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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF REQUESTS FOR RELIEF

FACILITY OPERATING LICENSE NO. DPR-52

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNIT 2

DOCKET NO. 50-260

1.0 INTRODUCTION

Technical Specifications 4.6.G.1 for Browns Ferry Nuclear Plant (BFN), Unit 2, states that inservice inspection and testing 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 each ten-year interval 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) on the date twelve months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein. 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.

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.

By letters dated April 8, 1993, April 27, 1993, and May 16, 1993, the Tennessee Valley Authority (the licensee) submitted to the NRC Requests for Relief (Nos. SPT-4, STP-5, and SPT-6) from the ASME Code Section XI requirements that the licensee determined to be impractical to perform during the Second Ten-Year ISI interval for Browns Ferry Power Plant, Unit 2. Additional information was provided by the licensee in letters dated May 12, 1993 and May 21, 1993. In the licensee's letter dated May 12, 1993, it withdrew Requests for Relief SPT-1, SPT-2, and SPT-3 regarding ASME Code Case N-498. The staff has evaluated the licensee's BFN, Unit 2 requests for relief from the Code requirements, and its evaluations and conclusions are discussed in the following sections.

2.0 EVALUATION

Request for Relief SPT-4 Leakage Detection of Reactor Pressure Vessel Retaining Boundary - Control Rod Drive Housing Cap Screws Pressure

Code Requirement: Section XI, IWA-5250(a) requires that the source of leakage detection during the conduct of a system pressure test shall be located and evaluated by the Owner for the corrective measures as follows:

> IWA-5250(a)(2) If leakage occurs at the bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3100.

Licensee's Basis for Requesting Relief: The licensee stated that compliance with subparagraph IWA-5250(a)(2) in the event of leakage at the CRD housing connection would result in an extreme hardship which is not commensurate with the increased level of safety that would be achieved. The hardship is due to the requirement that for any CRD connectors where leakage is detected during the pressure test the housing connector cap screws must be removed and examined. This requires that the reactor pressure vessel (RPV) be depressurized and the CRD housing be removed to permit removal of the CRD. Due to the torquing sequence requirements the cap screws cannot be removed individually for examination.

The reactor vendor, General Electric (GE), has informed boiling water reactor (BWR) owners that leakage from these cap screw connections is a common occurrence, and in most instances, leakage stops within eight (8) hours of the connection being pressurized to 1000 psig. GE has recommended the replacement of these cap screws with a new design and higher strength material cap screw and a new designed washer to facilitate drainage. This new design is being incorporated at Browns Ferry. The cap screws for 26 CRD housings have been changed during the BFN, Unit 2 cycle 6 refueling outage. The remaining cap screws will be changed as the CRDs are replaced or maintenance is performed.

GE has determined that based on the evaluation of crack data, structural integrity and plant safety are not affected by this situation. This is based in part on the following:

1. Three uniformly distributed uncracked cap screws are capable of supporting the CRD loads, and the probability that through-wall cracks will occur in five or more cap screws on a single CRD housing is extremely small;

2. If such a failure were to occur leakage at the connection would proceed failure, and the leak detection system and drywall temperature monitoring system would detect this leakage;

3. The CRD support structure under the reactor vessel would allow the CRD to drop a maximum of one inch;

4. The evaluation of the loss of one CRD has been considered in the plant safety analysis report.

<u>Licensee's Proposed Alternative Examination:</u> The licensee proposes that during the Class 1 component leakage test following refueling, all leakage from the CRD housing connections will be documented and evaluated based on the GE recommendations. The VT-3 examinations of all eight cap screws at CRD housing connections where leakage is detected during the leakage test will be deferred to the next refueling outage.

<u>Staff Evaluation:</u> The staff determined that it would be impractical for the licensee to perform the Code requirement to remove bolting, and VT-3 visually examined the removed bolting if a source of leakage should occur at a CRD housing bolted connection. The torquing sequence requirements do not allow the screws from being removed individually for examination and to remove the cap screws would require that the reactor pressure vessel (RPV) be depressurized and the CRD housing be removed to permit removal of the CRD.

The staff has reviewed the information presented by the licensee and has concluded that an acceptable level of quality and safety, and reasonable assurance that the structural integrity of the plant's systems, components, and supports, will be maintained.

<u>Request for Relief SPT-5 - Pressure Testing After Weld Repair Core Spray 12 inch Pipe Weld and 2-inch Socket Weld</u>

<u>Code Requirement:</u> IWA-4400(a) requires that after repairs by welding on the pressure retaining boundary, a system hydrostatic test shall be performed in accordance with IWA-5000.

<u>Licensee's Basis for Requesting Relief:</u> The licensee stated that the weld repaired areas are situated in their respective systems between the reactor pressure vessel (RPV) and the first isolation valve of the vessel. These locations provide no method to isolate the repaired areas of the RPV for the purpose of the hydrostatic test required by Article IWA-4000.

The performance of a hydrostatic pressure test of the repaired areas by pressurizing the RPV would require the removal and blanking of 8 of 13 main steam relief valves cartridge assemblies to prevent the valve lifting and possibly resulting in valve seat damage. The estimated exposure for the removal and replacement of the eight (8) relief valves is 0.48 man-rem. In addition, performance of the hydrostatic test requires the reactor vessel high pressure SCRAM switch to be disabled.

The integrity of the weld repairs have been ensured by the following nondestructive examinations:

1. The 12 inch core spray piping overlay repair was examined using volumetric (UT) and surface (PT) techniques;

2. The two 2-inch socket welds were surface examined (PT).

To further ensure the integrity of the repair welds, the reactor coolant system will be pressurized to a minimum test pressure of 1034 psig (measured at the RPV dome) and a VT-2 visual examination for leakage. This test will subject the repair welds to a minimum pressure of 29 psi (3%) above normal operating versus the hydrostatic test pressure of 86 psi (8.6%) above normal operating pressure. In addition, each of these areas will be subject to a system leakage test performed at nominal operating pressure (1005 psig at the RPV dome) prior to startup following the current outage. The decrease in the margin of safety due to the reduction in test pressure is minimal and does not present an undue risk to the plant or the public.

<u>Licensee's Proposed Alternative Examination:</u> The licensee proposes that a leakage test of the reactor coolant system will be performed at the highest pressure which can be obtained while ensuring that the main steam safety valves will not be challenged (1034 psig at the RPV dome). This test pressure is based on the following considerations:

1. The lowest main steam relief valve has a setpoint value of 1105 psi and a setpoint tolerance of ± 1%;

2. The main steam safety valves are located 41 feet below the RPV dome;

3. A 20 psi test pressure range is necessary due to expected perturbations during the test;

4. A 20 psi margin is deemed essential to prevent the commencing of weeping through the main steam safety valve.

In addition, as required by the ASME Section XI, the reactor coolant system will receive a leakage test at full system operating pressure (1005 psig) prior to unit startup following each refueling outage. A hydrostatic pressure test of these repairs will be performed in conjunction with the reactor coolant system static test near the end of the current inspection interval.

<u>Staff Evaluation:</u> The staff determined that it would be impractical for the licensee to perform the Code required hydrostatic pressure test of the

repaired 1-inch Core Spray weld repair and two 2-inch socket welds. These welds are situated in their respective systems between the reactor pressure vessel (RPV) and the first isolation valve of the vessel and the locations provide no method to isolate the repaired areas of the RPV for the purpose of the hydrostatic test required by Article IWA-4000. The licensee's nondestructive examinations of the repaired areas, and its proposed alternative examination to perform a hydrostatic test at 1034 psig (measured at the dome) should provide an acceptable level of quality and safety for the specific components listed, and reasonable assurance that the structural integrity of the plant's systems, components, and supports will be maintained.

REQUEST FOR RELIEF SPT-6 : Request for relief from removing body-to-bonnet bolts from VALVE FCV-68-33 for performing the VT-3 examination.

CODE REQUIREMENT: (IWA-5250 CORRECTIVE MEASURES)

(a) The source of leakages detected during the conduct of a system pressure test shall be located and evaluated by the Owner for corrective measures as follows:

(1) ...

(2) 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 BASIS FOR RELIEF:

During conduct of the Code Class 1 leakage test following completion of Cycle 6 refueling outage, leakage was discovered at the bonnet-to-body bolting. These studs are 1-3/4 inch diameter, A540 Grade B23 material. The valve body is A351 Grade CF8 material.

BFN has been unsuccessful in removing the studs for the VT-3 visual examination. The studs were soaked in penetrating oil for twenty-four hours prior to attempting removal. A force of approximately 2000 foot-pounds was applied to several of the studs with no resulting movement. The valve manufacturer, Anchor Darling, recommended that a force of 2300 foot-pounds on the studs not be exceeded to ensure that galling of the studs does not occur. The normal torque value for the studs is 1351 foot-pounds (maximum) per Anchor Darling Vendor Manual.

Should galling of the stud to valve body occur due to additional force being exerted on the studs, it would be necessary to rethread the valve body. Rethreading the valve body would require removal of the valve bonnet. This valve cannot be isolated from the reactor vessel. Removal of the valve bonnet would require that the drywell head and the reactor vessel head be removed to allow plugging of the jet pumps. The performance of this work would require approximately three weeks.

The estimated exposure for this work is approximately 17 man-REM. This estimate is based on a crew of two men, a galling rate of 50%, a stud removal rate of two hours and rethreading rate of 14 hours per galled stud, and a radiation field of 40 mREM.

This hardship due to personnel radiation exposure and additional unit shutdown time is excessive in light of the small additional safety margin achieved by removal of the studs above the proposed alternative examination.

LICENSEE'S PROPOSED ALTERNATIVE EXAMINATION:

Approximately four inches of each of the 24 studs is visible between the bonnet flange and the valve body. Both ends of the studs are also visible. BFN proposes to perform a VT-1 visual examination of all 24 studs in place in accordance with examination Category B-G-2, Item No. B7.70 of Table IWB-2500-1 of Section XI of the ASME Boiler and Pressure Vessel Code.

STAFF EVALUATION

The recirculation valve FCV-68-33 is located in the recirculation system and cannot be isolated from the reactor vessel without encountering extreme measures. In order for the licensee to perform the Code requirement in a safe manner, the studs would have to be removed one at a time, examined, and replaced prior to removal of another stud. The licensee has attempted to remove six of the twenty-four studs without success after having soaked the studs in penetrating oil for twenty-four hours and applying torque values approaching the maximum values recommended by the valve manufacturer, Anchor Darling.

The flange leakage observed during the leakage test was stopped by torquing the bolts to higher values, up to 2000 ft-lbs. The staff's concern was that the bolts might have been over-torqued which could lead to failure of the joint during service. By telephone conference on May 19, 1993, the licensee provided information to many of the questions concerning the bolts and the reasons why there was not a concern for the valve flange joint leakage and structural integrity. The licensee performed ultrasonic examination of all 24 studs from both ends with no indications noted. The accessible surfaces of the studs were visually examined with no indications noted. Hardness tests were performed on each of the studs, with the values reported to be in the range of 20 HRC to 34 HRC. It is unlikely that stress corrosion cracking will occur in ASTM A540 C23 material with the hardness readings reported.

Based on the information provided by the licensee, the staff finds the Code requirements impractical to perform at the Browns Ferry Unit 2 facility. The alternative examinations performed provide adequate assurance that the structural integrity of the flanged joint will be maintained during service. However, the staff deems it necessary to impose the Code requirements at the next refueling outage.

3.0 CONCLUSION

Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee determined that conformance with certain Code requirements are impractical for its facility and submitted supporting information. The staff has reviewed the licensee's submittal and

has concluded that these are cases where relief can be granted as requested. Pursuant to 10 CFR 50.55a(g)(6)(i), the staff concluded that the requirements of the Code are impractical and relief may be granted for requests for relief SPT-4, SPT-5, and SPT-6. However, for request for relief SPT-6 the staff deems it necessary to impose the Code requirements at the next scheduled refueling outage. Furthermore, the nondestructive examinations performed on the repaired areas and proposed alternative examinations should provide an acceptable level of quality and safety, and reasonable assurance that the structural integrity of the plant's systems, components, and supports will be maintained. Such relief is authorized by law and will not endanger life, property, or the common defense and security, and is otherwise in the public interest. Relief has been granted giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Principal Contributors: T. McLellan and G. Johnson

Dated: May 21, 1993