



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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION  
OF THE  
SECOND 10-YEAR INTERVAL INSERVICE INSPECTION  
REQUEST FOR RELIEF  
FOR  
TENNESSEE VALLEY AUTHORITY  
BROWNS FERRY NUCLEAR PLANT, UNIT 2  
DOCKET NUMBER: 50-260

1.0 INTRODUCTION

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 U.S. Nuclear Regulatory Commission (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 difficulty 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 pre-service 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 ten-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of the Section XI of the ASME Code incorporated by reference in the 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 Browns Ferry Nuclear Plant (BFN), Unit 2, is the 1986 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

By letter dated June 12, 1998, the licensee proposed an alternative to the Code requirements contained in the request for relief for BFN, Unit 2.

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Enclosure



## 2.0 EVALUATION

The staff, with technical assistance from its contractor, the Idaho National Engineering and Environmental Laboratory (INEEL), has evaluated the information provided by the licensee in support of the licensee's proposed alternative contained in its request for relief for BFN Unit 2. Based on the results of the review, the staff generally adopts the contractor's conclusions and recommendations presented in the Technical Letter Report (TLR) attached.

Request for Relief 2-SPT-11 (Revision 1): This request for relief involves the use of a detailed Engineering Evaluation as an Alternative to the ASME Code Section XI, Paragraph IWA-5250(a)(2), Corrective Measures For Bolted Connections, for Class 1,2, and 3 Pressure Retaining Bolted Connections.

In accordance with the ASME Code, 1986 Edition, Section XI, Paragraph IWA-5250(a)(2) requires bolting to be removed if leakage occurs at the connection, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee has proposed an alternative to the requirements of IWA-5250(a)(2), that upon completion of a detailed engineering evaluation it indicates the need for further evaluation, or no evaluation is performed, then only the bolt closest to the leakage shall be removed. The bolt will receive a VT-1 visual examination and be evaluated for corrosion in accordance with IWA-3100(a) and dispositioned in accordance with IWB-3140.

The Code states that if leakage occurs at a bolted connection, the bolting must be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100. However, IWA-3100 does not provide acceptance standards for VT-3 visual examination. As a result, the licensee proposed to evaluate the bolting to determine its susceptibility to corrosion. The proposed engineering evaluation by the licensee, in lieu of IWA-5250(a)(2) requirements, are similar to Code Case N-566. The alternative evaluation like Code Case N-566, will examine fastener materials, the corrosive nature of the process fluid, the leakage location and history, components in the vicinity of the leakage that may be degraded, and visual evidence of corrosion at the assembled connection.

## 3.0 CONCLUSION

Based on the items presented in the proposed alternative, the staff agrees that the evaluation proposed by the licensee provides an acceptable level of quality commensurate to the Code requirements. In addition, if initial evaluation indicates the necessity for detailed analysis, the bolt closest to the source of leakage will be removed, VT-1 visually examined, and evaluated in accordance with IWA-3100(a). The VT-1 examination criteria is more stringent than the corrosion evaluation described in IWA-5250(a)(2). Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the staff authorizes the licensee's alternative for the second interval for Class 1, 2, and 3 pressure retaining bolted connections at BFN, Unit 2.

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Attachment: INEEL Technical Letter Report

Date: April 8, 1999

TECHNICAL LETTER REPORT  
ON SECOND 10-YEAR INTERVAL INSERVICE INSPECTION  
REQUEST FOR RELIEF 2-SPT-11, REV. 1  
FOR  
TENNESSEE VALLEY AUTHORITY  
BROWNS FERRY NUCLEAR PLANT, UNIT 2  
DOCKET NUMBER: 50-260

1. INTRODUCTION

By letter dated June 12, 1998, the licensee, Tennessee Valley Authority, submitted Request for Relief 2-SPT-11, Rev. 1, seeking relief from the requirements of the ASME Code, Section XI, for the Browns Ferry Nuclear Plant (BFN), Unit 2, second 10-year inservice inspection (ISI) interval. The Idaho National Engineering and Environmental Laboratory (INEEL) staff's evaluation of the subject request for relief is in the following section.

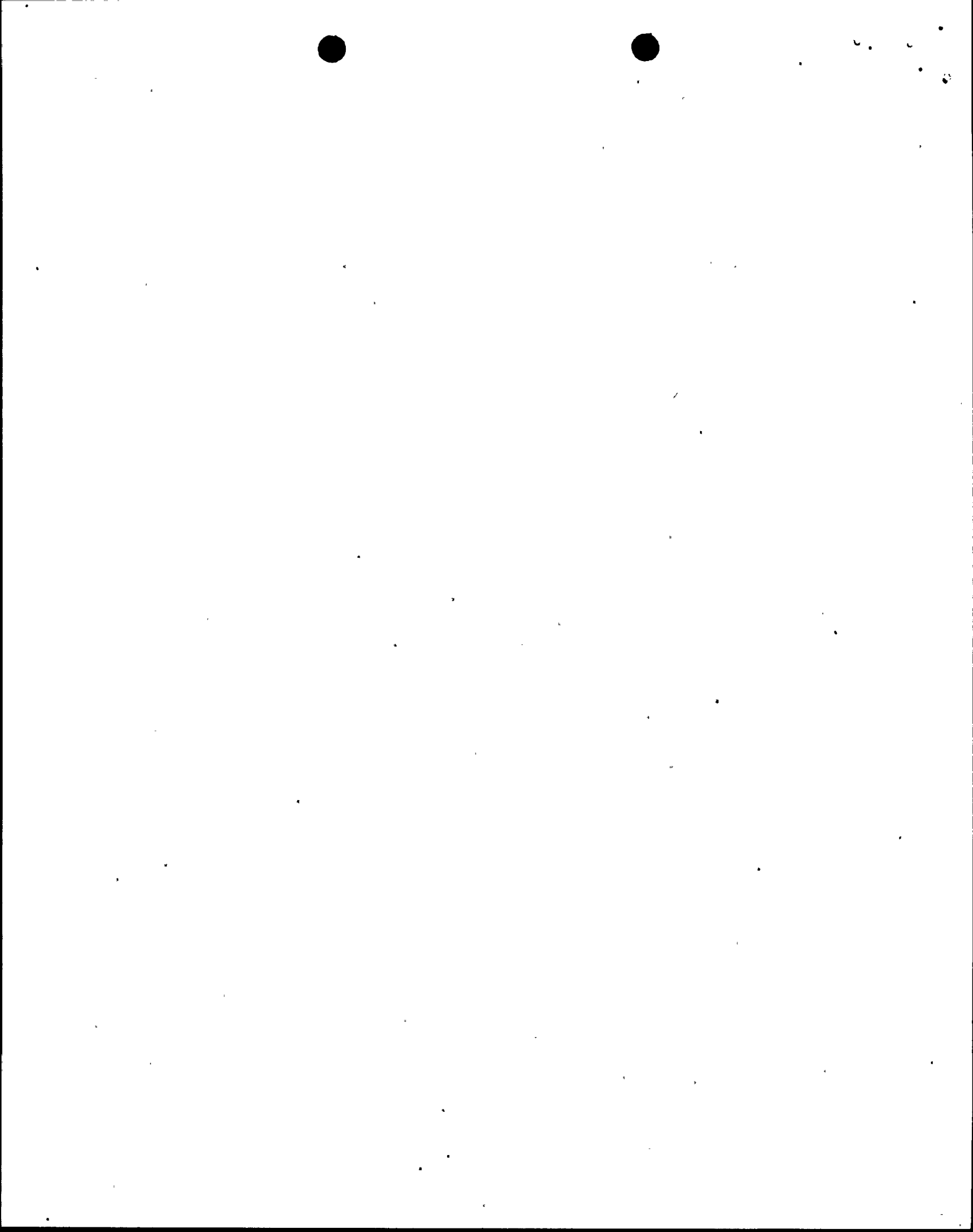
2. EVALUATION

The information provided by Tennessee Valley Authority in support of the request for relief from Code requirements has been evaluated and the basis for disposition is documented below. The Code of record for the Browns Ferry Nuclear Plant, Unit 2, second 10-year ISI interval, which began May 24, 1992, is the 1986 Edition of Section XI of the ASME Boiler and Pressure Vessel Code.

Request for Relief No. 2-SPT-11, Rev. 1, Proposed Alternative to IWA-5250, Corrective Measures for Leakage at Bolted Connections

Code Requirement: Paragraph IWA-5250(a)(2) states that if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Proposed Alternative: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative, namely an engineering evaluation of the bolted connection in lieu of mandatory removal of all bolts when leakage is detected. The



licensee stated:

"As a proposed alternative, in accordance with 10 CFR 50.55(a)(3)(i), to the mandatory removal of bolting from leaking bolted connections (ASME Code Section XI, Subarticle IWA-5250(a)(2)) it is requested that a corrective action plan should be allowed following a specific evaluation of the bolted connection structural integrity and susceptibility of the bolting to corrosion and potential failure. The corrective action plan may or may not require removal of bolting.

"As an alternative to the existing Section XI requirements, the source of all leakage at bolted connections detected by VT-2 examination during a system pressure test shall be evaluated to determine the susceptibility of the bolting to corrosion and potential failure. This evaluation will consider the following variables at a minimum:

- Location of leakage
- History of leakage
- Fastener materials
- Evidence of corrosion, with the connection assembled
- Corrosiveness of the process fluid
- History and studies of similar fastener material in a similar environment
- Other components in the vicinity that may be degraded due to the leakage

"When the evaluation of the above variables is concluded, and if the evaluation determines that the leaking condition has not degraded the fasteners, then no further action is required. However, reasonable attempts to stop the leakage shall be taken.

"If the bolted connection evaluation, using the variables above, indicates the need for further evaluation, or no evaluation is performed, then a bolt closest to the source of leakage shall be removed. The bolt will receive a VT-1 examination and be evaluated for corrosion in accordance with IWA-3100(a) and dispositioned in accordance with IWB-3140. If the information from the bolted connection evaluation is supportive, the removal of the bolt for VT-1 examination may be deferred to the next refueling outage. When the removed bolting shows evidence of rejectable degradation, all remaining bolts, in the connection, shall be removed and receive a VT-1 examination and evaluation in accordance with IWB-3140."

Licensee's Basis for Proposed Alternative (as stated):

"Relief from the bolt removal requirements of IWA-5250(a)(2) is requested under 10 CFR 50.55(a)(3)(i), in which the proposed alternative provides an acceptable level of quality and safety. Some of the problems associated with the current requirements of IWA-5250(a)(2) are summarized as follows:

IWA-3100 does not provide an acceptance standard for a VT-3 bolt inspection.

The requirement calls for bolt removal without regard to the size of the leakage.

The requirement increases the radiological dose to workers for leaks that are often not a challenge to operational or structural limits.

Bolts sometimes cannot be removed without damaging the bolt or cannot be removed due to component configuration.

It is not a requirement of the Code that the licensee must stop the leakage, and inspection of the bolting is not necessarily going to stop the leak.

Removing one bolt at a time, if allowed by system conditions, may actually increase the leakage.

In many cases, implementation of the requirement may cause the plant an unnecessary transient or delay startup.

"In addition to the problems associated with the requirements of IWA-5250(a)(2), the ASME Working Group-Pressure Testing concluded that the system integrity of a bolted connection is not necessarily compromised by leakage and recommended the approval of Code Case N-566. This relief request is essentially a conservative subset of the Code Case.

"This relief request is more prescriptive and more conservative than the Code Case. It also addresses many of the implementation and radiological hardships associated with IWA-5250(a)(2) and maintains the conclusions of the ASME Committee by assuring that a proper evaluation of the connection and/or the bolting is performed. The bolted connection evaluation must consider specific factors which, if indicative of degradation, must be dispositioned in accordance with IWB-3140 of ASME Section XI. Due to the fact that the bolted connection evaluation is more comprehensive than the simple bolt inspection currently required by IWA-5250, coupled with the benefit that these alternative requirements ensure structural integrity is maintained, and reduce the operational, maintenance, and radiological hardships of the current requirements, this relief request provides an acceptable level of quality and safety and should be considered as an acceptable alternative in accordance with 10 CFR 50.55a(a)(3)(i). This conclusion is further supported by the fact that the ASME has approved Code Case N-566 and this relief request is essentially a conservative subset of the Code Case."

Evaluation: In accordance with IWA-5250(a)(2), if leakage occurs at a bolted connection, the bolting must be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100. In lieu of this requirement, the licensee has proposed to evaluate the bolting to determine its susceptibility to corrosion. The proposed evaluation will consider, as a minimum, bolting materials, the corrosive nature of the process fluid, the leakage location and history, the service age of the bolting materials, and visual evidence of corrosion at the assembled connection. If the initial evaluation indicates the need for a more detailed analysis, the bolt closest to the source of leakage will be removed, VT-1 visually examined, and evaluated in accordance with IWA-3100(a).

Based on the items included in the evaluation process, the INEEL staff believes that the

evaluation proposed by the licensee is a sound engineering approach. In addition, the VT-1 examination criteria are more stringent than the simple corrosion evaluation described in IWA-5250. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), it is recommended that the licensee's proposed alternative be authorized for the second interval at BFN, Unit 2.

3. CONCLUSION

The INEEL staff has reviewed the licensee's submittal and concludes that for Request for Relief 2-SPT-11, Rev. 1, the licensee's proposed alternative will provide an acceptable level of quality and safety. Therefore, it is recommended that this proposed alternative be authorized for the second interval pursuant to 10 CFR 50.55a(a)(3)(i).



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**BROWNS FERRY NUCLEAR PLANT**

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