maximum allowable leakage rate (L_a) measured in weight percent/24 hours at the design pressure of 59 psig (ANO-1 TS require the ILRT to be performed at design pressure instead of peak accident pressure) is 0.200%/day. ANO-1 TS 4.4.1.1.5 and 10CFR50, Appendix J require the

measured Type A test acceptance criteria to be less than or equal to 75% of L_a, or 0.150%/day, to allow for deterioration of leakage paths between tests. The maximum allowable leakage rate provides an input assumption to the calculation required to ensure that the maximum allowable offsite dose during a design basis accident does not exceed that specified in 10CFR100.

The safety aspects of this request are similar to those approved on February 16, 1994, for a one time ILRT interval extension and technical specification amendment (amendment 67) at the Limerick Generating Station, Unit 1.

Type A Testing History

Six Type A tests have been conducted at ANO-1, with five being conducted at design pressure, 59 psig. Two different testing methods were employed in the performance of these tests; the mass point leakage rate method and the total time leakage rate method. Only the total time method was utilized for the preoperational ILRT. Trends of previous test results indicate that the proposed 18 month extension would not jeopardize the ability of the reactor building to maintain the leakage rate at or below the required Type A limits. The results of the individual ANO-1 Type A tests follow:

Preoperational Type A Test

The preoperational Type A test was successfully performed on November 8-13, 1973 with the following results: 1) A total time leakage rate of 0.0815%/day at a full test pressure of 59 psig, and 2) A total time leakage rate of 0.0292%/day at a reduced test pressure of 30 psig and a calculated acceptance limit of 0.107%/day. The ILRT test report was submitted to the NRC by letter dated March 8, 1974 (1CAN037404).

First Periodic Type A Test

The first periodic Type A test was successfully performed on March 10-12, 1978 with the following results: 1) A calculated mass point leakage rate of 0.009%/day and a 95% upper confidence limit (UCL) of 0.014%/day, and 2) A total time leakage rate of 0.012%/day and a 95% UCL of 0.069%/day. Since this test was performed at a reduced pressure of 30 psig, the allowable leakage rate was reduced to 0.054%/day. The ILRT test report was submitted to the NRC by letter dated July 8, 1978 (1CAN077803).

Second Periodic Type A Test

The second periodic Type A test was successfully performed at design pressure on February 24-25, 1981 with the following results: 1) A calculated mass point leakage rate of 0.038%/day and a 95% UCL of 0.046%/day, and 2) A total time leakage rate of 0.034%/day and a 95% UCL of 0.092%/day. The ILRT test report was submitted to the NRC by letter dated May 22, 1981 (1CAN058109).

Third Periodic Type A Test

The third periodic Type A test was successfully performed at design pressure on December 14-17, 1984 with the following results: 1) A calculated mass point leakage rate of 0.041%/day and a 95% UCL of 0.046%/day, and 2) A total time leakage rate of 0.031%/day and a 95% UCL of 0.103%/day. This test corresponded to the end of the first ILRT 10-year testing period. The ILRT test report was submitted to the NRC by letter dated March 21, 1985 (1CAN038507), as supplemented by letter dated July 15, 1985 ((1CAN078504).

Fourth Periodic Type A Test

The fourth periodic Type A test was successfully performed at design pressure on November 8-9, 1988 with the following results: 1) An as found calculated mass point leakage rate of 0.081%/day and a 95% UCL of 0.084%/day, 2) An as left calculated mass point leakage rate of 0.077%/day and a 95% UCL of 0.080%/day, 3) An as found total time leakage rate of 0.090%/day and a 95% UCL of 0.118%/day, and 4) An as left total time leakage rate of 0.086%/day and a 95% UCL of 0.114%/day. The ILRT test report was submitted to the NRC by letter dated March 21, 1989 (1CAN038913).

Fifth Periodic Type A Test

The fifth periodic Type A test was performed at design pressure on April 13-16, 1992 with the following results: 1) The as found leakage rate was greater than L_a due to the local leakage rate adjustments specified by NRC IE Information Notice No. 85-71 from leakage at one penetration, 2) An as left calculated mass point leakage rate of 0.0738%/day and a 95% UCL of 0.0844%/day, 3) An as found total time leakage rate of 0.131%/day without including the one penetration with a large leakage rate, and 4) An as left total time leakage rate of 0.0738%/day and a 95% UCL of 0.1248%/day. The ILRT test report was submitted to the NRC by letter dated June 25, 1992 (1CAN069203).

The failure of the as found ILRT leakage rate to meet the 75% L_a criterion after local leakage rate adjustments was due to the inability of reactor building purge penetration V-1 to maintain test pressure between valves CV-7402 and CV-7404 during the Type C test. The as found leakage rate at this penetration was not measurable because the penetration was unable to maintain test pressure and assumed to be greater than L_a. The as left leakage after repairs (replacement of 35 seal adjustment bolts on valves CV-7402 and CV-7404) was not detectable. The local leak rate tests were performed prior to the ILRT. As discussed in the next section of this submittal, the purpose of the ILRT, the testing of the passive containment structure, was successfully fulfilled by the fifth periodic ILRT.

DISCUSSION OF CHANGE

Factors affecting leak tightness of the reactor building may be categorized as: 1) active components which are leak rate tested by Type B and C tests, and 2) passive components which constitute the reactor building structure and are tested during the Type A test.

Active Components

The purpose of containment leak testing is to detect any containment leakage resulting from active or passive failures in the containment isolation boundaries before an accident occurs. The existing Type B and C testing programs are not being modified by this request and will continue to effectively detect reactor building leakage caused by the degradation of active reactor building isolation components (e.g., valves) as well as sealing material within reactor building penetrations. The only potential failures that would not be detected by Type B and C testing are mechanical failure of the reactor building shell, penetration guard pipes, or welds between pipes and the reactor building shell. Only these reactor building structural failures could possibly be affected by the proposed extension in Type A testing frequency.

Industry experience indicates that 97% of the failures associated with Type A tests are found to be due to Type B and C tested penetrations (Draft NUREG 1493, "Performance-Based Containment Leak Test Program"). The local leak rate testing frequencies of these penetrations are not affected by this proposed change. Therefore, continued overall leak tightness of the active reactor building components can be assured by the existing Type B and C testing program. At ANO-1, the only ILRT failure was due to leakage through an active component.

Passive Structure

Two mechanisms could adversely affect the passive structural capability of the reactor building. The first is deterioration of the structure due to pressure, temperature, radiation, chemical or other such effects. Secondly, modifications can be made to the structure which, if not carefully controlled, could leave the structure with reduced capability.

Absent actual accident conditions, structural deterioration is a gradual phenomenon requiring periods of time well in excess of the proposed 18 month interval extension. Other than accident conditions, the only pressure challenge to the reactor building is the ILRT itself.

10CFR50 Appendix J Section V.A. requires a general inspection of the accessible interior and exterior surfaces of the reactor building structures and components to be performed prior to any Type A test to uncover any evidence of structural deterioration which may affect either the reactor building structural integrity or leak tightness. At ANO-1 there has been no evidence of structural deterioration that would impact structural integrity or leak tightness.

Modifications which would alter the passive reactor building structure are infrequent and would receive extensive review to ensure reactor building capabilities are not diminished. The ANO design change and 10CFR50.59 programs have been demonstrated effective in providing a high quality oversight of such safety significant modifications.

Entergy Operations has reviewed modifications installed since the performance of the last ILRT which could affect reactor building integrity, as well as those scheduled for completion

in 1R12. Modifications installed in 1R11 received appropriate Type B or C tests after completion, or did not affect the passive components of the reactor building. Appropriate modifications to be installed in 1R12 will also receive Type B or C tests.

Risk Impact Assessment

The risk impact of reactor building structural leakage is measured by a pathway created for radionuclides in the event that the reactor building is challenged, such as in a loss of coolant accident (LOCA) or severe accident. Such leakage does not create any new accident scenarios, nor does it contribute to the initiation of any accident.

From a risk standpoint, the purpose of Appendix J leak testing is to detect any reactor building leakage resulting from failures in the reactor building isolation boundary before an accident occurs. Such leakage could be the result of leakage through reactor building penetrations, through airlocks, or through reactor building structural faults. The Appendix J Type B and C tests, which are unaffected by this proposed change, will continue to detect leakage through reactor building valves, penetrations, and airlocks. The only potential failures which would not be detected by Type B and C testing are mechanical failures of the containment shell (i.e., degradations or modifications to the containment shell), penetration guard pipes, or welds between pipes and the reactor building shell. Thus, the only potential effect of the proposed change to the Type A test frequency is the probability that reactor building structural leakage would go undetected between tests.

The reactor building structure is passive. Under normal operating conditions, there is no significant environmental or operational stress which could contribute to its degradation. A review of modifications for potential effects to the reactor building structure is discussed in the preceding section. Passive failures resulting in significant reactor building structural leakage are therefore extremely unlikely to develop between Type A tests. No such failures have occurred at ANO-1.

Postulated reactor building failure under severe accident conditions is primarily due to phenomenological effects associated with severe accidents. Such phenomenological effects were considered as part of the ANO-1 IPE. None of the identified reactor building failure mechanisms for severe accidents would be impacted by the proposed increase in the testing interval.

BASIS FOR EXEMPTION

The proposed interval extension meets the criteria for special circumstances as described in 10CFR50.12(a)(2)(ii) and (vi).

50.12(a)(2)(ii) Application of the Regulation is not Necessary to Achieve the Underlying Purpose of the Rule

The underlying purpose of 10CFR50, Appendix J, is still achieved. Appendix J states that the

leakage test requirements set forth in this appendix provide for periodic verification by tests of the leak tight integrity of the primary reactor containment. The appendix further states that the purpose of the tests are to assure that leakage through the primary reactor containment shall not exceed the allowable leakage rate values as specified in the technical specifications or associated bases.

Section III.D.1.(a) states that a set of three periodic tests shall be performed at approximately equal intervals during each 10-year period and that the third test shall be conducted when the plant is shutdown for the 10-year plant inservice inspections. This exemption would provide a one time 18 month extension to the interval between ILRTs, thus causing unequal intervals, while at the same time extending the ILRT 10-year period to match the inservice inspection extended interval. The methodology, acceptance criteria, and technical specification leakage limits for the performance of the Type A test will not change.

The testing history, structural capability of the reactor building, and the risk assessment discussed previously establish that ANO-1 has had acceptable reactor building leakage rates, that the structural integrity of the reactor building is assured and that there is negligible risk impact in extending the Type A test interval.

Thus, there is significant assurance that the extended interval between Type A tests will continue to provide periodic verification of the leak tight integrity of the reactor building.

<u>50.12(a)(2)(vi) Presence of Material Circumstances not Considered when the Regulation was Adopted</u>

There are material circumstances not considered when the regulation was adopted. The benefit of time has provided experience and information which provide a better perspective about containment integrity. Two important material circumstances are testing history and the development of probabilistic risk assessments (PRAs).

Since the promulgation of 10CFR50, Appendix J, in 1973, more than 20 years of nuclear power plant operating history has been obtained. Entergy Operations' personnel have performed a review of industry data and did not find any instances where a Type A test failed to meet Appendix J acceptance criteria as a result of a containment structural leak not due to initial fabrication or a plant modification. This additional operating history provides a significant indicator that containment structural integrity (passive) is not a safety concern.

Plant specific PRAs were not available and therefore were not considered when the regulation requiring compliance with Appendix J (10CFR50.54(o)) was adopted. Overall plant risk due to reactor building leakage is relatively small given the small probability of reactor building leakage itself. The predominant contributor to degraded reactor building integrity is the phenomenological effects of a severe accident, and not preexisting reactor building isolation conditions.

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATION

An evaluation of the proposed change has been performed in accordance with 10CFR50.91(a)(1) regarding no significant hazards considerations using the standards in 10CFR50.92(c). A discussion of these standards as they relate to this amendment request follows:

Criterion 1 - Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.

The proposed change revises Technical Specification 4.4.1.1.4 to reference the testing frequency requirements of 10CFR50, Appendix J, and to state that NRC approved exemptions to the applicable regulatory requirements are permitted. The current requirements of TS 4.4.1.1.4 paraphrase the requirements of Section III.D.1.(a) of Appendix J. The proposed administrative revision simply deletes the paraphrased language and directly references Appendix J. No new requirements are added, nor are any existing requirements deleted. Because the current TS language paraphrases Appendix J, any approved exemption to Section III.D.1.(a) of Appendix J would also affect the requirements of TS 4.4.1.1.4, unless the proposed clarification phrase permitting the use of approved exemptions is added. Any specific changes to the requirements of Section III.D.1.(a) will require a submittal from Entergy Operations under 10CFR50.12 and subsequent review and approval by the NRC prior to implementation. The proposed change is stated generically to avoid the need for further TS changes if different exemptions are approved in the future.

The proposed change, in itself, does not affect reactor operations or accident analysis and has no radiological consequences. The change provides clarification so that TS changes will not be necessary in the future to correspond to applicable NRC approved exemptions from the requirements of Appendix J. Therefore, this change does <u>not</u> involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2 - Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed change provides clarification to a specification which paraphrases a codified requirement. Since the proposed amendment would not change the design, configuration or method of operation of the plant, it would <u>not</u> create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3 - Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed change is administrative and clarifies the relationship between the requirements of TS 4.4.1.1.4, Appendix J, and any approved exemptions to Appendix J. It does not, in itself, change a safety limit, an LCO, or a surveillance requirement on equipment required to operate the plant. The NRC will directly approve change proposed exemption to III.D.1.(a) of Appendix J prior to implementation. Therefore, this change does <u>not</u> involve a significant

reduction in the margin of safety.

The Commission has provided Guidance in 51 FR 7750 dated March 6, 1986, concerning the application of these 10CFR50.92 standards by providing examples of amendments which are likely to involve no significant hazards considerations. The proposed amendment most closely matches example (i): "A purely administrative change to the technical specifications; for example, a change to achieve consistency throughout the technical specifications, correction of an error, or a change in nomenclature."

Therefore, based upon the reasoning presented above and the previous discussion of the amendment request, Entergy Operations has determined that the requested change does not involve a significant hazards consideration.