April 24, 2000

Mr. Craig G. Anderson
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - RELIEF AUTHORIZATION RE: ALTERNATIVE REPAIR TO A REACTOR COOLANT SYSTEM INSTRUMENT NOZZLE (TAC NO. MA8270)

Dear Mr. Anderson:

In your application dated February 24, 2000 (1CAN020006), as supplemented by letters dated February 25, 2000 (1CAN020007), February 28, 2000 (1CAN020008), February 29, 2000 (1CAN020009), and March 1, 2000 (1CAN030003), Entergy Operations, Inc. (Entergy or the licensee), submitted a request seeking approval for an alternate repair to a 1-inch instrument nozzle located on the reactor coolant system (RCS) hot leg for Arkansas Nuclear One, Unit 1 (ANO-1).

On February 5, 2000, ANO-1 entered a maintenance outage to repair the anti-rotation device for the "D" reactor coolant pump. During this maintenance outage, utility personnel identified a boric acid build-up on a RCS hot leg instrument nozzle. This nozzle is one of seven nozzles installed in 1986 as part of the hot leg level monitoring system. All seven nozzles utilized a similar design. Inspections of all seven nozzle locations revealed that six nozzles experienced through-wall leakage and the seventh nozzle contained flaw indications that had not yet progressed to a through-wall leak. You determined that the primary degradation mechanism affecting the nozzles was primary water stress corrosion cracking due to the incorporation of a susceptible material (i.e., Alloy 600) into the nozzle design. An eighth nozzle used in the hot leg level monitoring system was an existing design that utilized a different configuration and material of construction. You confirmed that this nozzle was not affected by the degradation mechanisms exhibited in the other nozzles.

You decided to repair all seven nozzles. You performed repairs to six of the nozzles in accordance with the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." However, the seventh nozzle was located below the elevation of the RCS mid loop water level. This would necessitate a full core off load in order to lower the water level below the elevation of the hot leg nozzle to permit a Code repair. Consequently, the licensee has requested approval from the Nuclear Regulatory Commission (NRC or the Commission) for an alternative, non-Code repair to this nozzle under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i). This repair would remain in service for the remainder of the current operating cycle. A Code repair would be made during refueling outage 1R16, which is currently scheduled for spring 2001.

The staff has reviewed and evaluated the information in your request and finds that your proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i). The staff's authorization of this alternative was based, in part, on your commitment to limit the use of this non-Code repair to the remainder of this operating cycle, after which the nozzle would be replaced with a configuration that complies with the requirements of the ASME Code. The staff's conclusions are detailed in the enclosed Safety Evaluation.

Evidence of through-wall leakage from the RCS hot leg instrument nozzles was identified during a maintenance outage. Code repairs to six of the nozzles and the implementation and authorization of the non-Code repair for the seventh nozzle was required prior to the restart of ANO-1. Therefore, the Commission chose to authorize this alternative in verbal form due to the undue regulatory burden that would have resulted from the delay inherent in a written authorization. This verbal authorization was based on the staff's review of the licensee's application dated February 24, 2000, as supplemented by letter dated February 25, 2000. The verbal authorization was made at approximately 2:05 p.m. eastern standard time (EST) on February 25, 2000. The principal NRC staff members who participated in the telephone conversation were Mr. Robert A. Gramm, Chief, Section 1, Project Directorate IV & Decommissioning; Mr. M. Christopher Nolan, Project Manager, Section 1, Project Directorate IV & Decommissioning; and Mr. Barry J. Elliot, Acting Chief, Component Integrity Section, Material and Chemical Engineering Branch.

During the implementation of the non-Code repair, you identified a non-conservative assumption in your evaluation of the existing flaw. The original evaluation assumed that the existing flaw was confined to the J-groove nozzle weld. However, during the implementation of the non-Code repair, the licensee could not verify that the associated fillet weld was free of existing flaws. Therefore, the licensee performed a new evaluation assuming that the flaw existed in both the J-groove and the fillet portions of the weld. The licensee updated their application with supplemental letters dated February 28, 2000, and February 29, 2000. The Commission chose to re-authorize this alternative in verbal form due to the undue regulatory burden that would have resulted from the delay in a written authorization. This verbal authorization was based on the staff's review of the licensee's application dated February 24, 2000, as supplemented by letters dated February 25, 2000, February 28, 2000, and February 29, 2000. The verbal authorization was made at approximately 9:35 p.m. EST on February 29, 2000. The principal NRC staff members who participated in the telephone conversation were Mr. Robert A. Gramm, Chief, Section 1, Project Directorate IV & Decommissioning; Mr. M. Christopher Nolan, Project Manager, Section 1, Project Directorate IV & Decommissioning; Mr. Barry J. Elliot, Acting Chief, Component Integrity Section, Material and Chemical Engineering Branch; and Mr. Chi-Fu Sheng, Senior Technical Reviewer, Component Integrity Section, Material and Chemical Engineering Branch. This approval was conditional on the licensee's commitment to demonstrate addition conservatism to their evaluation in the form of a 1.15 reduction in the allowable stress in the weld material in accordance with ASME Code, Section XI, Paragraph IWB-3641. The licensee documented the effect of this additional conservatism in a supplemental letter dated March 1, 2000.

Sincerely,

/RA by John Nakoski Acting for/

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

Docket No. 50-313

cc: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE REPAIR

FOR A REACTOR COOLANT SYSTEM HOT LEG INSTRUMENT NOZZLE

ARKANSAS NUCLEAR ONE, UNIT NO. 1

DOCKET NO. 50-313

1.0 INTRODUCTION

By letter dated February 24, 2000 (1CAN020006), as supplemented by letters dated February 25, 2000 (1CAN020007), February 28, 2000 (1CAN020008), February 29, 2000 (1CAN020009), and March 1, 2000 (1CAN030003), Entergy Operations, Inc. (Entergy or the licensee), submitted a request seeking relief from the requirements of the Amercian Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to allow an alternative, non-Code repair to a 1-inch instrument nozzle located on the reactor coolant system (RCS) hot leg for Arkansas Nuclear One, Unit 1 (ANO-1).

On February 5, 2000, ANO-1 entered a maintenance outage to repair the anti-rotation device for the "D" reactor coolant pump. During this maintenance outage, utility personnel identified a boric acid build-up on a RCS hot leg instrument nozzle indicating that through-wall leakage of RCS fluid had previously occurred at this location. This nozzle is one of seven nozzles installed in 1986 as part of the hot leg level monitoring system. All seven nozzles utilized a similar design. Inspections of all seven nozzle locations revealed that six nozzles had experienced previous through-wall leakage and the seventh nozzle contained flaw indications that had not yet progressed to a through-wall leak. The licensee determined that the primary degradation mechanism affecting the nozzles was primary water stress corrosion cracking (PWSCC) due to the incorporation of a susceptible material into the nozzle design. These seven nozzles, which were made of Inconel-600, were welded onto the RCS hot legs in 1986 using Inco-182 weld material to provide hot leg level indication. The licensee was not able to eliminate other potential degradation mechanisms as potential contributors to the failure of the original design until a more detailed root cause investigation was completed. An eighth nozzle used in the hot leg level monitoring system was an existing design that utilized a different configuration and material for construction. The licensee confirmed that this nozzle was not affected by the degradation mechanisms exhibited in the other nozzles.

The licensee decided to repair all seven nozzles. The licensee performed repairs to six of the nozzles in accordance with ASME Code, Section XI. However, the seventh nozzle was located below the elevation of the RCS mid loop water level. This would necessitate a full core off load in order to lower the water level below the elevation of the hot leg nozzle to permit a Code repair. Consequently, the licensee has requested approval from the Nuclear Regulatory Commission (NRC or the Commission) for an alternative, non-Code repair to this nozzle under the provisions of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(a)(3)(i).

This repair would remain in service for the remainder of the current operating cycle. A Code repair would be made during refueling outage 1R16, which is currently scheduled for spring 2001.

2.0 EVALUATION

2.1 Applicable Requirements

Title 10 CFR 50.55a(g), requires nuclear power facility piping and components to meet the applicable requirements of Section XI of the Code. However, 10 CFR 50.55a(a)(3)(i) permits alternatives to the ASME Code requirements, when "[t]he proposed alternatives would provide an acceptable level of quality and safety...." For non-Code repaired piping and components, where flaw indications have not been eliminated completely, an analytical evaluation using criteria and procedures described in ASME Code, Section XI, Paragraph IWB-3600 is required to demonstrate acceptable performance.

2.2 Licensee's Evaluation

2.2.1 Description of the Non-Code Repair

The proposed repair would embed the existing flaws in the nozzle weld through the application of a pad applied to the outside surface of the hot leg surrounding the nozzle. In addition, a fillet weld would be applied to transition from the weld pad to the nozzle. Both the pad and the fillet would be applied with Inco-152 weld metal using a shielded metal arc welding (SMAW) process.

2.2.2 Root Cause

The licensee has assessed potential mechanisms that might cause the flaw to initiate and grow in the original design for the hot leg nozzle weld, and attributed the failure to either fatigue or environmentally assisted PWSCC. Qualitative analyses have been performed to rule out three sources of fatigue contributors for the non-Code repair: (1) low cycle fatigue from heat ups and cool downs, (2) thermal fatigue from system transients, and (3) flow induced thermal fatigue due to in leakage. A fourth source, high cycle fatigue from reactor coolant pump (RCP) vibrations, has been identified as having a potential impact on the crack initiation, and, therefore, a thorough analysis has been performed to assess the degradation due to this mechanism quantitatively.

The licensee performed a vibration analysis based on measured data on the cold leg. Because of its proximity to the RCP, the licensee believes the cold leg data bounds the vibration on hot legs. The licensee then conservatively used a 1 mil peak to peak amplitude, which was 25% more than the measured amplitude on cold legs, to calculate the vibratory stresses due to the RCP rotor excitation (20 Hz) and the RCP vane passing excitation (100 Hz). The resulting vibratory stresses were calculated to be 5 ksi, compared to an endurance limit of 16.5 ksi. Based on this, the licensee discounted this mechanism as the cause for the nozzle weld failure.

2.2.3 Primary Stress Evaluation of the Non-Code Repair

To demonstrate that the non-Code repair satisfies the primary stress criteria of ASME Code, Section III, Subsection NB, "Rules for Construction of Nuclear Facility Components," Article NB-3200, the licensee performed a finite element method (FEM) analysis using the bounding design load, which includes pressure, deadweight, temperature, and operating basis earthquake (OBE) load. For conservatism, the model did not include the original J-weld. The resulting membrane and bending stress intensities from the FEM analysis indicated that they satisfy the ASME Code allowable.

2.2.4 Flaw Evaluation of the Non-Code Repair

Since the weld overlay repair did not remove completely the original weld containing the through wall flaw, a flaw evaluation in accordance with Section XI is needed on the non-Code repair. To demonstrate that the non-Code repair satisfies Section XI requirements on flawed components, the licensee performed a limit-load analysis using procedures and acceptance criteria similar to that of ASME Code, Section XI, Appendix C. The licensee applied the FEM results of Section 2.2.2 and employed a safety factor of 3 and a Z factor of 1.15 to calculate the applied stress for the repaired weld fabricated by the SMAW process. The allowable stress (flow stress) was based on information from Code Case N-525. Since the computed membrane stress of 58.3 ksi (including the safety factor and the Z factor) is less than the flow stress of 60 ksi, the licensee concluded that the non-Code repair satisfies Section XI requirements and has adequate structural integrity.

2.2.5 Crack Growth for the Non-Code Repair

The licensee estimated the crack growth for a postulated flaw in the non-Code repair to be 0.042 inch in one fuel cycle based on the following assumptions: (1) a crack length of 1.375 inches, (2) a center cracked panel fracture mechanics model, (3) 10 heatup and 10 cooldown cycles, (4) two times the Section XI reference fatigue crack growth rate for austenitic materials in water environment, and (5) a residual stress of 27.7 ksi. To account for this growth, the licensee reserved extra weld thickness (0.06 inch) in its weld overlay design. Further, the licensee demonstrated that the non-Code repair mitigates all possible degradation mechanisms experienced in the nozzles of the original configuration. The non-Code repair used the more environment resistant Inco 52/152 instead of the Inco-182 used in the original design to mitigate PWSCC. In addition, the non-Code repair used the weld overlay to build up a stiffer nozzle base to mitigate the effect due to RCP pump vibrations by de-tuning and lowering the vibratory stresses at the nozzle base.

2.3 Evaluation

The licensee performed qualitative evaluation on possible degradation mechanisms and performed quantitative evaluation on the most probable mechanism - fatigue due to RCP vibrations. Although the licensee discounts RCP vibrations as the root cause of the nozzle weld cracking, the staff believes that the root cause is likely to be a combination of PWSCC and the RCP vibrations. How these two mechanisms interacted to cause cracking is important, but not essential in this evaluation, because the licensee demonstrated that the current non-Code repair tends to mitigate both mechanisms. Even if the low cycle fatigue due to heat ups and cool downs played a part as an additional mechanism to the failure, it will not affect the following staff evaluation because of the limitations placed on the non-Code repair.

The licensee demonstrated that the non-Code repair satisfied the primary stress criteria of NB-3200 of Section III of the ASME Code. This is required if the original weld containing the through wall flaw had been removed completely before the weld overlay repair was performed. However, since the non-Code repair did not remove the original weld containing the through wall flaw completely, a flaw evaluation in accordance with Section XI needs to be performed instead.

The licensee conducted a limit-load analysis using procedures and acceptance criteria similar to that of ASME Code, Section XI, Appendix C. The staff determined that the licensee's FEM model exclusion of the original J-weld is conservative, and the application of a safety factor of 3 and a Z factor of 1.15 to the applied stress for the repaired weld fabricated by the SMAW process is in accordance with Appendix C. The licensee's allowable stress (flow stress) was based on information from Code Case N-525 for Alloy 690 which is representative of Inco 52/152. The allowable stress for the non-Code repair weld material is not available in the literature, therefore, there is uncertainty in the allowable stress used in the limit load analysis. However, judging from the fact that the FEM model did not include the J-weld, the staff determined that licensee's applied stress at the critical location has been overestimated considerably. If the "cavity" were filled with the original weld in the FEM model, the additional load path (about twice the area of the critical location) would reduce the applied stress by at least 30% based on the area of the critical location in the FEM model and the area of the J-weld not being considered in the FEM model. Since the computed membrane stress of 58.3 ksi is less than the flow stress of 60 ksi, the staff agrees with the licensee's conclusion that the non-Code repair satisfies the Section XI requirements.

To completely satisfy the Section XI requirements, the crack growth during the proposed period of operation has to be estimated. The licensee's estimate of the crack growth in the non-Code repair is 0.042 inch in one fuel cycle based on conservative assumptions: the crack length of 1.375 inches is based on the largest dimension of the weld geometry, a three-dimensional crack is modeled as a two-dimensional center-cracked panel, and 10 heatup and 10 cooldown cycles were assumed as a conservative estimate for the remainder of the fuel cycle. Further, using two times the Section XI reference fatigue crack growth rate for austenitic materials in water environment, and using a residual stress of 27.7 ksi are appropriate. Since this calculated growth is within the extra 0.06 inches considered in the weld overlay design, which was not included in the FEM model, the above Section XI analysis is still acceptable.

3.0 CONCLUSION

The staff has completed its review of the licensee's submital which described the alternate repair of the hot leg level nozzle. The staff has determined that the alternate repair described by the licensee satisfies both ASME Code, Section III requirements on stress intensity, and ASME Code, Section XI requirements on flawed components. The staff concludes that the proposed alternative provides an acceptable level of quality and safety and, therefore, the licensee's proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i). This alternative is authorized for the remainder of the current operating cycle. A Code repair to this nozzle will be required to allow restart following refueling outage 1R16, which is currently scheduled for spring 2001.

Principal Contributor: S. Sheng
C. Nolan

Date: April 24, 2000

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