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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE SECOND TEN YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

REQUEST FOR RELIEF NO. 95-001

FOR

ENTERGY OPERATIONS, INC.

ARKANSAS NUCLEAR ONE, UNIT 1

DOCKET NO. 50-313

## 1.0 INTRODUCTION

The Technical Specifications for Arkansas Nuclear One, Unit 1 state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable edition of Section XI of the ASME Code for the Arkansas Nuclear One, Unit 1 second 10-year inservice inspection (ISI) interval is the 1980 Edition through the Winter 1981 Addenda. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

ENCLOSURE 1

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and a request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are otherwise in the public interest, giving due consideration to the burden upon the licensee that could result if the requirements were imposed. In a letter dated May 31, 1995, Entergy Operations, Inc., submitted to the NRC its Second 10-Year Interval Inservice Inspection Program Plan Request for Relief No. 95-001 for Arkansas Nuclear One, Unit 1. Additional information was provided by the licensee in its letters dated October 24, 1995, and March 25, 1996.

## 2.0 EVALUATION AND CONCLUSIONS

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its Second 10-Year Interval Inservice Inspection Program Plan Request for Relief No. 95-001 for Arkansas Nuclear One, Unit 1.

Based on the information submitted, the staff adopts the contractor's conclusions and recommendations presented in the Technical Letter Report. The staff has concluded that based on its evaluation that the examination device currently deployed to examine the reactor vessel circumferential bottom-head-weld 01-006 may cause damage if the device impacts with in-vessel components thus, requiring the licensee to perform the second 10-year Code-required examination with the subject examination device will result in a burden without a compensating increase in quality and safety. Therefore, the alternative contained in Request for Relief No. 95-001 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second interval only, on a one-time basis. In addition, the licensee should address its concern regarding the ultrasonic equipment that could cause in-vessel damage with the examination vendor to minimize this potential for future examinations.

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# TECHNICAL LETTER REPORT
# ON THE SECOND 10-YEAR INSERVICE INSPECTION INTERVAL
# REQUEST FOR RELIEF NO. 95-001
# FOR
# ARKANSAS NUCLEAR ONE, UNIT 1
# ENTERGY OPERATIONS, INC.
# DOCKET NO. 50-313

## 1.0 INTRODUCTION

By letter dated May 31, 1995, Entergy Operations, Inc. submitted Request for Relief No. 95-001 for the reactor pressure vessel transition-piece-to-head circumferential Weld 01-006. In letters dated October 24, 1995, and March 25, 1996, the licensee provided additional information in support of the request for relief.

## 2.0 EVALUATION

The Code of record for the Arkansas Nuclear One, Unit 1, second 10-year inservice inspection (ISI) interval, which began December 1984, is the 1980 Edition through the Winter 1981 Addenda of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI*. The information provided by the licensee in support of the request for relief from Code requirements has been evaluated, and the basis for disposition is documented below.

Request for Relief No. 95-001, Examination Category B-A, Item B1.21, Reactor Pressure Vessel Circumferential Head Weld 01-006

Code Requirement: Table IWB-2500-1, Examination Category B-A, Item B1.21 requires 100% volumetric examination of the accessible length of one reactor pressure vessel circumferential head weld.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required 100% volumetric examination of the accessible portions of lower head circumferential Weld 01-006.

ENCLOSURE 2

Licensee's Basis for Requesting Relief (as stated):

"Accessibility to this weld is severely limited by the flow stabilizer lugs, and the incore instrumentation nozzles. ASME Code coverage of the weld and base material would be limited to approximately 10%. In order to achieve this very limited examination, Entergy Operations estimates that the critical-path outage time required would be a minimum of 12 hours, which is estimated to cost approximately $250,000.

"The obstructions located on and adjacent to this weld produce an area in which it is very difficult to maneuver the ultrasonic transducer manipulator. Of particular interest are the incore instrumentation nozzles. These nozzles are small and manufactured to close tolerances. If inadvertently bumped by the manipulator, these nozzles could be damaged. A damaged nozzle could prevent the reinsertion of an incore instrument or could require a critical-path in-vessel repair.

"The majority of the neutron flux escaping from the nuclear core impacts the reactor vessel beltline area, rather than the bottom head. Therefore, the potential for neutron embrittlement of this bottom-head is considerably less than for the beltline welds. If a flaw were to exist in this weld, it would be far less likely to propagate than if a flaw of the same size and configuration were to exist in one of the shell welds. During the most recent refueling outage (1R12), which ended in April 1995, the remainder of the reactor vessel welds scheduled to be inspected during the second interval were examined, including all of the beltline welds. Since no service-induced flaws were found, Entergy Operations has a high degree of confidence that the structural integrity of the reactor vessel is assured.

"The likelihood of a significant flaw existing in this weld is very small. When the vessel was originally fabricated, this full-penetration weld was completely radiographed and found to be acceptable. Since that time, the weld has been ultrasonically inspected once preservice, prior to installation (essentially 100% coverage), and once during the first interval (approximately 10% coverage). Both of these examinations determined the weld to be satisfactory."

In the licensee's October 24, 1995 submittal, the licensee provided the following information.

"Weld 01-006 was examined 100% during the preservice examination when the RPV was still in a shop environment being fabricated. After the RPV had been completed, but before the various lugs and instrumentation nozzles were added, the entire circumference of this weld was examined manually from the outside surface of the vessel using ultrasonic equipment. Performing the examination at this time provided access that would not ever be available again. A full preservice examination of this weld was determined to be desirable and beneficial.

"Arkansas Nuclear One, Unit 1 (ANO-1), was committed to two different editions of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code for inservice inspections during the first 10-year interval. During the first period (approximately three years) of the first 10-year interval, inservice inspections were performed to the 1971 edition with addenda through the Summer 1972 (71-S72). For the last two periods of the first 10-year interval, ANO-1 was committed to the 1974 edition with addenda through the Summer 1975 (74-S75). While the 74-S75 Code required only 5% coverage of this circumferential weld, all accessible portions of the weld were examined to the fullest extent possible.

"The length of the weld is approximately 40 feet on the outside diameter. On the interior of the RPV, two sets of lugs are attached to the cladding by welding. The 12 flow stabilizer lugs are located on and above the subject weld. These lugs are especially restrictive to the examinations since they are mounted in the vessel at a 30-degree angle to the vertical. In addition, immediately above the flow stabilizer lugs are the 12 guide lugs, also called core stop lugs. Although these lugs are farther from the weld, they still interfere with the manipulator arm and transducer head of the inspection tool while it attempts to examine the required base metal adjacent to the weld. Detailed drawings of the lower head and lower shell areas are attached."[1]

In the March 25, 1996 submittal, the licensee provided additional information in support of the request for relief. This information included *Flaw Acceptance Standards for Arkansas Nuclear One, Unit 1 Reactor Pressure Vessel Weld Inspections*, prepared by Structural Integrity Associates, Inc., Report No. SIR-95-017, Revision 0, dated February 22, 1995, and *Input to Items 2A and 2C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Arkansas Nuclear One, Unit 1*, prepared by Structural Integrity Associates, Inc., Report No. SIR-96-022, Revision 0, dated February 1996.

In the Conclusions of the *Input to Items 2A and 2C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Arkansas Nuclear One, Unit 1*, the following statement was given:

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Drawings provided by the licensee are not included with this evaluation.

"This evaluation has demonstrated that service-induced degradation of the transition piece to bottom head weld as a result of corrosion, fatigue, or thermal embrittlement mechanisms is extremely unlikely. The primary contributor to the presence of flaws in this weld is due to fabrication. Prior vessel inspections did not identify any flaws such that the existence of any significant fabrication related defects in the bottom head weld is unlikely. Relief from inspection of the transition piece to bottom head weld in the ANO-1 reactor vessel appears to be justified."

Licensee's Proposed Alternative Examination (as stated):

"As part of the recent 1R12 scheduled outage scope, all of the reactor vessel shell welds were examined ultrasonically. In addition, the reactor vessel interior surfaces and interior welded attachments received visual (VT-1 and VT-3) inspections as required by Section XI of ASME Code. Also, a visual (VT-2) examination is performed on the exterior of the vessel each refueling. No service-induced cracking or degradation has been found either with the ultrasonic examinations or with the visual inspections."

Evaluation: The Code provides examination requirements for reactor pressure vessel welds. In the case of the reactor pressure vessel transition piece-to-bottom head weld (Item B1.21), a volumetric examination of the accessible portion of the subject weld is required. Entergy Operations, Inc. requested relief from performing the Code-required volumetric examination of the reactor pressure vessel transition piece-to-bottom head weld. The licensee has cited hardships associated with the examination of the subject weld that include minimal volumetric coverage (less than 7%[2]), potential for damage to and subsequent repair of incore instrumentation tubes, and the minimal possibility of flaws existing in the subject weld.

The licensee contends that the requirement to perform a volumetric examination on the accessible portion of the subject weld results in a hardship. This is due in part to the currently available ultrasonic

[2]As stated in the March 25, 1996 submittal.

inspection technique being employed by the licensee. Reactor pressure vessel inservice examinations will be performed utilizing a contact ultrasonic examination technique. This technique requires that the scanning head be in contact with the vessel inside surface. As a result of the use of the contact technique, scanning to maximize volumetric coverage becomes more critical and limiting due to obstructions such as guide lugs, flow stabilizers, and incore instrumentation guide tubes. The licensee estimates that the volumetric coverage of the subject weld with the contact technique will be less than 7%.

Considering (1) the licensee's concern for potential damage to in-vessel components that could occur when examining with the contact ultrasonic technique, (2) the minimal coverage obtainable on the subject weld, and (3) the fracture analysis[3] that supports the low potential for flaw initiation and larger allowable flaw size, based on lower overall stresses associated with the subject weld and material ductility (affected less by neutron embrittlement), the INEL staff concurs that performing the Code-required volumetric examination for the second interval would result in a burden without providing a compensating increase in quality and safety. However, the licensee's concern over ultrasonic equipment that could cause in-vessel damage should be addressed with the examination vendor to minimize this potential for all future examinations.

The licensee has proposed to perform VT-1 and VT-3 visual examinations within and beyond the beltline region and a VT-2 visual examination on

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[3] *Flaw Acceptance Standards for Arkansas Nuclear One, Unit 1 Reactor Pressure Vessel Weld Inspections*, prepared by Structural Integrity Associates, Inc., Report No. SIR-95-017, Revision 0, dated February 22, 1995, and *Input to Items 2A and 2C of NRC's Questions on Relief Request for Inspection of Transition Piece to Bottom Head Weld at Arkansas Nuclear One, Unit 1*, prepared by Structural Integrity Associates, Inc., Report No. SIR-96-022, Revision 0, dated February 1996.

the exterior of the reactor vessel during the inservice leak test. The INEL staff believes that the proposed alternative examination in conjunction with the volumetric examination of other reactor pressure vessel welds will provide reasonable assurance of continued structural integrity of this component. Therefore, in consideration of the licensee's concerns associated with the Code-required examination for the second interval, for Arkansas Nuclear One, Unit 1, the INEL staff believes that the licensee's alternative contained in Request for Relief No. 95-001 should be authorized on a one-time basis only, pursuant to 10 CFR 50.55a(a)(3)(ii).

## 3.0 CONCLUSION

Based on this evaluation, the INEL staff concurs that requiring the licensee to perform the second 10-year Code-required examination with the examination device currently being deployed for in-vessel examinations will result in a burden without a compensating increase in quality and safety. This is based on the potential damage that may occur if the examination device impacts with in-vessel components. However, considering that the Code requires the examination of only the accessible portion of the subject weld, it is recommended that the licensee's alternative contained in Request for Relief No. 95-001 should be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second interval only, on a one-time basis.