Tennessee Valley Authority, 1101 Market Street, Chattanooga, Tennessee 37402

June 26, 2013

10 CFR 50.4
10 CFR 50.55a

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

Sequoyah Nuclear Plant, Units 1 and 2
Facility Operating License Nos. DPR-77 and DPR-79
NRC Docket Nos. 50-327 and 50-328

Subject: **American Society of Mechanical Engineers Request for Alternative 11-SPT-1**

In accordance with Title 10 of the Code of Federal Regulations (10 CFR), Part 50.55a, "Codes and Standards," paragraph (a)(3)(ii), Tennessee Valley Authority (TVA) proposes an alternative to the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," as applicable to Sequoyah Nuclear Plant (SQN), Units 1 and 2. The Code of Record for the current third 10-year interval for SQN Units 1 and 2 is the ASME Section XI B&PV Code, 2001 Edition with Addenda through 2003.

TVA is submitting a Request for Alternative (RFA) 11-SPT-1 for Nuclear Regulatory Commission (NRC) approval of a proposed alternative to the requirement of ASME Code, Section XI, Subparagraph IWB-5222(b) to extend the pressure boundary to all Class 1 pressure retaining components during the system leakage test conducted at or near the end of the inspection interval. The details of RFA 11-SPT-1 are contained in the enclosure to this letter.

TVA's proposed alternative to perform the examination of selected Class 1 piping and valves at plant conditions other than those required by the ASME Code is considered to provide an acceptable level of quality and safety. The proposed alternative is requested on the basis that hardship and unusual difficulty exists in establishing a pressurized system configuration that will subject all Class 1 components to Reactor Coolant System operating pressure.

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TVA proposes to apply the alternative examination methodology of RFA 11-SPT-1 to the piping and valve examinations to be performed on SQN Units 1 and 2 at or near the end of the third 10-year Inservice Inspection (ISI) Program interval. TVA requests approval of the RFA by July 1, 2014. Both SQN units are currently in the third period of their third ISI Program interval, which extends from June 1, 2006 to April 30, 2016.

This RFA is similar to previous alternatives granted for Constellation Energy's Calvert Cliffs Nuclear Power Plant, by letter dated February 12, 2009 [ADAMS Accession No. ML090280282], and Progress Energy's Shearon Harris Nuclear Power Plant, by letter dated April 17, 2006 [ADAMS Accession No. ML060870387].

There are no regulatory commitments associated with this submittal. If you have any questions about this request, please contact Clyde Mackaman at (423) 751-2834.

Respectfully,

Beth A. Wetzel, for

J. W. Shea
Vice President, Nuclear Licensing

Enclosure: Request for Alternative 11-SPT-1

cc (Enclosure):

NRC Regional Administrator – Region II
NRC Senior Resident Inspector – Sequoyah Nuclear Plant

ENCLOSURE

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT

UNITS 1 AND 2

AMERICAN SOCIETY OF MECHANICAL ENGINEERS CODE

REQUEST FOR ALTERNATIVE 11-SPT-1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2
THIRD 10-YEAR INTERVAL
REQUEST FOR ALTERNATIVE 11-SPT-1

I. ASME Code Components Affected

American Society of Mechanical Engineers (ASME) Code Class 1 pressure retaining components that are beyond the first normally closed valve to the second boundary component. Specifically, this request relates to ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.10.

II. ASME Code Edition and Addenda

2001 Edition through the 2003 Addenda
Current Interval: Third (Start Date: June 1, 2006; End Date: April 30, 2016)
Submittal Date: April 21, 2006 (ADAMS Accession No. ML061210105)

III. Applicable Code Requirement

Code Requirement:
ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.10 requires that a system leakage test be conducted prior to startup following each refueling outage in accordance with the requirements of IWB-5220. IWB-5222(b) requires "The pressure retaining boundary during the system leakage test conducted at or near the end of each [10-year] inspection interval shall extend to all Class 1 components within the system boundary."

Code Requirement for Which Alternative is Requested:
An alternative is requested from ASME Code, Section XI, Subparagraph IWB-5222(b). The ASME Code requires the pressure boundary to be extended to all Class 1 pressure retaining components during the system leakage test conducted at or near the end of the inspection interval.

IV. Reason for Request or Impracticality of Compliance

Pursuant to 10 CFR 50.55a(a)(3)(ii), Sequoyah Nuclear Plant (SQN) requests approval to perform the examination of select Class 1 piping and valves at plant conditions other than those required by the ASME Code. An alternative is requested in accordance with 10 CFR 50.55a(a)(3)(ii) on the basis that hardship and unusual difficulty exists in establishing a system configuration that would subject all Class 1 pressure boundary components to Reactor Coolant System (RCS) pressure during the system pressure test without a compensating increase in the level of quality and safety. Extending the pressure retaining boundary during the system pressure test to all Class 1 pressure retaining components beyond the first normally closed valve would require a number of off-normal temporary system configurations that includes the use of temporary piping installations in order to achieve test pressures at system segments beyond the first isolation valve. The specific locations covered by this request include portions of the Safety Injection System, Chemical Volume and Control System (CVCS), Residual Heat Removal (RHR) system, and RCS, as described in the following tables. Most of these locations are in close proximity to RCS loop piping and require personnel entry into radiation areas within containment. The tables include estimated radiation levels near the respective valves. In general, the material used to fabricate the piping and fittings in these locations is heavy wall stainless steel (e.g., Schedule 160 or Schedule 140, ASME SA376 TP304 or TP316 for piping, ASME SA182 F316 or F304 for fittings).

Table 1: Safety Injection System

Piping Segment Description	Between Valve	And Valve
CLA	SQN-1/2-VLV-63-560 (Loop 1 Cold Leg Injection Check Valve) ~15 mREM/hour	SQN-1/2-VLV-63-622 (Accumulator 1 Outboard Check Valve) ~3 mREM/hour
		SQN-1/2-VLV-63-633 (RHR Pump Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-551 (Loop 1 Cold Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-117 (Loop 1 Cold Leg Leak Test Check Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-586 (Loop 1 Cold Leg Inboard Boron Injection Check Valve) ~50 mREM/hour	SQN-1/2-VLV-63-581 (Outboard Boron Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-24 (Accumulator Fill Line Isolation Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-641 (Inboard RHR Heat Exchanger Discharge Check Valve) ~15 mREM/hour	SQN-1/2-VLV-63-640 (Outboard RHR Heat Exchanger Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-543 (Loop 1 Hot Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-163 (Loop 1 Hot Leg Test Line) ~2 mREM/hour
CLA	SQN-1/2-VLV-63-561 (Loop 2 Cold Leg Injection Check Valve) ~20 mREM/hour	SQN-1/2-VLV-63-623 (Accumulator 2 Outboard Check Valve) ~4 mREM/hour
		SQN-1/2-VLV-63-632 (RHR Pump Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-553 (Loop 2 Cold Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-97 (Loop 2 Cold Leg Leak Test Check Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-587 (Loop 2 Cold Leg Inboard Boron Injection Check Valve) ~50 mREM/hour	SQN-1/2-VLV-63-581 (Outboard Boron Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-24 (Accumulator Fill Line Isolation Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-559 (Loop 2 Hot Leg Inboard Injection Check Valve) ~10 mREM/hour	SQN-1/2-VLV-63-547 (Loop 2 Hot Leg Outboard Injection Check Valve) ~1 mREM/hour
		SQN-1/2-FCV-63-165 (Loop 2 Hot Leg Test Line) ~1 mREM/hour
CLA	SQN-1/2-VLV-63-562 (Loop 3 Cold Leg Injection Check Valve) ~15 mREM/hour	SQN-1/2-VLV-63-624 (Accumulator 3 Outboard Check Valve) ~3 mREM/hour
		SQN-1/2-VLV-63-634 (RHR Pump Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-555 (Loop 3 Cold Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-79 (Loop 3 Cold Leg Leak Test Check Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-588 (Loop 3 Cold Leg Inboard Boron Injection Check Valve) ~50 mREM/hour	SQN-1/2-VLV-63-581 (Outboard Boron Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-24 (Accumulator Fill Line Isolation Valve) ~2 mREM/hour

Piping Segment Description	Between Valve	And Valve
ECCS	SQN-1/2-VLV-63-644 (Inboard RHR Heat Exchanger Discharge Check Valve) ~10 mREM/hour	SQN-1/2-VLV-63-545 (Loop 3 Hot Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-643 (Outboard RHR Heat Exchanger Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-164 (Loop 3 Hot Leg Test Line) ~2 mREM/hour
CLA	SQN-1/2-VLV-63-563 (Loop 4 Cold Leg Injection Check Valve) ~15 mREM/hour	SQN-1/2-VLV-63-625 (Accumulator 4 Outboard Check Valve) ~3 mREM/hour
		SQN-1/2-VLV-63-635 (RHR Pump Discharge Check Valve) ~2 mREM/hour
		SQN-1/2-VLV-63-557 (Loop 4 Cold Leg Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-69 (Loop 4 Cold Leg Leak Test Check Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-589 (Loop 4 Cold Leg Inboard Boron Injection Check Valve) ~50 mREM/hour	SQN-1/2-VLV-63-581 (Outboard Boron Injection Check Valve) ~2 mREM/hour
		SQN-1/2-FCV-63-24 (Accumulator Fill Line Isolation Valve) ~2 mREM/hour
ECCS	SQN-1/2-VLV-63-558 (Loop 4 Hot Leg Inboard Injection Check Valve) ~10 mREM/hour	SQN-1/2-VLV-63-549 (Loop 4 Hot Leg Outboard Injection Check Valve) ~1 mREM/hour
		SQN-1/2-FCV-63-166 (Loop 4 Hot Leg Test Line) ~1 mREM/hour

Table 2: Chemical Volume and Control System

Piping Segment Description	Between Valve	And Valve
RCP Seal Injection	SQN-1/2-VLV-62-576 (Loop 1 Inboard Seal Water Injection Check Valve) ~3 mREM/hour	SQN-1/2-VLV-62-560 (Loop 1 Outboard Seal Water Injection Check Valve) ~2 mREM/hour
RCP Seal Injection	SQN-1/2-VLV-62-577 (Loop 2 Inboard Seal Water Injection Check Valve) ~4 mREM/hour	SQN-1/2-VLV-62-561 (Loop 2 Outboard Seal Water Injection Check Valve) ~2 mREM/hour
RCP Seal Injection	SQN-1/2-VLV-62-578 (Loop 3 Inboard Seal Water Injection Check Valve) ~3 mREM/hour	SQN-1/2-VLV-62-563 (Loop 3 Outboard Seal Water Injection Check Valve) ~2 mREM/hour
RCP Seal Injection	SQN-1/2-VLV-62-579 (Loop 4 Inboard Seal Water Injection Check Valve) ~3 mREM/hour	SQN-1/2-VLV-62-562 (Loop 4 Outboard Seal Water Injection Check Valve) ~2 mREM/hour
Normal Charging	SQN-1/2-VLV-62-660 (Loop 4 Charging Inboard Check Valve) ~50 mREM/hour	SQN-1/2-VLV-62-717 (Loop 4 Charging Outboard Check Valve) ~3 mREM/hour
Alternate Charging	SQN-1/2-VLV-62-659 (Loop 1 Charging Inboard Check Valve) ~50 mREM/hour	SQN-1/2-VLV-62-716 (Loop 1 Charging Inboard Check Valve) ~3 mREM/hour
Pressurizer Spray	SQN-1/2-VLV-62-661 (RCS Spray Check Valve) ~15 mREM/hour	SQN-1/2-FCV-62-84 (Charging Flow to RCS Check Valve) ~4 mREM/hour

Table 3: Residual Heat Removal System

Piping Segment Description	Between Valve	And Valve
RHR	SQN-1/2-FCV-74-1 (RHR Supply Inboard Isolation Valve) ~20 mREM/hour	SQN-1/2-FCV-74-2 (RHR Supply Outboard Isolation Valve) ~15 mREM/hour

Table 4: Reactor Coolant System

Piping Segment Description	Between Valve	And Valve
RCS-Loop 1 Drain	SQN-1/2-VLV-68-549 (Loop 1 Inboard Drain Valve) ~12 mREM/hour	SQN-1/2-VLV-68-550 (Loop 1 Outboard Drain Valve) ~12 mREM/hour
RCS-Loop 2 Drain	SQN-1/2-VLV-68-553 (Loop 2 Inboard Drain Valve) ~12 mREM/hour	SQN-1/2-VLV-68-554 (Loop 2 Outboard Drain Valve) ~12 mREM/hour
RCS-Loop 3 Drain	SQN-1/2-VLV-68-581 (Loop 3 Inboard Drain Valve) ~12 mREM/hour	SQN-1/2-VLV-68-582 (Loop 3 Outboard Drain Valve) ~12 mREM/hour
RCS-Loop 4 Drain	SQN-1/2-VLV-68-557 (Loop 4 Inboard Drain Valve) ~12 mREM/hour	SQN-1/2-VLV-68-558 (Loop 4 Outboard Drain Valve) ~12 mREM/hour
Reactor Vessel Vent	SQN-1/2-VLV-68-597 (Reactor Vessel Vent Valve) ~85 mREM/hour	SQN-1/2-FLG-68-597 (Reactor Vessel Vent Blind Flange) ~85 mREM/hour

(Emergency Core Cooling System - ECCS, Cold Leg Accumulator - CLA, Residual Heat Removal - RHR, Reactor Coolant Pump - RCP, Flow Control Valve - FCV, Valve - VLV, Flange - FLG, Roentgen Equivalent Man - REM, and one thousandth of one REM - mREM)

V. Burden Caused by Compliance

While the RCS is being brought to normal operating temperature and pressure (approximately 2235 pounds per square inch gauge (psig) and 547 degrees Fahrenheit (°F)) in accordance with the plant Technical Specifications (TSs), the piping segments listed in the above tables become isolated from the RCS either by the check valves going to their closed position or the FCVs being closed by plant procedure. This causes the piping segments to be at pressures and temperatures less than full RCS pressure and temperature. An exception is the CVCS seal injection piping segments, which provide seal injection to the RCP seals. This piping is normally in service with the check valves open. Thus, the piping segments are at a system pressure equal to or greater than RCS pressure but are not at a corresponding RCS temperature.

The plant design configuration complies with the RCS boundary requirements for double isolation but cannot satisfy the code test requirement for nominal operating pressure associated with 100 percent rated reactor power (i.e., full RCS pressure and temperature between the piping segments). While manual manipulation of valves may be possible to achieve full RCS test pressure as required by the ASME Code, this would defeat the double isolation feature. Use of single valve isolation is a significant personnel safety hazard, may result in over-pressurization of downstream piping, and could damage permanent plant equipment. The use of temporary hoses to connect the RCS to the piping segment volumes in order to raise

them to full test pressure and temperature constitutes a personnel safety hazard and could adversely affect plant safety. Temporary hoses are not qualified to meet all aspects of the plant design, (e.g., pressure, temperature, ASME Code, seismic, dead load). The failure of unqualified temporary hoses during full pressure testing could result in personnel injury, and the loss of the reactor coolant pressure boundary and reactor coolant inventory. In addition, establishment of and restoration from such temporary configurations would take a considerable amount of time, result in unwarranted increase in worker radiation exposure, contaminate test equipment, and would contribute to delaying normal plant startup following the refueling outage. It is estimated that extending the boundary to all Class 1 components would result in total accrual of ~6.7 REM. This estimate is based on the radiation levels at the associated valves presented in Tables 1 through 4, and accounts for scaffold erection, potential insulation removal, valve manipulations, leak examinations, re-installation of insulation, and scaffold removal. Estimates for each respective piping system in person-REM are given below in the corresponding discussions of the affected piping segments. The only alternative to the use of temporary hoses would be to modify the plant configuration by installation of qualified piping to allow the subject piping segments to be connected directly to the RCS. This is considered impractical and cost prohibitive because of the limited use and additional administrative controls that would be needed.

CLA Piping
The piping segments associated with the CLAs will reach approximately 650 psig as a result of the CLA pressure required by TS Limiting Condition for Operation (LCO) 3.5.1.1 when the RCS is at 100 percent rated reactor power. These segments cannot be raised to full RCS pressure and temperature because this would require isolation of the CLAs during plant operation. Isolation of the CLAs would be required to prevent overpressurization of the accumulator tank above the relief valve setpoint of 700 psig, and would require entry into the TS LCO 3.5.1.1 action statement to restore the inoperable accumulator to OPERABLE status within 24 hours or be in HOT STANDBY within the next six hours. Therefore, failure of the lone isolation valve or leakage past the valve during the test, and compliance with the TS requirements could ultimately result in an unplanned reactor shutdown. It is estimated that extending the boundary as required by the ASME Code for testing these piping segments would require two people working four hours to erect scaffolding, perform the leak inspection, and remove scaffolding. In addition, it would require one person working four hours to isolate the accumulator tanks and realign the valves following completion of the testing. In total, these piping segments would result in accrual of ~0.4 person-REM (~1.2 REM total) of additional radiological dose.

Seal Injection, Charging, and Pressurizer Spray
The piping segments associated with RCP seal injection, the alternate and normal charging paths, and the pressurizer spray path will be pressurized to approximately 2400 psig (the charging pump discharge pressure) and approximately 110°F when the RCS is at normal operating pressure and temperature. These segments are at a pressure higher than RCS pressure but below the temperature of the RCS. The temperature is maintained below RCS temperature to protect the RCP seals from premature failure, preclude failure of the lower radial bearing, and prevent warping of the RCP shaft. At full power, the temperature of the RCP seal injection piping remains at approximately 110°F. If the temperature exceeds 179°F, the main control room receives an alarm for high temperature. Above 230°F, operating procedures require the pump to be tripped. In addition, because the design temperature for the volume control tank (seal flow suction source) is 250°F, increasing the temperature to nominal RCS temperature would violate the design criteria for the tank. Therefore, applying the ASME Code required temperature to these piping segments would adversely affect plant safety by requiring

actions and procedural provisions intended to safely operate plant equipment and prevent equipment damage to be bypassed during the conduct of the testing.

RHR
The RHR piping segment is in service for heat removal from the RCS. When shutting down the plant, plant procedures place the RHR system in service when the RCS temperature is less than 235°F and the RCS pressure is less than 350 psig. When starting up the plant, RHR remains in service until the RCS temperature is greater than 200°F and RCS pressure is between 325 and 350 psig. The SQN Updated Final Safety Analysis Report (UFSAR) specified safety system pressure interlocks prevent the RHR supply valves (SQN-1/2-FCV-74-1 and -2) from being opened until RCS pressure is less than 380 psig. In addition, the circuit breakers to the valves are opened and administratively locked prior to raising RCS temperature above 350°F. These actions have been put in place to prevent overpressurization of the downstream RHR piping, which has design ratings of 600 psig and 400°F, and to prevent loss of RCS inventory to the Refueling Water Storage Tank in the event that the valves do not isolate. Pressurizing the piping between these valves to nominal RCS operating pressure and temperature by opening the SQN-1/2-FCV-74-1 or installing test jumpers would require defeating a safety feature and the redundancy afforded by double isolation valves, thus reducing plant safety. It is estimated that extending the boundary as required by the ASME Code for this piping segment would require two people working eight hours to install temporary test hoses (or open the inboard valve), perform the leak inspection, and return the configuration to normal. In total, this piping segment would result in accrual of ~0.28 person-REM (0.56 REM total) of additional radiological dose.

RCS Vent and Drain Piping
The piping segments for the Class 1 vents and drains in the RCS are equipped with inboard isolation valves and outboard isolation valves (or blind flanges) as described in Table 4. The valves are maintained in the closed position during normal plant operation, and the piping downstream of the inboard valve is not normally pressurized. In order to pressurize those piping segments as required by ASME Code IWB-5222(b), it would be necessary to manually open the inboard valves to pressurize the piping and connections. Pressurization by this method defeats the double isolation and reduces the margin of personnel safety for those performing the test, and could result in overpressurization of the Reactor Coolant Drain Tank if the lone isolation valve were to leak during the test. In addition, there is currently no available method to depressurize the downstream piping following test completion; thus, placing the plant in an abnormal condition. It is estimated that extending the boundary as required by the ASME Code for these piping segments would require five people working four hours to open the manual valves, perform the leak inspection, and restore the valves to their normal configuration. In total, these piping segments would result in accrual of ~0.10 person-REM (0.52 REM total) of additional radiological dose.

ECCS
The remaining piping segments are associated with ECCS injection and are at static head pressure and temperature when the RCS is at normal operating pressure and temperature, and are separated from RCS conditions by self-operating check valves. The only method of achieving full RCS conditions in the Class 1 piping between the primary and secondary isolation check valves would be by: (1) installing temporary hoses around the primary check valve which would defeat the double isolation requirement for the reactor coolant pressure boundary or (2) making permanent modifications to the plant and installing full qualified piping solely for the purpose of conducting the infrequently performed ASME Code pressure test. It is estimated that extending the boundary for the ASME Code testing of these piping segments would require

four people working four hours to erect scaffold, install temporary test hoses, perform the leak inspection, disassemble scaffolding, and return the configuration to normal. In total, these piping segments would result in accrual of ~1.1 person-REM (4.4 REM total) of additional radiological dose.

Based on the personnel and plant safety concerns, as well as the radiological dose considerations, extension of the boundary subjected to RCS pressure to include all Class 1 pressure retaining components within the system boundary as required by ASME Code IWB-5222(b) represents a hardship and unusual difficulty.

VI. Proposed Alternative and Basis for Use

Proposed Alternative:
SQN will conduct the required end of interval system pressure tests as prescribed by Table IWB-2500-1, Examination Category B-P, using the boundary conditions and full examination coverage of IWB-5222(a). In addition, those Class 1 pressure boundary portions of the systems described above that are statically pressurized during normal operation will be visually examined during the RCS system leakage test and specific piping segments will be examined as described below.

CLA Piping
For the piping segments associated with the CLAs, the test pressure and temperature will be consistent with the CLA tank pressure (~650 psig) and temperature (~130°F). The pressure in the segment will be at least the operating pressure of the accumulators (~650 psig). No pressure boundary leakage will be allowed for the test to be successful. These piping segments would not function as designed if they are pressurized above ~700 psig because the accumulator tank relief pressure is ~700 psig, and the accumulator is only able to inject at pressures less than ~700 psig. Because these piping segments can only perform their intended function if the pressure is less than ~700 psig, pressurizing the piping segments to the higher nominal RCS operating pressure solely for testing does not serve to provide any increase in safety. The proposed alternative testing at accumulator operating pressure will provide the requisite assurance that the combined first and second isolation devices are effective in maintaining the reactor coolant pressure boundary at normal operating temperature and pressure.

Seal Injection, Charging, and Pressurizer Spray
For the piping segments associated with the RCP seal injection, alternate charging, charging, and pressurizer spray, the test pressure and temperature will be consistent with their normal segment pressure (~2400 psig) and temperature (~110°F). No pressure boundary leakage will be allowed for the test to be successful. While the test temperature is not elevated to RCS operating temperature, the pressure is actually above RCS pressure to ensure adequate flow is supplied to the RCP seals. Any pressure boundary flaws in this piping would be exposed with the high pressure injection flow. As such, there is no significant benefit from additional pressure testing at elevated RCS temperature conditions.

RHR
For the RHR piping segment, the test pressure and temperature will be consistent with the plant conditions during the time when RHR is in service for heat removal (less than 350 psig and less than 235°F). No pressure boundary leakage will be allowed for the test to be successful. As

such, the proposed alternative test will verify that the combined first and second isolation valves are effective in maintaining the reactor coolant pressure boundary at normal operating temperature and pressure. This test will provide assurance that the RHR piping segment does not experience leakage within its normal operating parameters. The additional testing at the elevated RCS pressure and temperature conditions as required by the ASME Code would compromise plant safety by having to bypass the safety related isolation and interlock features of the system in order to perform the testing.

RCS Vent and Drain Piping
For the Class 1 vent and drain piping segments described in Table 4, the test will be conducted as prescribed by Table IWB-2500-1, Examination Category B-P, only the boundaries of IWB-5222(a) will apply. No pressure boundary leakage will be allowed for the test to be successful. This test methodology is consistent with that described in ASME Code Case N-798, which is currently under NRC review for acceptance. This test will provide assurance that the combined first and second isolation valves (or blind flanges) are effective in maintaining the reactor coolant pressure boundary at normal operating temperature and pressure. There is no commensurate safety benefit to be gained by extending the test boundary, in that doing so would involve bypassing the double isolation safety boundary and increase the potential for system overpressurization, and could adversely affect plant and personnel safety.

ECCS
The ECCS piping segments are for providing injection water to the hot and cold legs to mitigate accidents involving loss of RCS inventory. The primary and secondary isolation devices are check valves oriented to flow into the RCS. These piping segments provide the design-required double isolation barrier for the reactor coolant pressure boundary. Leakage testing of these piping segments at nominal operating pressure in MODE 3 would require a modification to allow pressurizing to the normal operating RCS temperature and pressure conditions. The proposed alternative leakage test consists of aligning the safety injection test header and pressurizing the piping using a safety injection pump. No pressure boundary leakage will be allowed for the test to be successful. This test method is consistent with ASME Code Case N-800, which is currently under NRC review for acceptance. This test will provide a pressure of approximately 1500 psig, and will be at ambient temperature conditions since suction to the safety injection pump would be from the refueling water storage tank, which is open to atmospheric conditions. These piping segments are not normally operated at RCS pressure and temperature and, like the CLA piping, would not perform their intended safety function if pressurized to RCS pressure. Therefore, performing a pressure test at nominal RCS pressure and temperature does not serve a specific purpose or provide an increase in safety. The proposed alternative test will provide assurance that the isolation valves can maintain the reactor coolant pressure boundary as they are designed.

Basis for Use:
10 CFR 50.55a(a)(3) states:

"Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section, or portions thereof, may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. Any proposed alternatives must be submitted and authorized prior to implementation. The applicant or licensee shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety; or
(ii) Compliance with specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

The objective of the required extended pressure boundary conditions is to detect evidence of leakage, and thereby verify the integrity of the RCS pressure boundary beyond the first isolation valve. As discussed above, in order to meet the ASME Code requirement of IWB-5222(b), the establishment of and the return from the required temporary configurations would involve considerable time to accomplish, result in unwarranted increase in worker radiation exposure, would expose personnel to industrial safety risks due to use of a single isolation valve, potentially overpressurize downstream piping in the event of valve leakage, and would delay normal plant startup following the refueling operation. In addition, these temporary configurations could require bypassing UFSAR specified safety system protective features and interlocks. As a result, Tennessee Valley Authority (TVA) has concluded that compliance with the ASME Code requirement constitutes a hardship without a compensating increase in the level of quality and safety. The proposed alternative testing methods would provide a level of assurance that the RCS pressure boundary is maintaining structural integrity at elevated pressure. Furthermore, the majority of welds encompassed by the boundaries described in Tables 1 through 4 fall into the population of welds that would be examined by the Risk Informed Inservice Inspection (RI-ISI) program. The RI-ISI program includes requirements for examinations that verify structural integrity of ASME Code Class 1 and 2 piping welds. During the 2nd and 3rd intervals at SQN, there have been 378 examinations performed on welds in the RI-ISI program, with no failures.

Therefore, for the reasons cited above, TVA requests authorization to perform the requested alternative to the ASME Code requirement pursuant to 10 CFR 50.55a(a)(3)(ii). TVA considers the proposed alternative an acceptable method of testing to satisfy the ASME Code requirements, including reasonable assurance of piping structural integrity.

VII. Duration of the Proposed Alternative

Alternative is requested for the Third 10-Year Inspection Interval for SQN Unit Nos. 1 and 2.

VIII. Precedents

Similar requests for alternatives have been approved for:

Calvert Cliffs Nuclear Power Plant, Units 1 and 2, Third Inspection Interval Relief Request PT-3-01, authorized per Safety Evaluation Report (SER) dated February 12, 2009. (ADAMS Accession No. ML090280282)

Shearon Harris Nuclear Power Plant, Unit 1, Second Inspection Interval Relief Request 2R1-015, authorized per SER dated April 17, 2006. (ADAMS Accession No. ML060870387)

Cooper Nuclear Station, Fourth Inspection Interval Relief Request PR-11, authorized per SER dated October 2, 2006. (ADAMS Accession No. ML062260220)

Millstone Power Station, Unit 2, Third Inspection Interval Relief Request RR-04-04, authorized per SER dated July 27, 2011. (ADAMS Accession No. ML111881029)

IX. References

TS 3.4.6.3, Reactor Coolant System Pressure Isolation Valve Leakage
TS 3.5.1, Accumulators
TS 3.5.2, ECCS Operating
TS 3.5.3, ECCS Shutdown
TS 3.6.3, Containment Isolation Valves
UFSAR Drawing 5.1-1, Flow Diagram Reactor Coolant System
UFSAR Drawing 5.5.7-1, Flow Diagram Residual Heat Removal System
UFSAR Drawing 6.3.2-1, Flow Diagram Safety Injection System
UFSAR Section 7.6.2, Residual Heat Removal Isolation Valves
UFSAR Drawing 9.3.4-1, Flow Diagram Chemical and Volume Control System