April 8, 2005

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

SUBJECT: SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2 - SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUEST FOR RELIEF NO. ISPT-09 (TAC NOS. MC3946 AND MC3947)

Dear Mr. Singer:

The U.S. Nuclear Regulatory Commission (the staff), with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL) has reviewed and evaluated the information provided by Tennessee Valley Authority (the licensee) in its letter dated August 6, 2004, as supplemented by letter dated January 3, 2005, which proposed its Second 10-Year Interval Inservice Inspection (ISI) Program Plan Request for Relief No. ISPT-09 for the Sequoyah Nuclear Plant, Units 1 and 2.

The staff has concluded that compliance with the requirements of Section XI of the American Society of Mechanical Engineers (ASME) Code would result in a significant hardship or unusual difficulty without a compensating increase in quality and safety. The alternative proposed by the licensee in Request for Relief ISPT-09, Revision 1, provides reasonable assurance of the continued structural integrity of the subject components. Therefore, Request for Relief ISPT 09, Revision 1, is authorized pursuant to Title 10 *Code of Federal Regulation* Section 50.55a(a)(3)(ii) for the second 10-year ISI interval at the Sequoyah Nuclear Plant, Units 1 and 2 . All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The staff's evaluation and conclusions are contained in the staff’s safety evaluation provided in Enclosure 1. Enclosure 2 is the PNNL Technical Letter Report.

Sincerely,

/RA/
Michael L. Marshall, Jr., Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-327 and 50-328

Enclosures: As stated

cc w/enclosures: See next page

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The staff has concluded that compliance with the requirements of Section XI of the American Society of Mechanical Engineers (ASME) Code would result in a significant hardship or unusual difficulty without a compensating increase in quality and safety. The alternative proposed by the licensee in Request for Relief ISPT-09, Revision 1, provides reasonable assurance of the continued structural integrity of the subject components. Therefore, Request for Relief ISPT-09, Revision 1, is authorized pursuant to Title 10 *Code of Federal Regulation* Section 50.55a(a)(3)(ii) for the second 10-year ISI interval at the Sequoyah Nuclear Plant, Units 1 and 2 . All other requirements of the ASME Code, Section XI for which relief has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

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Tennessee Valley Authority

SEQUOYAH NUCLEAR PLANT

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION

REQUEST FOR RELIEF NO. ISPT-09

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

TENNESSEE VALLEY AUTHORITY

DOCKET NUMBERS 50-327 AND 50-328

1.0 INTRODUCTION

The U.S. Nuclear Regulatory Commission (the Commission or, the staff), with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), has reviewed and evaluated the information provided by Tennessee Valley Authority (the licensee) in its letter dated August 6, 2004, which proposed its Second 10-Year Interval Inservice Inspection (ISI) Program Plan Request for Relief No. ISPT-09 for the Sequoyah Nuclear Plant, Units 1 and 2. The licensee provided additional information in its letter dated January 3, 2005.

2.0 REGULATORY REQUIREMENTS

ISI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Class 1, 2, and 3 components is performed in accordance with Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," of the ASME Code and applicable addenda as required by Title 10 *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated in 10 CFR 50.55a(a)(3), 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 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 preservice examination requirements, set forth in the ASME Code, Section XI, 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. The applicable ASME Code of record for the Sequoyah Nuclear Plant, Units 1 and 2, second 10-year ISI interval, which began on December 16, 1995, is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code, with no addenda.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Requirement

The ASME Code, Section XI, Table IWB-2500-1, Examination Category B-P, Item B15.50, requires that a system hydrostatic test be performed on Class 1 components at or near the end of each ISI interval. The pressure retaining boundary during the test shall include all Class 1 components within the system boundary. The test pressure, as required by Paragraph IWB-5222(a), is required to be between 102 percent and 110 percent of the nominal operating pressure associated with 100 percent rated reactor power and corresponding to the system temperature during the test, as specified in Table IWB-5222-1.

3.2 Licensee's ASME Code Request

The licensee proposed an alternative to the pressure test requirements for portions of piping in the safety injection and residual heat removal systems that connect to the reactor coolant system components listed in Table 3.1 of PNNL's Technical Letter Report (TLR) provided in Enclosure 2. The licensee's alternative is to perform the hydrostatic tests at pressures less than those specified by the ASME Code based on the hardship that would be incurred if the ASME Code-required pressures are imposed.

As an alternative to pressurizing the subject line segments in accordance with the ASME Code requirements noted above, the licensee has proposed the following:

- For the subject safety injection system piping line segments, use the safety injection pumps running at minimum recirculation mode, to pressurize segments to approximately 1500 psig.

- For the subject residual heat removal (RHR) line segment, visually examine the piping when RHR is operating at 350 psig during plant start-up following the refueling outage.

3.3 Staff Evaluation

The staff determined that the ASME Code requirements would be a significant hardship for the licensee to perform. The licensee would have to make plant design modifications to enable the use of high pressure hoses as temporary jumpers around these valves or employ hydrostatic pumps connected directly to the piping segments. Either of these options would conflict with plant technical specifications and operational design requirements by potentially defeating the reactor coolant system boundary double isolation, which is mandated when fuel is present in the reactor vessel.

The licensee's proposal represents the highest test pressures that can be obtained without significant plant modifications and are intended to test the subject piping segments to conditions similar to those that may be experienced during postulated design basis events. It is expected that the proposed test pressures will be sufficient to produce detectable leakage from significant service-induced degradation sources, should these exist, as well as verify that connections in these piping segments that may have been opened during the outage have been properly secured. The licensee has also committed to meeting the hold times for insulated (4 hours) and noninsulated (10 minutes) components, as shown in paragraph IWA-5213, prior to performing the required VT-2 visual examinations.

To require the licensee to pressurize the subject piping segments in accordance with the ASME Code requirements noted above would require significant plant modifications and would subject the licensee to an undue burden with no compensating increase in quality or safety. The staff determined that the results from the proposed pressure tests will provide reasonable assurance of continued leakage integrity of the subject components and systems, in particular bolted connections within the test boundary. Furthermore, other system examinations, including ASME Code, Section XI nondestructive evaluations, as part of the licensee's ISI program plan and typical system walk downs looking for evidence of borated water leakage during the prior operating cycle, are expected to provide confidence in the structural integrity of the system boundaries.

4.0 CONCLUSIONS

The Sequoyah Nuclear Plant, Units 1 and 2, Request for Relief No. ISPT-09 from the ASME Code requirements, has been reviewed by the staff with the assistance of its contractor, PNNL. The TLR Enclosure 2 provides PNNL's evaluation of the request for relief. The staff has reviewed the TLR and adopts the evaluations and recommendations for authorizing the licensee's request for relief.

The staff has concluded that, for Request for Relief ISPT-09, Revision 1, compliance with the Code requirements would result in a significant hardship or unusual difficulty without a compensating increase in quality and safety. The alternative proposed by the licensee provides reasonable assurance of the continued structural integrity of the subject components. Therefore, Request for Relief ISPT-09, Revision 1, is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the second 10-year ISI interval at the Sequoyah Nuclear Plant, Units 1 and 2. All other requirements of the ASME Code, Section XI, for which relief has not been specifically requested remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: Thomas McLellan, NRR

Date: April 8, 2005

TECHNICAL LETTER REPORT

ON SECOND 10-YEAR INSERVICE INSPECTION INTERVAL

REQUEST FOR RELIEF ISPT-09, REVISION 1

FOR

TENNESSEE VALLEY AUTHORITY

SEQUOYAH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NUMBERS 50-327 AND 50-328

1.0 INTRODUCTION

By letter dated August 6, 2004, the licensee, Tennessee Valley Authority, submitted Request for Relief ISPT-09 from the requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code*, Section XI, for Sequoyah Nuclear Plant (SQN), Units 1 and 2. In response to an NRC Request for Additional Information, the licensee revised the request and submitted further information by letter dated January 3, 2005. Pacific Northwest National Laboratory (PNNL) has evaluated the revised request for relief and supporting information submitted by the licensee in Section 3 below.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (B&PV Code), and applicable addenda, as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). The regulation at 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 the licensee demonstrates that (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 preservice 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, which was 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. The ASME Code of record for SQN 1-2 second 10-year intervals inservice inspection, which began on December 16, 1995, is the 1989 Edition of Section XI, with no addenda.

3.0 TECHNICAL 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.

3.1 Request for Relief ISPT-09, Revision 1, Examination Category B-P, All Pressure Retaining Components

ASME Code Requirement: Examination Category B-P, Item B15.50, requires that a system hydrostatic test be performed on Class 1 components at or near the end of each inservice inspection interval. The pressure retaining boundary during the test shall include all Class 1 components within the system boundary. The test pressure, as required by Paragraph IWB-5222(a), is required to be between 102% and 110% of the nominal operating pressure associated with 100% rated reactor power and corresponding to the system temperature during the test, as specified in Table IWB-5222-1.

Licensee's ASME Code Relief Request: In accordance with 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the pressure test requirements for portions of piping in the safety injection and residual heat removal systems that connect to the reactor coolant system (see Table 3.1 below for descriptions of the piping segments included in this alternative). The licensee's alternative is to perform the hydrostatic tests at pressures less than those specified by ASME Code, based on the hardship that would be incurred if the ASME Code-required pressures are imposed.

Table 3.1 - Piping Segments in Request for Relief ISPT-09, Revision 1

Segment Description	NPS Diameter (inches)	Segment Length (feet)	Wall Thickness (inches)
Safety injection Accumulator 1 to Loop 1 cold leg (CKV-63-622 to CKV-63-560)	10	20	1.0
Low pressure safety injection from RHR system CKV-63-633 to 10-inch Loop 1 cold leg injection	6	25	0.719
High pressure safety injection piping from 3-inch common header to Loop 1 cold leg piping (CKV-63-581 to CKV-63-586)	1.5	60	0.281
Safety injection pump piping from CKV-63-543 to 8-inch Loop 1 hot leg injection	2	5	0.375
Safety injection pump piping from CKV-63-551 to 6-inch Loop 1 cold leg injection	2	25	0.375
Low pressure safety injection from RHR system to Loop 1 hot leg injection (CKV-63-640 to CKV-63-641)	8 6	30 5	0.812 0.719
Safety injection Accumulator 2 to Loop 2 cold leg (CKV-63-623 to CKV-63-561)	10	20	1.0

Table 3.1 - Piping Segments in Request for Relief ISPT-09, Revision 1			
Segment Description	**NPS Diameter (inches)**	**Segment Length (feet)**	**Wall Thickness (inches)**
Low pressure safety injection from RHR system CKV-63-632 to 10-inch Loop 2 cold leg injection	6	15	0.719
High pressure safety injection piping from 2.5-inch common header to Loop 2 cold leg piping (CKV-63-581 to CKV-63-587)	1.5	105	0.281
Safety injection pump piping from CKV-63-547 to 6-inch Loop 2 hot leg injection	2	40	0.375
Safety injection pump piping from CKV-63-553 to 6-inch Loop 2 cold leg injection	2	20	0.375
Safety injection Accumulator 3 to Loop 3 cold leg (CKV-63-624 to CKV-63-562)	10	20	1.0
Low pressure safety injection from RHR system CKV-63-634 to 10-inch Loop 3 cold leg injection	6	15	0.719
High pressure safety injection piping from 3-inch common header to Loop 3 cold leg piping (CKV-63-581 to CKV-63-588)	1.5	40	0.281
Safety injection pump piping from CKV-63-545 to 8-inch Loop 3 hot leg injection	2	5	0.375
Safety injection pump piping from CKV-63-555 to 6-inch Loop 3 cold leg injection	2	25	0.375
Low pressure safety injection from RHR system to Loop 3 hot leg injection (CKV-63-643 to CKV-63-644)	8	35	0.812
Safety injection Accumulator 4 to Loop 4 cold leg (CKV-63-625 to CKV-63-563)	10	25	1.0
Low pressure safety injection from RHR system CKV-63-635 to 10-inch Loop 4 cold leg injection	6	20	0.719
High pressure safety injection piping from 3-inch common header to Loop 4 cold leg piping (CKV-63-581 to CKV-63-589)	1.5	25	0.281
Safety injection pump piping from CKV-63-549 to CKV-63-558 in 6-inch Loop 4 hot leg injection	2 6	40 5	0.375 0.719
Safety injection pump piping from CKV-63-557 to 6-inch Loop 4 cold leg injection	2	5	0.375
RHR piping between FCV-74-1 and FCV-74-2	14	35	1.406

Licensee Basis for Relief (as stated):

The piping segments listed in [Table 3.1] are connected directly to the reactor coolant system, and, in accordance with the reactor coolant pressure boundary definition in 10 CFR 50 paragraph 50.2, are classified as ASME Class 1 up to and including the second isolation valve. Each of these piping segments, except for the RHR system piping, is isolated from the primary reactor coolant system (RCS) by a self-actuating check valve designed to prevent primary reactor coolant from escaping the RCS, while providing a passive injection flow-path for coolant injection. The use of check valves in these piping segments for isolation from the RCS prevents, by design, their pressurization by the primary RCS, and conversely, their pressurization to any pressure greater than that in the RCS.

The RHR piping segment is also connected directly to the RCS; however, this piping is isolated from the RCS by two in-series motor-operated valves (MOVs). These MOVs are interlocked to ensure redundant isolation of the RCS from the lower design pressure (600 psig) RHR system. Plant operating instructions require that these MOVs be closed when the RCS pressure exceeds 350 psig.

During performance of the Section XI inservice hydrostatic pressure test, the RCS would be brought to system normal operating pressure of approximately 2235 psig, at which time the subject piping segments are isolated from the RCS by their respective check valves, or FCV-74-1 in the RHR segment. No method currently exists for pressurizing these piping segments to full test pressure during the Section XI hydrostatic pressure test.

Two methods that TVA investigated are: (1) the use of temporary high pressure hoses connected to RCS test connections, vent or drain piping to “jumper” around the isolation check valves, and (2) the use of hydrostatic pumps connected to each piping segment. Both of these methods conflict with plant design requirements and 10 CFR 50.55a(c)(ii) by eliminating the double isolation boundary required for the reactor coolant pressure boundary when the reactor vessel contains nuclear fuel. The use of either of these methods would require a redesign of the RCS and the installation of new piping designed to meet the plant construction code and licensing commitments. This option is cost prohibitive and imposes a burden to TVA which is not commensurate with the increase to plant safety achieved through compliance with the ASME Section XI pressure test requirement versus use of the proposed alternative test method.

The purpose of the ASME Section XI pressure test is to detect existing through-wall defects in the pressure retaining boundary by the identification of leakage from the boundary. The detection of pressure boundary leakage from such through-wall defects can be achieved at pressures lower than the pressure associated with 100% rated reactor power.

Licensee's Proposed Alternative Examination (as stated):

The proposed alternate testing method will achieve the highest test pressure in each piping segment listed in [Table 3.1] that can be achieved without plant modification, and while continuing to comply with plant Technical Specifications and design requirements when nuclear fuel is contained in the reactor. The difference in the amount of leakage at the proposed alternative test pressure versus the ASME Section XI required test pressure is estimated by the following equation:

$$L_P = L_{XI} \text{ X } (P_P/P_{XI})^{1/2}$$

Where: L_P = the leakage at the proposed test pressure
L_{XI} = the leakage at the Section XI required pressure
P_P = the proposed test pressure of 1500 psig
P_{XI} = the Section XI required pressure of 2235 psig

For the safety injection system piping, the expected leakage from a through-wall defect would be approximately:

$$L_P = L_{XI} \text{ X } (1500/2235)^{1/2} = L_{XI} \text{ X } 0.82$$

or, 82% of the leakage at the higher Section XI test pressure.

For the RHR system piping, the expected leakage from a through-wall defect would be approximately:

$$L_P = L_{XI} \text{ X } (350/2235)^{1/2} = L_{XI} \text{ X } 0.4$$

or, 40% of the leakage at the higher Section XI test pressure.

The Section XI test procedure requires a holding time (4 hours for insulated components and 10 minutes for non-insulated components) after attaining test pressure in order to allow sufficient fluid leakage to collect to ensure detection by the visual, VT-2, examination.

As shown above, the estimated reduction in the amount of leakage from a through-wall defect would not be expected to prevent detection of a leak during a visual VT-2 examination.

The piping segments from the high pressure and intermediate pressure safety injection system and the safety injection accumulators will be pressurized using the safety injection pumps to approximately 1500 psig which is the pressure achieved with the safety injection pumps running in the minimum recirculation flow mode.

The piping segments from the RHR system segment will be pressurized to approximately 350 psig and visually examined when the RHR system is providing shutdown cooling during plant startup following the refueling outage.

Based on the hardships associated with costly plant modifications and redesign, TVA considers the proposed alternative test method to be acceptable for satisfying pressure boundary integrity of the segments identified in [Table 3.1] while maintaining compliance with plant design requirements, plant Technical Specifications and the requirement of 10 CFR 50.55a(a)(c)(ii). Sufficient test pressure in conjunction with the test pressure holding time will allow detection of any leakage from the pressure retaining boundary of the subject piping segments. Accordingly, TVA requests relief from the ASME code in accordance with 10 CFR 50.55a(a)(3)(ii).

<u>Licensee's Response to Request for Additional Information (as stated):</u>

The piping segments providing safety injection flow from the low pressure accumulators do not receive inservice examinations in the risk informed program due to the low probability of failure and their low failure consequence. These piping segments are constantly monitored during plant operation by the automatic monitoring of accumulator pressure and level. Any leakage in these segments would be immediately identified to plant operators and Technical Specification actions taken. In the event of failure of these segments all leakage will be directed to the containment sump and will remain available for recirculation through the RHR and Containment Spray systems. The high pressure and intermediate pressure safety injection segments have been evaluated to determine their risk ranking. Those segments which were determined to be high in safety significance receive volumetric inservice examinations of selected welds, where possible. High safety significant segments, which are small diameter (1 ½ inches) joined by socket welds, cannot be adequately examined by volumetric methods. These high safety segments and all low safety significant segments receive a VT-2 visual examination each refueling outage during unit startup.

<u>Evaluation:</u> The ASME Code requires that a system hydrostatic test be performed once each interval to include all Class 1 components within the reactor coolant system (RCS) boundary. The hydrostatic test must be performed at or near the end of the inservice inspection interval, and the test pressure is required to be between 102% and 110% of the nominal operating RCS system pressure associated with 100% rated reactor power, depending on the system temperature during the test. However, several piping line segments are connected to the RCS through self-actuating check valves, or inter-locked motor controlled valves, which does not allow normal RCS pressure to be used to pressurize these segments. In order to test the subject piping segments to normal operating RCS pressure (approximately 2235 psig), the licensee would have to make plant design modifications to enable the use of high pressure hoses as temporary jumpers around valves or employ hydrostatic pumps connected directly to the piping segments. Either of these options would conflict with plant technical specifications and operational design requirements by potentially defeating the RCS boundary double isolation, which is mandated when fuel is present in the reactor vessel. To require the licensee to make plant modifications in order to pressurize the subject line segments to normal RCS pressure would result in a considerable hardship.

Pressure testing of the RCS is typically performed during the *return to power* sequence at the end of a refueling outage using reactor coolant pumps and pressurizer heaters to bring the RCS to normal operating temperature and pressure, prior to initiating core criticality. At this time, the subject safety injection system (SIS) and residual heat removal (RHR) piping segments are isolated from the RCS. These segments are described in Table 3.1, and primarily consist of limited runs of piping between the first and second isolation valves in the SIS connections on each of the four primary coolant loops. In addition, a section of RHR piping between the first and second isolation valves is also included. The piping segments are fabricated of austenitic stainless steel and range in diameter from 1.5 to 14-inch NPS (see Table 3.1 for specific sizes and wall thicknesses). These segments, including the first and second isolation valves, are considered part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2.

For SIS piping segments connecting to RCS Loops 1 through 4, the self-actuating isolation check valves are designed to prevent back-flow of primary coolant into the respective high and low pressure SIS piping, while providing a passive flow-path for injecting coolant during normal start-ups and shutdowns, as well as during postulated emergency events. Therefore, the design and function of these valves do not allow piping upstream of the first isolation check valve in each line segment to experience normal RCS pressures. In order to subject the identified piping segments to RCS pressure, the first isolation valve would have to be by-passed. This would require the licensee to make pressure boundary modifications to the existing piping to accommodate fittings, valves, or other appurtenances needed to support this activity. Another option would be for the licensee to use a stand-alone hydrostatic pump connected to the subject piping between the first and second isolation valves to obtain a pressure equivalent to that during normal RCS operation. Again, this may require modifications to the piping pressure boundary, and could potentially inject water into the primary system if pump pressure slightly exceeds normal RCS pressure. Either of these methods would result in a significant hardship for the licensee.

Similar problems exist for the RHR piping segment, which has redundant isolation from the RCS by two, inter-locked motor operated valves. The RHR system has a maximum design pressure of 600 psig, and is normally only operated during shutdown and start-up sequences. The motor operated valves are closed and locked prior to the RCS pressure exceeding 350 psig, therefore the RHR piping segment cannot be pressurized during a normal RCS pressure test sequence.

As an alternative to pressurizing the subject line segments in accordance with the ASME Code requirements noted above, the licensee has proposed the following:

1. For the subject SIS piping line segments, use the safety injection pumps running at minimum recirculation mode, to pressurize segments to approximately 1500 psig.
2. For the subject RHR line segment, visually examine the piping when RHR is operating at 350 psig during plant start-up following the refueling outage.

The licensee's proposal represents the highest test pressures that can be obtained without significant plant modifications and are intended to test the subject piping segments to conditions similar to those that may be experienced during postulated design basis events. It is expected that the proposed test pressures will be sufficient to produce detectable leakage from significant service-induced degradation sources, should these exist, as well as verify that connections in these piping segments that may have been opened during the outage have been properly secured. The licensee has also committed to meeting the hold times for insulated (4 hours) and non-insulated (10 minutes) components, as shown in paragraph IWA-5213, prior to performing the required VT-2 visual examinations.

It is concluded that, to require the licensee to pressurize the subject piping segments in accordance with the ASME Code requirements noted above would require significant plant modifications and would subject the licensee to an undue burden with no compensating increase in quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that the licensee’s proposed alternative be authorized.

4.0 CONCLUSIONS

Pacific Northwest National Laboratory has reviewed the licensee's submittal and concludes, for Request for Relief ISPT-09, Revision 1, that compliance with the ASME Code requirements would result in a hardship or unusual difficulty with no compensating increase in quality or safety. The results from the alternative pressure tests proposed by the licensee provide reasonable assurance of the continued leakage integrity of the subject piping segments. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Request for Relief ISPT-09, Revision 1, be authorized for the second 10-year interval at Sequoyah Nuclear Plant, Units 1 and 2.