February 1, 2007

Mr. Jeffrey S. Forbes
Site Vice President
Arkansas Nuclear One
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72802

SUBJECT: ARKANSAS NUCLEAR ONE, UNIT 1 - REQUEST FOR ALTERNATIVE NO. ANO1-ISI-006 TO EXTEND THE THIRD INSERVICE INSPECTION INTERVAL FOR REACTOR VESSEL EXAMINATION CATEGORY B-F WELD EXAMINATIONS (TAC NO. MD1397)

Dear Mr. Forbes:

By letter dated April 24, 2006, Entergy Operations, Inc. (Entergy), the licensee for Arkansas Nuclear One, Unit 1 (ANO-1), submitted a request for authorization to extend the third 10-year inservice inspection (ISI) interval for the reactor core flood line nozzle-to-safe end weld examinations. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(a)(3)(ii), the licensee requests approval for the use of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Paragraph IWB-2412, for the ANO-1 nuclear power plant. In response to the Nuclear Regulatory Commission's (NRC) Request for Additional Information, Entergy resubmitted a revised request for authorization in a letter dated September 8, 2006.

Based on the NRC staff's review of the information provided by the licensee in its letters dated April 24 and September 8, 2006, authorizing the proposed alternative provides reasonabel assurance of structural integrity and is justified on the basis that compliance with the specified requirements would result in hardship without a compensating increase in the level of quality and safety. Therefore, the staff authorizes the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval at ANO-1. The proposed alternative is authorized until the end of the ANO-1 fall 2008 refueling outage.

The NRC staff's safety evaluation is enclosed.

Sincerely,

***/RA/***

David Terao, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure: Safety Evaluation

cc w/encl: See next page

The NRC staff's safety evaluation is enclosed.

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Plant Licensing Branch IV
Division of Operating Reactor Licensing
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR ALTERNATIVE TO ASME SECTION XI

PROPOSED ALTERATIVE TO EXTEND THE THIRD 10 YEAR INSERVICE

INSPECTION INTERVAL FOR REACTOR VESSEL

EXAMINATION CATEGORY B-F WELD EXAMINATIONS

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NUMBER 50-313

## 1.0 INTRODUCTION

By letter dated April 24, 2006, Entergy Operations, Inc. (Entergy), the licensee for Arkansas Nuclear One, Unit 1 (ANO-1), submitted a request for authorization to extend the third 10-year inservice inspection (ISI) interval for the reactor core flood line nozzle-to-safe end weld examinations. Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(a)(3)(ii), the licensee requests approval for the use of an alternative to the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, Paragraph IWB-2412, for the ANO-1 nuclear power plant. In response to the Nuclear Regulatory Commission's (NRC) Request for Additional Information, Entergy resubmitted a revised request for authorization in a letter dated September 8, 2006.

ANO-1 is currently in its third ISI interval, which began June 1, 1997, and ends May 31, 2007. ASME Code, Section XI, IWA-2430(d) allows for a 1-year extension of an interval, which would extend the current interval to May 31, 2008. (Use of this 1-year extension does not require approval from the NRC.) In order to comply with ASME Code requirements, third interval visual examinations of the Category B-F welds must be performed during ANO-1's spring 2007 refueling outage (1R20). Entergy proposes to perform these examinations during the fall 2008 refueling outage (1R21).

## 2.0 REGULATORY REQUIREMENTS

In accordance with 10 CFR 50.55a, licensees are required to perform periodic inspections of components. Paragraph 50.55a(g) of 10 CFR requires that licensees perform surveillance testing in accordance with the ASME Code, Section XI requirements.

Paragraph 50.55a(a)(3) of 10 CFR states: "Proposed alternatives to the requirements of (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety." Entergy believes that compliance with the requirements of ASME Code, Section XI, IWA-2432 would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, Entergy requests the NRC staff approve this proposed alternative in accordance with 10 CFR 50.55a(a)(3)(ii).

## 3.0 TECHNICAL EVALUATION

### 3.1 Components for which Relief is Requested

The affected component is the ANO-1 reactor vessel (RV) dissimilar metal weld associated with the reactor core flood-line nozzle-to-safe end welds. They are ASME Code, Section XI, Examination Category B-F, and item number B5.10.

These examination categories and item numbers are from IWB-2500 and Table IWB-2500-1 of the ASME Code, Section XI, 1992 Edition.

### 3.2 ASME Code Requirements

ASME Code, Section XI, IWB-2412, 1992 Edition, Inspection Program B, requires volumetric examination of RV and piping pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. Paragraph IWA-2430(d) allows inspection intervals to be extended by as much as 1 year if this adjustment does not cause successive intervals to be altered by more than 1 year.

Item B5.70 of Table IWB-2500-1 requires that the nozzle-to-safe end butt welds have both volumetric and surface examinations every inspection interval. Note 1 of Table IWB-2500 for Examination Category B-F welds states, "[f]or the reactor vessel nozzle safe ends, the examinations may be performed coincident with the vessel nozzle examinations required by Examination Category B-D."

### 3.3 Licensee's Request for Relief

Entergy requests to extend the ISI interval for the identified examination Category B-F, identified in Sections 3.1 of this safety evaluation, to the end of 1R21 (approximately 180 days beyond the currently scheduled interval and the ASME Code-allowed 1-year extension).

## 3.4 Basis for Proposed Alternative

Given approval to extend the ANO-1's 10-year reactor vessel ISI interval to the end of 1R21 as contained in Entergy's Request for Alternative ANO1-ISI-005 (Reference 3), Entergy also seeks to extend the ASME Code-required core flood-line weld inspections until 1R21. In order to perform the core flood-line Examination Category B-F weld inspections, access to the weld from inside of the reactor vessel as described below is required. Therefore, performing this examination during the upcoming 1R20 refueling outage will result in a hardship without a compensatory increase in the level of quality and safety. The licensee’s justification supporting this request is provided below.

### A. Need to Perform Reactor Core Flood Line Dissimilar Metal Weld Examinations during Reactor Vessel ISI

The ANO-1 reactor vessel has two (2) ASME Code Class 1 (14-inch) carbon steel core flood nozzles with stainless steel safe ends. The carbon steel nozzles are lined with stainless steel cladding and have Alloy-82 butter with an Alloy-182 butt-weld to the stainless steel safe ends. The core flood lines connect directly into the reactor vessel and experience temperatures near that of the cold leg temperatures (approximately 577°F) during power operation.

As shown in Figure 1 [of the April 24, 2006, submittal], the nozzle butt welds are located between the reactor vessel and the concrete shield wall. Above these butt welds is the permanent reactor cavity seal plate used for refueling. This plate is located at the reactor vessel flange which is located at plant elevation 376 feet, 6 inches. The centerline of the core flood lines is at plant elevation 371 feet, 6 inches, which is 5 feet below the seal plate. Just below the seal plate is a series of four (4) concrete shield plugs. The core flood nozzles are located approximately one (1) foot below the shield plugs. Below the shield plugs is metal reflective insulation that surrounds the core flood lines.

Access to the core flood nozzles from above would require a modification to the reactor cavity seal plate to allow removal of the reactor cavity shield plugs. The existing man-ways in the reactor cavity seal plate do not allow for removal of the shield plugs. Access to the core flood nozzle outside diameter (OD) from below is not feasible due to the reactor vessel skirt configuration, which does not provide a man-way into the reactor vessel cavity. Access to the core flood nozzle OD from the shield wall penetration is also not feasible since that portion inside the reactor cavity is covered by removable reactor vessel insulation panels that cover the subject welds. Removal of the insulation panels to allow inspections is not feasible due to the distance through the penetration (approximately 6 feet), and the limited clearance between the piping and penetration walls (approximately 4 inches).

Therefore, the core flood nozzle examinations per ASME Section XI, IWB-2500, are performed from the inside of the reactor vessel as part of the reactor vessel

ISI at the normal interval frequency. As noted above, Entergy has submitted Request for Alternative ANO1-ISI-005 to defer the reactor vessel ISI (Reference 3).

B. Plant-Specific Reactor Vessel ISI History

ANO-1 is in its third inservice inspection interval for the reactor vessel; therefore, the preservice and two inservice inspections have been performed on the Examination Category B-F welds. These inspections achieved acceptable coverage; no reportable indications were found. Based on the examination method and coverage obtained, it is reasonable to conclude that the examinations were of sufficient quality to detect any significant flaws that could challenge reactor vessel integrity. The last examination findings are provided in the following table.

| ANO-1 ISI Results | | | | | | |
|---|---|---|---|---|---|---|
| **Weld ID** | **ASME Weld Category** | **Date Last Inspected** | **% Coverage Obtained** | **# of Reportable Indications** | **# of Indications Currently being Monitored** | **Growth of Indications Currently being Monitored (in)** |
| 01-025 | B-F | 1995 | 100 | 0 | 0 | N/A |
| 01-026 | B-F | 1995 | 100 | 0 | 0 | N/A |

C. Safety Significance of Core Flood Line Deferral for One Outage

General experience has shown that the incidence of primary water stress corrosion cracking (PWSCC) in PWR [pressurized-water reactors] primary coolant systems is due to tensile stresses and the operating temperature. The primary sources of tensile stresses are weld residual stresses and applied operating stresses (such as pressure and temperature). Higher mechanical stresses are typically due to fabrication activities where welding residual stresses cause shrinkage of the material around the weld.

For materials of equal PWSCC susceptibility with equal applied stresses, the time to crack initiation is a function of operating temperature. Locations that operate at higher temperatures, such as pressurizers, typically exhibit cracking much sooner than locations that operate at lower temperatures, such as cold legs. Therefore, Alloy 600 butt welds that operate closer to cold leg temperatures are generally much less susceptible to PWSCC than those that are close to Reactor Coolant System (RCS) operating temperatures.

EPRI [Electric Power Research Institute] document MRP-112 (Reference 1) documents the susceptibility of various locations of butt welds on Babcock & Wilcox (B&W) designed reactors. The assessment included a review of crack orientations and sizes, welding stresses, crack growth rates, limiting flaw sizes and the probability to determine the susceptibility of various butt welds. The

B&W core flood line butt welds were specifically evaluated in this report. The core flood lines operate at temperatures that are only slightly above the cold leg temperature (577°F). Critical flaw sizes for a through-wall flaw were determined using the methodology of ASME Code Section XI [and Code Case N-513]. The critical flaw sizes were determined for Axial and Circumferential through wall flaw lengths and for a 75% through-wall circumferential flaw depth. The core flood nozzle critical flaw sizes are as follows:

| Axial Flaw Length | Circ Flaw Length | Circ Flaw Depth |
|---|---|---|
| 22.3 inches | 20.7 inches | 0.75% |

Using the critical flaw data, crack growth analyses were performed under both PWSCC conditions and fatigue.

The results of this report for the B&W core flood lines show that the time for propagation of a flaw from an initial flaw size that produces an applied stress intensity factor equal to the PWSCC threshold stress intensity factor until 75% through-wall would take greater than 40 years. The time from identification of a 1-gpm [gallon per minute] leak until it reached a critical flaw size would be in excess of 70 years. The results of this analysis were also reflected in MRP-113 (Reference 2). The core flood lines at ANO-1 were inspected in 1995 during the last 10-year inspection interval and no flaws were identified in these welds. Therefore, the potential of the core flood line butt welds to represent a safety concern by deferral until 1R21 is very small.

In addition, the EPRI MRP performed a risk assessment for failures of Alloy 82/182 butt welds in MRP-116 (Reference 4). In this report, probabilistic fracture mechanics (PFM) evaluations were performed for a variety of locations of butt welds in B&W, Combustion Engineering (CE), and Westinghouse reactors. The PFM models are acceptable to the NRC in showing appropriate results. Specifically for B&W designs, the decay heat line and the surge line were modeled. The decay heat line would be more comparable to the core flood line based on RCS temperatures. The results of the analysis when conducting normal 10-year ISI exams over a 40-year life results in only a 3.75E-09 increase in core damage frequency (CDF). In addition, a sensitivity analysis was also performed that shows that if no ISI is performed over the 40 years, the change in relative risk only increases by 2%. Therefore, the relative risk increase from the decay heat line (comparable to core flood line) butt welds is fully acceptable within the guidance of Regulatory Guide 1.174 (Reference [5]).

D. Dose Reduction

As discussed above, Entergy has submitted to the NRC staff a request to defer performing the complete reactor vessel ISI via Request for Alternative ANO 1-ISI-005 (Reference 3). The complete reactor vessel ISI requires removing the core barrel. Deferring the examination of the core flood line DM [dissimilar metal] welds until the performance of the complete reactor vessel ISI [in the fall 2008 refueling outage] will consolidate activities and reduce personnel radiological exposure. Specifically, removing and replacing the core barrel in order to perform the weld examination involves approximately 600 mrem dose. The additional dose involved in performing the weld inspection separate from the remainder of the inspection is expected to be approximately 200 to 400 mrem. Therefore, performing this examination during the same planned evolution will result in a dose savings of up to 1 Rem since activities to support this examination (i.e., removing and replacing the core barrel and setup and performing the core flood line DM weld examination) will be performed only once rather than twice.

E. Evaluation of Core Flood Alloy 600 Mitigation

Entergy is evaluating mitigation of the Alloy 600 core flood nozzle DM weld during 1R21. As discussed above, these welds are only readily accessible from the inside diameter (ID) of the reactor vessel after the core barrel has been removed. Due to the relatively new technology to perform ID weld inlay mitigation and given the specific ANO-1 core flood line configuration, Entergy will utilize the additional time to 1R21 to investigate inlay mitigation design, analysis, and tooling.

## 3.5 Staff Evaluation

Inservice inspection of RV and piping pressure-retaining welds helps to ensure structural integrity by identifying flaw growth before flaws become large enough to represent challenges to pressure boundary integrity. The licensee provided a summary of examinations performed on the reactor core flood line nozzle-to-safe end welds in 1995. These examination findings show that with 100% coverage obtained, no reportable indications were found and no indications are currently being monitored. Therefore, the NRC staff agrees with the licensee's assessment that the prior examinations were of sufficient quality to identify any significant flaws that would challenge reactor vessel and piping pressure-retaining weld integrity.

The licensee discussed the safety significance of core flood line inspection deferral for one outage. They stated that the incidence of PWSCC in PWR primary coolant systems is due to tensile stresses and the operating temperature. The primary sources of tensile stresses are weld residual stresses and applied operating stresses (such as pressure and temperature). For materials of equal PWSCC susceptibility with equal applied stresses, the time to crack initiation is a function of operating temperature. Locations that operate at higher temperatures, such as pressurizers, typically exhibit cracking much sooner than locations that operate at lower temperatures, such as cold legs. Therefore, Alloy 600 butt welds that operate closer to cold-leg temperatures are generally much less susceptible to PWSCC than those that are close to RCS operating temperatures. The core flood lines operate at temperatures that are only slightly above the cold-leg temperature (577 °F). Critical flaw sizes for a through-wall flaw were determined using the methodology of ASME Code, Section XI. The critical flaw sizes were determined for axial and circumferential through-wall flaw lengths and for a 75-percent through-wall circumferential flaw depth. The core flood nozzle critical flaw sizes are as follows: axial flaw length - 22.3 inches, circumferential flaw length - 20.7 inches, and circumferential flaw depth - 75 percent. Using the critical flaw data, crack growth analyses were performed under both PWSCC conditions and fatigue. The results of these analyses show that the time for propagation of a flaw from an initial flaw size that produces an applied stress intensity factor equal to the PWSCC threshold stress intensity factor until 75-percent through-wall would take greater than 40 years. The time from identification of a 1-gpm leak until it reached a critical flaw size would be in excess of 70 years. The core flood lines were inspected in 1995 during the last 10-year inspection interval and no flaws were identified in these welds. Therefore, the staff agrees with the licensee's assessment that the potential of the core flood line butt welds to represent a safety concern by deferral until 1R21 is very small.

In addition, a risk assessment for failures of Alloy 82/182 butt welds was performed. In this assessment, PFM evaluations were performed for a variety of locations of butt welds. Specifically for ANO-1 type designs, the decay-heat line and the surge line were modeled. The decay-heat line would be more comparable to the core flood line based on RCS temperatures. The results of the analysis when conducting normal 10-year ISI exams over a 40-year life results in only a 3.75E-09 increase in CDF. In addition, a sensitivity analysis was also performed that shows that if no ISI is performed over the 40 years, the change in relative risk only increases by 2 percent. Therefore, the relative risk increase from the decay-heat line (comparable to core flood line) butt welds is fully acceptable within the guidance of Regulatory Guide 1.174 and this assessment is acceptable to the staff.

As discussed above, Entergy has submitted to the NRC staff a request to defer performing the complete reactor vessel ISI via Request for Alternative ANO 1-ISI-005. The complete reactor vessel ISI requires removal of the core barrel. Deferring the examination of the core flood line dissimilar metal welds until the performance of the complete reactor vessel ISI will consolidate activities and reduce personnel radiological exposure. Specifically, removing and replacing the core barrel in order to perform the weld examination involves approximately 600 mrem dose. The additional dose involved in performing the weld inspection separate from the remainder of the inspection is expected to be approximately 200 to 400 mrem. Therefore, performing this examination at the same time will result in a dose savings of up to 1 Rem since activities to support this examination (i.e., removing and replacing the core barrel and setup and performing the core flood line DM weld examination) will be performed only once rather than twice. Staff agrees with the licensee that by performing the same additional work during two separate outages, which could incur additional dose, a hardship condition is created which can be avoided by performing all of the inspections during the one refueling outage (1R21).

In summary, the NRC staff concurs with the licensee's assessment that their reactor core flood line nozzle-to-safe end welds have a low likelihood of having significant flaws and that there is a low likelihood of experiencing a severe event during the proposed extension period. Therefore, the staff believes that performance of ISI of the reactor core flood line nozzle-to-safe end welds during the next refueling outage (1R20) would not provide an additional level of safety or quality in comparison to deferring the examination for one refueling cycle (1R21). The staff finds that the risk associated with the one-cycle extension of the examination interval is sufficiently small and coupled with the dose savings that would result, the alternative provides reasonable assurance of structural integrity and that compliance with the specified requirements of ASME Code, Section XI, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

## 4.0 CONCLUSION

Based on the above evaluation, the NRC staff concludes that the licensee's proposed alternative provides reasonable assurance of structural integrity and that compliance with the specified requirements of ASME Code, Section XI, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the NRC staff authorizes the extension of the third 10-year ISI interval not to exceed the end of the fall 2008 refueling outage, to complete the ANO-1 reactor core flood-line nozzle-to-safe end welds examinations.

All other requirements of the ASME Code for which relief has not been specifically requested remain applicable including third-party review by the Authorized Nuclear Inservice Inspector.

## 5.0 REFERENCES


1. Alloy 82/182 Pipe Butt Weld Safety Assessment for US PWR Plant Designs: Babcock & Wilcox Design Plants (MRP-112), EPRI (October 2004).

2. Alloy 82/182 Pipe Butt Weld Safety Assessment for U.S. PWR Plant Designs (MRP-113), EPRI (July 2004).

3. Entergy Operations, Inc., letter CNRO-2006-00024, Request for Alternative ANO-ISI-005 - Proposed Alternative to Extend the Third Inservice Inspection Interval for Reactor Vessel Inservice Examinations, dated April 24, 2006.

4. Probabilistic Risk Assessment of Alloy 82/182 Piping Butt Welds (MRP-116), EPRI (August 2004).

5. NRC Regulatory Guide 1.174, Revision 1, An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plan Specific Changes to the Licensing Basis, November 2002.


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Date: February 1, 2007