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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 203 TO

FACILITY OPERATING LICENSE NO. DPR-51

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated December 16, 1999 (1CAN129904), Entergy Operations, Inc. (Entergy or the licensee), submitted a request for changes to the Arkansas Nuclear One, Unit No. 1 (ANO-1), Technical Specifications (TSs). The requested changes would revise TS 4.18.5.b to allow tube 110/60, located in the "A" steam generator, to remain in service through the current operating fuel cycle (cycle 16) with two axial indications that have potential depths (as a percentage of nominal wall thickness) greater than the plugging limit. The axial indications are located in the roll transition region and are contained within the upper tubesheet.

2.0 BACKGROUND

On September 10, 1999, Entergy initiated a plant shutdown of ANO-1 to begin the refueling outage following the completion of operating cycle 15 (1R15). During the 1R15 outage, Entergy conducted an inservice inspection of the once-through steam generators (OTSGs) in accordance with TS 4.18, "Steam Generator Tubing Surveillance." The inservice inspection included a non-destructive examination by eddy-current testing (ECT) of the OTSG tubing. During a review of ECT data collected during the last refueling outage for ANO-1, it was determined that a tube containing two parallel axial indications that exceeded the plugging limit was not repaired as required. During the in-process evaluation of the ECT data conducted in 1R15, two small parallel axial indications in tube 110/60 were identified. One indication was sized as 0.04 inch in length with a 59 percent average and 97 percent maximum through-wall depth, and the other was sized as 0.05 inch in length with a 74 percent average and 97 percent maximum through-wall depth.

During a review of the resolution analysis compare sheets for this tube, Entergy noted that the indications were identified in two different locations; one was called at the tube end and the other at the upper roll transition (URT). The indication at the URT was given an appropriate coding of "repairable" by the primary and secondary production analysts but at an incorrect location of upper tube end (UTE)-3. The primary and secondary resolution analysts agreed on the repairable call but kept it at the incorrect location. During the licensee's independent oversight process, an analyst reviewed the indication and corrected the flaw location to UTE-1. However, the independent analyst made an error by changing the call from repairable to "non-repairable." This error was confirmed by the licensee during a reevaluation following ANO-1's return to power operations that found the correct classification was repairable as originally determined.

Condition Report (CR) No. ANO-1-1999-0577 was written at 9:43 a.m. eastern standard time (EST) on December 15, 1999, to document this concern. This CR was brought to the attention of the ANO-1 Control Room in order to assess the impact of this issue on the operability of the "A" steam generator. Failure to repair a tube with a flaw that exceeds the plugging limit constitutes a failure to comply with the surveillance requirements of the ANO-1 TSs. TS 4.18.5.b states that, "The steam generator shall be determined operable after completing the corresponding actions (plug, reroll, or sleeve all tubes exceeding the plugging limit and all tubes containing non-TEC [tube end cracks] through-wall cracks) required by Table 4.18-2." The plugging limit is defined as the imperfection depth at or beyond 40 percent of the nominal tube wall thickness for which the tube shall be sleeved, rerolled, or removed from service because it may become unserviceable prior to the next inspection. There is no action associated with the failure to comply with surveillance requirement 4.18.5.b, however, the surveillance specification implies that operability of the "A" steam generator cannot be demonstrated without the performance of a valid surveillance. As a result, the "A" steam generator was declared inoperable at 9:43 a.m. EST and TS 3.1.1.5, "Reactor Coolant System - Reactor Coolant Loops," was entered at that time. TS 3.1.1.5 states that, "With the reactor coolant average temperature above 280 °F, the reactor coolant loops listed below shall be operable: 1) Reactor Coolant Loop (A) and at least one associated reactor coolant pump, and 2) Reactor Coolant Loop (B) and at least one associated reactor coolant pump...." The action associated with this TS requires the restoration of the required loops to operable status within 72 hours or reduce the reactor coolant average temperature to less than or equal to 280 °F within the next 12 hours.

Based on discussions with the Nuclear Regulatory Commission (NRC) staff and a further review by your plant technical staff, Entergy indicated that TS 3.1.1.2 should have been entered as the appropriate specification covering this condition. TS 3.1.1.2 states that, "Two steam generators shall be operable whenever the reactor coolant average temperature is above 280 °F." There is no specified action associated with this limiting condition for operation. Therefore, TS 3.0.3, "Limiting Condition for Operation (General)," was entered, as appropriate, to address this condition. TS 3.0.3 states that, "When a Limiting Condition for Operation is not met, except as provided in the associated Action requirements, within one hour action shall be initiated to place the unit in an OPERATING CONDITION in which the Specification does not apply by placing it, as applicable, in: 1. At least HOT STANDBY within the next 6 hours, 2. At least HOT SHUTDOWN within the following 6 hours, and 3. At least COLD SHUTDOWN within the subsequent 24 hours...." TS 3.1.1.2 and TS 3.0.3 were entered at 3:46 p.m. EST.

By letter dated December 16, 1999 (1CAN129905), the licensee requested that the NRC exercise discretion not to enforce compliance with TS 3.1.1.2, "Reactor Coolant System - Steam Generator," and 3.0.3, "Limiting Condition For Operation (General)," for ANO-1. This letter documented information previously discussed with the NRC in a telephone conference on December 15, 1999, from 6:30 p.m. to 7:45 p.m. EST.

The licensee requested that the NRC issue a Notice of Enforcement Discretion (NOED) pursuant to the NRC's policy regarding exercise of discretion for an operating facility, set out in Section VII.c of the "General Statement of Policy and Procedures for NRC Enforcement Actions" (Enforcement Policy), NUREG-1600, and be effective immediately and remain in effect until such time that the NRC staff acts on the licensee's proposed exigent TS change request to be submitted within 48 hours of authorization of the NOED. By letter dated December 17, 1999, the staff documented the issuance of the NOED for ANO-1. The NOED had been issued verbally on December 15, 1999, at 7:35 p.m. EST after the staff concluded that the licensee's technical basis for the request was satisfactory. By letter dated December 16, 1999 (1CAN129904), the licensee submitted a request for an exigent amendment to revise TS 4.18.5.b to allow continued operation with tube 110/60 in service. The licensee requested that this exception be authorized for the remainder of the current operating cycle. "A" Steam generator tube 110/60 will be repaired or removed from service during the next refueling outage.

3.0 EVALUATION

The licensee determined that the degradation mechanism associated with the axial indications in the URT was primary water stress corrosion cracking (PWSCC) in the URT. Entergy stated that the upper roll areas of the OTSGs have been inspected during refueling outages 1R13, 1R14, and 1R15, and indications have been detected by ECT in roll transitions during each of these outages using a 0.115 inch diameter rotating pancake coil and/or a Plus-Point coil. The licensee stated that one tube was pulled and evaluated during the 1R13 outage that confirmed the degradation was from PWSCC with an axial orientation. The cracking was believed to be caused by residual hoop stresses. Entergy also stated that it considered the axial size determination from the ECT to bound the actual length. This was based on a 1999 Babcock and Wilcox Owners Group project that compared the results using a Plus-Point probe to that from destructive examinations. In most cases, the Plus-Point probe oversized the actual flaw length.

In the December 16, 1999 (1CAN129904), letter, Entergy stated that there has been no evidence of leakage through a URT flaw in the operating history of ANO-1. Further, a leak (bubble) test was performed at the end of cycle 14 with no leakage identified from this degradation mechanism. The current reactor coolant system (RCS) primary-to-secondary operational leakage is at about the minimum detectable level. Based on the size of the indications measured by ECT, Entergy estimated that the accident-induced leakage that could result from each flaw in tube 110/60, assuming through-wall cracking over the entire flaw length, would be negligible. After allowing for crack growth of 0.042 inch during the remainder of the cycle, Entergy also calculated the leak rate using the Opcon leaker module. The crack growth rate was based on bounding data from an Electric Power Research Institute (EPRI) study for PWSCC in tube hardrolls in foreign and domestic steam generators, including recirculating steam generators. Including growth, the estimated leak from a 0.1 inch flaw at the end of cycle (EOC) and 100 percent through-wall over the entire length would be 0.0026 gallon per minute (gpm). The combined leakage from both flaws was estimated at 0.0052 gpm, which when combined with expected leakage associated with other flaws in the OTSGs at EOC, yields a total leak rate estimate that is less than 0.9 gpm under accident conditions. The licensee identified the main steam line break scenario as the limiting accident analysis for this condition. The total estimated leakage from the OTSGs is within the current inputs and assumptions of the existing analysis. Entergy stated that the Opcon Code had been benchmarked against a Framatome Code for estimating leakage from through-wall tube cracks. The NRC staff believes that there are typically large uncertainties associated with these leakage prediction codes. However, since these indications are contained within the tubesheet region, there would be some constraint against leakage provided by the tubesheet if leakage were to occur. In addition, the predicted accident leakage of 0.0052 gpm is significantly smaller than the 1 gpm limit assumed in the accident analyses. Therefore, the NRC staff finds the leak rate assessment to be reasonable.

The licensee indicated that ANO-1 has a robust monitoring program to detect and mitigate the effects of primary to secondary steam generator tube leakage. In addition, procedures are in place to direct the response of plant operators in the event that steam generator tube leakage is detected. Finally, the ANO-1 plant operations staff has been trained on these procedures. The methodology for monitoring the secondary system for leakage includes the use of process monitors to check radiation levels in the condenser off-gas, nitrogen-16 (N-16) gamma levels from the OTSGs, chemistry samples, and RCS mass balances to detect and calculate leakage. Additionally, ANO-1 has a procedural limit of 0.069 gpm (100 gallons per day (gpd)) that is more restrictive than the 0.104 gpm (150 gpd) limit allowed by TS 3.1.6.3.b. The N-16 gamma detectors provide continuous monitoring and alarm indication in the Control Room. This system has a dual alarm setpoint system. One alarm is set at slightly above the baseline reading to detect small changes in activity. This alarm can be adjusted by plant operators. The second alarm has a fixed setpoint which when reached requires mitigating actions on the part the operations staff. The NRC staff has reviewed the licensee's program for the detection and mitigation of steam generator tube leakage and concluded that the appropriate actions, as described above, would be taken by the licensee if leakage were to occur. Therefore, the NRC staff finds that continued operation for the remainder of the current operating cycle (cycle 16) with tube 110/60 left in service is acceptable.

On the basis of the staff's evaluation of the December 16, 1999 (1CAN129904), request from Entergy, we have concluded that the proposed change has no adverse impact on public health and safety. Entergy's assessment of primary-to-secondary leakage during normal operations and accident conditions is believed to be conservative and will remain within the leakage requirements for steam generator integrity assumed in the main steam line break accident analysis. Therefore, the staff finds the proposed changes to the TSs to be acceptable.

4.0 EXIGENT CIRCUMSTANCE

The Commission's regulations, as stated in 10 CFR 50.91, contain exceptions for issuance of amendments when the usual 30-day public comment period cannot be met. One type of special exception is an exigency. Pursuant to 10 CFR 50.91(a)(b), exigent circumstances exist if: (a) the staff and licensee need to act promptly and time does not permit the staff to publish a Federal Register notice allowing 30 days for prior public comment; and (b) the staff determines that the amendment involves no significant hazards considerations.

Under such circumstances, the Commission notifies the public in one of two ways: by issuing a Federal Register notice providing an opportunity for hearing and allowing at least two weeks for public comments, or by issuing a press release discussing the proposed changes, using the local media. In this case, the Commission used the first approach.

The licensee submitted the request for amendment on December 16, 1999. It was noticed in the Federal Register on December 29, 1999 (64 FR 73080), at which time the staff provided an opportunity for hearing and proposed no significant hazards consideration determination. The public was allowed 14 days after the date of publication of that notice to provide comments. No comments were received.

In its application, the licensee requested that the amendment be processed as an exigent request pursuant to 10 CFR 50.91(a)(6). The exigency was created when the licensee declared the "A" steam generator inoperable when it was discovered that a tube containing two axial indications that exceeded the tube plugging limit was left in service following the start-up from the last refueling outage. Therefore, TS 3.1.1.2, "Reactor Coolant System - Steam Generator," was entered. This TS has no associated required action for an inoperable steam generator. Therefore, TS 3.0.3 was entered, as appropriate, to address this condition. TS 3.0.3 requires, within one hour, that action be taken to place the unit in an operating condition in which the TS does not apply through the initiation of a plant shutdown.

In order to allow continued operation of the plant in noncompliance with TS 3.0.3 and 3.1.1.2, enforcement discretion was verbally requested by the licensee and granted by the NRC on December 15, 1999. The NOED, which was documented in a letter dated December 17, 1999, granted enforcement discretion from the requirements of TS 3.0.3 and 3.1.1.2 until an exigent TS amendment is processed to revise TS 4.18.5.b.

Accordingly, pursuant to 10 CFR 50.91(a)(6), the Commission has determined that the licensee used its best efforts to make a timely application and that exigent circumstances exist in that the licensee and the Commission must act quickly to prevent unnecessary interruption of plant operations. Further, the Commission has determined that the exigency could not have been avoided and that the licensee did not create the exigency to take advantage of this procedure.

5.0 FINAL NO SIGNIFICANT HAZARDS CONSIDERATION DETERMINATION

The Commission's regulations in 10 CFR 50.92 state that the Commission may make a final determination that a license amendment involves no significant hazards considerations if operation of the facility in accordance with the amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety.

As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

> 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 OTSGs are used to remove heat from the reactor coolant system (RCS) during normal operation and during accident conditions. The OTSG tubing forms a substantial portion of the reactor coolant pressure boundary. An OTSG tube failure is a violation of the reactor coolant pressure boundary and is a specific accident analyzed in the ANO-1 Safety Analysis Report (SAR).

The purpose of the periodic surveillance performed on the OTSGs in accordance with ANO-1 Technical Specification 4.18 is to ensure that the structural integrity of this portion of the RCS will be maintained. The technical specification plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the technical specifications as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a steam line break. Of these accidents, the most severe condition with respect to axial cracking in the upper roll transition (URT) of a tube within the tubesheet is a main steam line break (MSLB). During this event the differential pressure across the tube could be as high as 2500 psid [pounds per square inch differential]. The rupture of a tube during this event could permit the flow of reactor coolant into the secondary system thus bypassing the containment.

From testing performed on simulated flaws within the tubesheet it has been shown that the axial indications within the upper tube sheet left in service during cycle 16 do not represent structurally significant flaws which would increase probability of a tube failure beyond that currently assumed in the ANO-1 SAR.

Burst tests were conducted on tubing with simulated flaws within the tubesheet. In these tests, through-wall holes of varying sizes up to 0.5 inch in diameter were drilled in test specimens. The flawed specimen tubes were then inserted into a simulated tubesheet and pressurized. In all cases the tube burst away from the flaw in that portion of the tube that was outside the tubesheet. The size of these simulated flaws bound the indications left in service within the upper tubesheet during 1R15. These tests demonstrate, for flaws similar to the axial indications in the ANO-1 upper tubesheet, that the tubes will not fail at this location under accident conditions.

The dose consequences of a MSLB accident are analyzed in the ANO-1 accident analysis. This analysis assumes a 1 gpm OTSG tube leak and that the unit has been operating with 1% defective fuel. The postulated accident induced leak rate contribution at the end of cycle from these indications is negligible.

Therefore, this change does not 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 OTSGs are passive components. The intent of the technical specification surveillance requirements is being met by this change in that adequate structural and leakage integrity will be maintained. The proposed change introduces no new modes of plant operation.

Therefore, this change does not 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 ANO-1 Technical Specification Bases specify that the surveillance requirements (which includes the plugging limit) are to ensure the structural integrity of this portion of the RCS pressure boundary. The technical specification plugging limit of 40% of the nominal tube wall thickness requires tubes to be repaired or removed from service because the tube may become unserviceable prior to the next inspection. Unserviceable is defined in the technical specifications as the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an operating basis earthquake, a loss-of-coolant accident, or a MSLB. Of these accidents, the most severe condition with respect to flaws within the tubesheet is the MSLB.

Testing of simulated through wall flaws of up to 0.5 inch in diameter within a tubesheet showed that the tubes always failed outside of the tubesheet. Thus the structural requirement of the bases of the surveillance specification is satisfied.

Leakage under accident conditions would be limited due to the small size of the flaws and would be low enough to ensure offsite dose limits are not exceeded.

Therefore, this change does not involve a significant reduction in the margin of safety.

Based on the above considerations, the staff concludes that the amendment meets the three criteria of 10 CFR 50.92(c). Therefore, the staff has made a final determination that the proposed amendment does not involve a significant hazards consideration.

6.0 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.

7.0 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 determined that the amendment involves no significant hazards consideration, and there was no public comment on the previously-issued proposed finding on this matter (64 FR 73080, December 29, 1999). Accordingly, the amendment meets 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 amendment.

8.0 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.

Principal Contributor: C. Nolan

Date: January 13, 2000

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

January 13, 2000

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT NO. 1 - ISSUANCE OF AMENDMENT RE: REVISION TO STEAM GENERATOR SURVEILLANCE REQUIREMENTS (TAC NO. MA7370)

Dear Mr. Hutchinson:

The Commission has issued the enclosed Amendment No. 203 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 December 16, 1999 (1CAN129904).

The amendment revises TS 4.18.5.b to allow tube 110/60 to remain in service through the current operating fuel cycle (cycle 16) with two axial indications that have potential through-wall depths greater than the plugging limit.

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,

/RA/

M. Christopher Nolan, 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. 203 to DPR-51
2. Safety Evaluation
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