Mr. Jerry W. Yelverton
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

February 15, 1996

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION PERTAINING TO INSERVICE INSPECTION RELIEF REQUEST 95-001 - ARKANSAS NUCLEAR ONE, UNIT 1 (TAC M94384)

Dear Mr. Yelverton:

By letter dated May 31, 1995, Entergy Operations, Inc. submitted Inservice Inspection Relief Request 95-001 for Arkansas Nuclear One, Unit 1. The requested relief pertains to the inservice inspection of the transition-piece-to-bottom-head weld inside the Unit 1 reactor vessel. In response to questions from our technical staff, you forwarded additional information in a letter dated October 24, 1995. We have reviewed your submittals and need additional information to complete our review. The enclosure to this letter contains the request for additional information. To expedite our review, please send a copy of your response directly to the Idaho National Engineering Laboratory (INEL) at the following address:

INEL Research Center
2151 North Boulevard
P. O. Box 1625
Idaho Falls, Idaho 83415-2209.
Attn: Mr. Michael T. Anderson

INEL in under contract to the NRC to participate in the review of inservice inspection relief requests. Should you have any questions with regard to this request, please communicate with your NRC project manager. This requirement affects nine or fewer respondents and, therefore, is not subject to the Office of Management and Budget review under P.L. 96-511.

Sincerely,

ORIGINAL SIGNED BY:
George Kalman, Senior Project Manager
Project Directorate IV-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-313

Enclosure: Request of Additional Information

cc: See next page
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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George Kalman, Senior Project Manager
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Mr. Jerry W. Yelverton
Entergy Operations, Inc.

Arkansas Nuclear One, Unit 1

cc:

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ARKANSAS NUCLEAR ONE, UNIT 1
INSERVICE RELIEF REQUEST 95-001
REQUEST FOR ADDITIONAL INFORMATION

1. Scope/Status of Review

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 meet the requirements, except design and access provisions and 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. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during the successive 120-month inspection interval shall comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of a successive 120-month interval, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of the Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Nuclear Regulatory Commission (NRC) approval. The licensee, Entergy Operations, Inc., prepared the *Arkansas Nuclear One, Unit 1, Second 10-Year Interval Inservice Inspection (ISI) Program Plan* to meet the requirements of the 1980 Edition of Section XI of the ASME Code with Addenda through the Winter 1981.

As required by 10 CFR 50.55a(g)(5), if the licensee determines that certain Code examination requirements are impractical and requests relief, the licensee shall submit information to the NRC to support that determination.

The staff has reviewed the available information in the Arkansas Nuclear One, Unit 1, Request for Relief 95-001, submitted May 31, 1995, and additional information submitted October 24, 1995.

2. Additional Information Required

Based on the above review, the staff has concluded that additional information and/or clarification is required to complete the review of Request for Relief 95-001, relief from the examination of the reactor pressure vessel transition-piece-to-bottom-head weld (01-006).

ENCLOSURE

Code Requirement: The 1980 Edition of Section XI with Addenda through the Winter 1981, 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 after the first interval. Note: The 1988 Addenda of Section XI and later, require the examination of the accessible portions of all reactor pressure vessel welds during each interval.

Licensee's basis for relief:

a) Cost Reduction (12 hours Critical Path, $250,000).

b) Limited coverage (Approximately 10%).

c) The subject weld was examined 100% preservice and 10% inservice in the first interval and found to be satisfactory.

d) The transition-to-bottom head weld is not subjected to neutron flux similar to the beltline region and is therefore less susceptible to neutron embrittlement.

e) Difficulty in maneuvering the examination tool because of obstructions, (i.e., instrumentation nozzles and lugs make it difficult to maneuver the ultrasonic transducer), and potential for damage of incore instrumentation tubes if bumped by the manipulator.

To obtain relief from examination of the reactor pressure vessel transition-piece-to-bottom-head weld, the basis for relief should address the key areas that are described below.

A. Discussion of Potential Damage Mechanisms - The licensee has cited neutron embrittlement as a potential damage mechanism for the shell welds in the beltline region only. The licensee should also address the following:

   The reactor pressure vessel transition-piece-to-bottom-head weld is of a lesser wall thickness than the shell welds. Address the stresses and potential damage mechanisms associated with this weld. The discussion should include but not be limited to affects of potential neutron embrittlement on the subject weld (considering the reduced wall thickness), corrosion, loads associated with welded attachments (12 flow stabilizer lugs are located on and above the subject weld), lower head penetrations, expansion/contraction stresses associated with reactor operation cycles and operating conditions.

B. Confidence that no flaw is present in the weld - The licensee has stated that the likelihood of a significant flaw existing in this weld is very small. In the case of the fabrication, preservice, and inservice examinations, the weld was found to be satisfactory. Confirm that there are no preexisting, recordable flaws, acceptable by Code.

C. Structural integrity - The licensee essentially proposes the elimination of the subject volumetric Code examination of the accessible portions of the weld. This implies that other RPV welds are more susceptible to failure than the subject weld. Based on a qualitative comparison of the fracture toughness of the beltline weld to the lower head weld, what is the estimated critical flaw size for the lower head weld (Appendix G ASME Code flaw size)?

D. Radiation fields - The licensee has not addressed the radiation dose potential associated with the examination of the subject weld. Provide information on the estimated exposure associated with the examination of the subject weld.

E. Potential for Damage Caused by Examinations - The licensee cites limited access for examination and the potential for damage of incore instrumentation by the examination tool. Provide a detailed access study and determine the actual probability for potential damage due to the inspection tooling, (i.e., considering clearance requirements, tool operations, etc.). In addition, provide instances where damage, if any may have been associated with the subject weld has occurred, the result of the use of the inspection tool at your plant or at any other plant with similar reactor pressure vessel designs.