February 12, 2001

Mr. J. A. Scalice
Chief Nuclear Officer
and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - RELIEF FROM CODE REQUIREMENTS FOR RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGER INSERVICE INSPECTION (TAC NOS. MA9898 AND MA9900)

Dear Mr. Scalice:

By letter dated June 29, 2000, as supplemented December 12, 2000, the Tennessee Valley Authority (TVA) submitted a request for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code inservice inspection (ISI) requirements for the Sequoyah Nuclear Plant, Units 1 and 2. The request was submitted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(5)(iii). The relief request is based upon design limitations that preclude full code examination of the nozzle inside radius sections of the residual heat removal system heat exchangers. TVA requested that the U.S. Nuclear Regulatory Commission (NRC) provide relief in accordance with the provisions of 10 CFR 50.55a(g)(6)(i).

The NRC staff has reviewed the information provided in the TVA letter. The staff's evaluation and conclusions are contained in the Enclosure. Based on the information provided in Relief Requests 1-ISI-15 and 2-ISI-15, the staff concludes that compliance with the Code requirements is impractical to meet, and reasonable assurance of the structural integrity of the subject components will be provided by the examinations that will be completed. Therefore, relief is hereby granted pursuant to 10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Sincerely,

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure: Safety Evaluation

cc w/encl: See next page

February 12, 2001

Mr. J. A. Scalice
Chief Nuclear Officer
and Executive Vice President
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, Tennessee 37402-2801

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - RELIEF FROM CODE REQUIREMENTS FOR RESIDUAL HEAT REMOVAL SYSTEM HEAT EXCHANGER INSERVICE INSPECTION (TAC NOS. MA9898 AND MA9900)

Dear Mr. Scalice:

By letter dated June 29, 2000, as supplemented December 12, 2000, the Tennessee Valley Authority (TVA) submitted a request for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code inservice inspection (ISI) requirements for the Sequoyah Nuclear Plant, Units 1 and 2. The request was submitted in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(5)(iii). The relief request is based upon design limitations that preclude full code examination of the nozzle inside radius sections of the residual heat removal system heat exchangers. TVA requested that the U.S. Nuclear Regulatory Commission (NRC) provide relief in accordance with the provisions of 10 CFR 50.55a(g)(6)(i).

The NRC staff has reviewed the information provided in the TVA letter. The staff's evaluation and conclusions are contained in the Enclosure. Based on the information provided in Relief Requests 1-ISI-15 and 2-ISI-15, the staff concludes that compliance with the Code requirements is impractical to meet, and reasonable assurance of the structural integrity of the subject components will be provided by the examinations that will be completed. Therefore, relief is hereby granted pursuant to 10 CFR 50.55a(g)(6)(i). The staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Sincerely,

/RA/

Richard P. Correia, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosure: Safety Evaluation

cc w/encl: See next page

DISTRIBUTION:
PUBLIC
PDII-2 (paper copy)
RidsNrrDlpmLpdii
RidsNrrDlpmLpdiiII
BClayton (paper copy)
RidsNrrPMRHernan (paper copy)
RidsOgcRp
GHill (2) (paper copy)
RidsAcrsAcnwMailCenter
SRosenberg (e-Mail)
PFredrickson, RII (RidsRgn2MailCenter)
ESullivan (RidsNrrDeEmcb)

DOCUMENT NAME: C:\RELA9900.wpd

OFFICE	PDII-2/PM	E	PDII-2/LA	E	EMCB/SC	OGC		PDII-2/SC	
NAME	RHernan		BClayton		ESullivan	KMBarber/"NLO"		RCorreia	
DATE	1/22/01		1/19/01		1/25/01	2/6/01		2/9/01	

OFFICIAL RECORD COPY

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF RELIEF REQUESTS FROM ASME SECTION XI REQUIREMENTS

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF NOS. 1-ISI-15 AND 2-ISI-15

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

TENNESSEE VALLEY AUTHORITY

DOCKET NUMBERS: 50-327 AND 50-328

1.0 INTRODUCTION

Inservice inspection (ISI) of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g), except where specific written relief has been granted by the U.S. Nuclear Regulatory Commission (NRC or the Commission) pursuant to 10 CFR 50.55a(g)(6)(i). It is stated in 10 CFR 50.55a(g)(6)(i) that the NRC will evaluate a licensee's determination that conformance with certain Code requirements is impractical for its facility and may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

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 (ISI) 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.

By letter dated June 29, 2000, as supplemented December 12, 2000, the Tennessee Valley Authority (TVA, the licensee), submitted Requests for Relief (Nos. 1-ISI-15 and 2-ISI-15) from the requirements of the ASME Code, Section XI, for the Sequoyah Nuclear Plant (SQN), Units 1 and 2. These relief requests are for the second 10-year ISI interval.

ENCLOSURE

2.0 EVALUATION

The NRC staff has reviewed the information concerning ISI program Requests for Relief Nos. 1-ISI-15 and 2-ISI-15, for SQN Units 1 and 2, respectively, which was submitted by TVA in their letter dated June 29, 2000, as supplemented by letter dated December 12, 2000. The Code of Record for the SQN Units 1 and 2, second 10-year ISI interval, which began December 16, 1995, is the 1989 Edition of Section XI of the ASME B&PV Code.

Request for Relief Nos. 1-ISI-15, 2-ISI-15 - Examination Category C-B, C2.22, Pressure Retaining Nozzle Welds in Vessels

Code Requirement: ASME Section XI, Table IWC-2500-1, Examination Category C-B, Item C2.22 requires 100% volumetric examination of pressure retaining nozzle welds in vessels. Specifically, Appendix I, Paragraph I-2200, states:

> Ultrasonic examination of vessel welds less than or equal to 2-inch thickness and all piping welds shall be conducted in accordance with Appendix III, as supplemented by this Appendix. Supplements identified in Table I-2000-1 shall be applied.

Appendix III, Supplement 4, “Austenitic and Dissimilar Metal Welds,” Paragraph (c) states:

> Qualification - In recognition of the difficulty in ultrasonic examination of the welds and materials in (a), it is recommended that the examiners and procedures be qualified using welded samples, and simulated or actual flaws, or both, located in positions where geometry may make them more difficult to detect (e.g., the counterbore or adjacent to the weld root). The purpose of the examination procedure qualification is to determine that the proposed examination technique is capable of detecting the specified flaws of interest and that its capabilities and limitations will be identified. Requirements for the qualification of examiners and procedures are in course of preparation.

Licensee’s Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required volumetric examination of the “A” and “B” Residual Heat Removal (RHR) Heat Exchanger nozzle inside radius sections.

Licensee’s Basis for Requesting Relief:

> The design configuration of the RHR heat exchanger nozzle and the shell and component support configuration prohibit an effective ultrasonic examination of the required volume for the nozzle inside radius section. [The actual nozzle configuration doesn’t have an indicated inside radius.]
>
> The design configuration of the subject nozzle-to-shell weld [and the fact that the material is stainless steel] preclude effective volumetric examination of the nozzle inside radius section. In order to examine the nozzle inside radius, the RHR heat exchanger would require extensive design modifications.

An in-depth investigation was initiated by TVA to determine the feasibility of performing an acceptable code volumetric examination. The investigation reviewed the nozzle type, weld placement and actual outside diameter weld profiles as well as ultrasonic measurements to verify inside diameter configuration. To assist in the evaluation of performing an acceptable code volumetric inner radius examination, computer modeling and studies were performed by Electric Power Research Institute (EPRI) personnel to determine if performing an ultrasonic examination of the inner radius region was feasible. The application was determined to be similar to a study performed by EPRI for the Ginna [Nuclear] Plant in the application of the regenerative heat exchanger. A copy of the EPRI report for this evaluation is shown in Attachment 2.[1]

EPRI concluded that the RG&E [Rochester Gas and Electric] Ginna Plant studies could be applied directly to the TVA RHR heat exchanger inner radius application. The RG&E Ginna Plant study concluded that a feasible examination procedure with optimized inspection angles with either shear or longitudinal wave mode could not be developed which would detect ultrasonic responses from a 30 percent through-wall notch.

The RG&E Ginna Plant study utilized a full scale mockup with four electro discharge machine notches located in the inner radius region. Computer modeling indicted that several different transducers were required to interrogate the inner radius at the correct angle. Examinations from the shell surface proved to be greatly affected by attenuation and scattering from the nozzle-to-shell weld and [stainless steel] material characteristics. Examination from the boss region of the nozzle proved that detection of the 10 percent notches was not achievable. Notches were increased from 10 to 30 percent through-wall and the detection was still not achievable. The experiments conclude that because of the sound beam attenuation, reflections from even a 30 percent through-wall notch do not return to the transducer and provide an adequate detection response.

To achieve the desired sound beam orientation in the nozzle inner radius region, metal paths can get extremely long. Long metal paths combined with the poor signal-to-noise ratios inherent to anisotropic [stainless steel] materials make manual detection a difficult challenge. The TVA RHR heat exchanger configuration revealed that metal paths were longer than those experienced in the RG&E Ginna Plant studies; therefore, results are expected to be significantly reduced due to increased attenuation associated with the longer metal paths.

Radiographic examination from the outside surface as an alternative volumetric examination method was determined to be impractical due to the component wall thickness and the configuration of a heat exchanger divider plate inside the component head area affecting radiographic quality. Performing radiographic examination from the inside surface of the heat exchanger would require placing

---

[1]Attachments, drawings, photographs, and sketch's submitted by the licensee are not included in this Safety Evaluation.

a radiographic source near the center of the head. This would require extensive modifications in order to gain access to the inside for source placement and disassembly of the heat exchanger. A long exposure time would be required because of the thickness and obtaining the required sensitivity would be improbable due to the geometric configuration. Extensive decontamination and personnel protection from contamination would be required. Personnel would be required to work extended hours in a face mask to reduce exposure to internal contamination. Thus, additional radiography and/or ultrasonic examinations from the inner surface of the nozzle to obtain any coverage are impractical.

However, the scheduled surface and ultrasonic examinations performed on the accessible areas (to the maximum extent practical) of the scheduled nozzle-to-shell weld will provide reasonable assurance of the structural integrity for the general area of the nozzle and shell assembly. In addition, the Code required pressure test VT-2 examinations will provide an additional measure of assurance.

Justification for TVA proposed request for relief is summarized as follows:

- Mechanical limitations (support pads, limited scan area from the boss side at the 0 and 180 degree locations, and close proximity to the vessel-to-flange weld, etc.) cause obtaining the required code coverage impossible.

- A feasible examination procedure with optimized examination angles with either shear or longitudinal wave mode could not be developed to detect notches in a mockup with a flaw 30 percent through-wall depth.

- Long metal paths needed to access the inner radius region combined with poor signal-to-noise ratios inherent to anisotropic materials, make manual detection a difficult challenge.

- Radiography would require taking the component out of service and significant modification to access the inner diameter for film placement.

- Additional ultrasonic techniques and/or surface methods to examine the inner radius would require access to the inner diameter which would require the component be taken out of service and significant modifications performed to access the inner diameter.

- A surface and ultrasonic examination on the accessible areas of the scheduled nozzle-to-vessel welds will provide sufficient information to judge the overall integrity of the nozzle in conjunction with performing the code required pressure test VT-2 examinations.

Therefore, pursuant to 10 CFR 50.55a(g)(5)(iii), it is requested that relief be granted, in accordance with 10 CFR 50.55a(g)(6)(i).

<u>Licensee's Proposed Alternative Examination</u> (as stated):

> In lieu of the code required 100 percent ultrasonic examination of the nozzle inside radius section, the surface and best effort ultrasonic examination performed during the scheduled nozzle-to-vessel weld examinations and the scheduled pressure test visual examination will be used to provide sufficient information to assess the overall integrity of the nozzle.

<u>Staff Evaluation</u>: The Code requires 100% surface and volumetric examination of Class 2 pressure vessel nozzle inside radius sections. However, sketches, photographs, and examination reports provided by the licensee show that complete volumetric examination of the subject RHR heat exchanger nozzle inside radius sections is limited due to the design configuration. The base material and nozzle-to-shell weld material, both stainless steel, and the nozzle geometry complicate inspection of the inner-radius region. The stainless steel base and weld materials cause increased attenuation of the ultrasonic wave, affecting the overall signal-to-noise ratio. The negative effects of a material with high attenuation can, in many cases, be overcome by optimizing the beam angle to be more normal to the flaw. However, the nozzle geometry, in this case, restricts optimization in that it does not allow introduction of an ultrasonic beam that is oriented such that detection is possible. In some nozzle geometries, the misorientation angle (skew at the flaw) can be reduced significantly by scanning on the blend radius. But the radius of the SQN RHR nozzle is small and, in most cases, irregular. From a demonstration, TVA determined that the longitudinal wave mode could direct a sound beam to most of the inner radius but was unable to detect ultrasonic responses from a 30-percent through-wall notch. Therefore, the Code examination requirements are impractical for this weld. To meet the Code requirements, the subject component would require significant engineering redesign and modification to allow access to the subject areas. Imposition of the Code requirements would result in a considerable burden on the licensee.

TVA proposes, as an alternative inspection, to perform the surface and best effort ultrasonic examination during the scheduled nozzle-to-vessel weld examinations and the scheduled pressure test visual examination will be used to provide sufficient information to assess the overall integrity of the nozzle. Because of the close proximity of the nozzle-to-vessel weld to the nozzle inside radius (2-inches or less), a degrading mechanism affecting the nozzle inside radius should also affect the nozzle-to-vessel weld. Therefore, based on the volume examined and the Code-required surface examinations performed, it is concluded that significant patterns of degradation, if present, would be detected and reasonable assurance of the structural integrity of the pressure-retaining nozzle weld would be provided.

Based on the impracticality of meeting the Code examination requirements for the subject weld, and the reasonable assurance of structural integrity provided by the examinations that will be completed, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i). The NRC staff has determined that granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property, or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

Principal Contributor: Ronald W. Hernan, NRR

Date: February 12, 2001

Mr. J. A. Scalice
Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

cc:
Mr. Karl W. Singer, Senior Vice President
Nuclear Operations
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Jack A. Bailey
Vice President
Engineering & Technical Services
Tennessee Valley Authority
6A Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Richard T. Purcell
Site Vice President
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

General Counsel
Tennessee Valley Authority
ET 10H
400 West Summit Hill Drive
Knoxville, TN 37902

Mr. Robert J. Adney, General Manager
Nuclear Assurance
Tennessee Valley Authority
5M Lookout Place
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Mark J. Burzynski, Manager
Nuclear Licensing
Tennessee Valley Authority
4X Blue Ridge
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Pedro Salas, Manager
Licensing and Industry Affairs
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. D. L. Koehl, Plant Manager
Sequoyah Nuclear Plant
Tennessee Valley Authority
P.O. Box 2000
Soddy Daisy, TN 37379

Mr. Russell A. Gibbs
Senior Resident Inspector
Sequoyah Nuclear Plant
U.S. Nuclear Regulatory Commission
2600 Igou Ferry Road
Soddy Daisy, TN 37379

Mr. Lawrence E. Nanney, Director
Division of Radiological Health
Dept. of Environment & Conservation
Third Floor, L and C Annex
401 Church Street
Nashville, TN 37243-1532

County Executive
Hamilton County Courthouse
Chattanooga, TN 37402-2801

Ms. Ann Harris
305 Pickel Road
Ten Mile, TN 37880