November 1, 2005

Mr. David A. Christian
Sr. Vice President and Chief Nuclear Officer
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

SUBJECT: SURRY POWER STATION, UNITS 1 AND 2 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS (ASME), SECTION XI FOURTH 10-YEAR INSERVICE INSPECTION (ISI) PROGRAM (TAC NOS. MC5586, MC5587, MC5596, MC5597, MC5598, AND MC5600)

Dear Mr. Christian:

By letter dated January 10, 2005, as supplemented by letter dated July 19, 2005, Virginia Electric and Power Company (VEPCO) requested relief from the ASME Code, Section XI requirements for the fourth 10-year ISI interval at Surry Power Station, Units 1 and 2. In this submittal, VEPCO requested Nuclear Regulatory Commission (NRC) staff approval of Relief Requests SPT-005, SPT-006, and SPT-007 for Surry, Unit 1 and SPT-004, SPT-005, and SPT-006 for Surry, Unit 2.

Our evaluations of Relief Requests SPT-005 and SPT-006 for Surry, Unit 1 and SPT-004 and SPT-005 for Surry, Unit 2 are Enclosed. In its letter dated July 19, 2005, VEPCO withdrew Relief Requests SPT-007 for Surry, Unit 1 and SPT-006 for Surry, Unit 2.

For Relief Requests SPT-005 and SPT-006 for Surry, Unit 1 and SPT-004 and SPT-005 for Surry, Unit 2, the NRC staff has determined that the ASME Code-required examinations would cause significant hardship without a compensating increase in the level of quality and safety. VEPCO's proposed alternative provides reasonable assurance of leak tightness and structural integrity of the subject components. Therefore, VEPCO's Relief Requests SPT-005 and SPT-006 for Surry, Unit 1 and SPT-004 and SPT-005 for Surry, Unit 2 are granted pursuant to Title 10 of the *Code of Federal Regulations* Section 50.55a(a)(3)(ii) for the fourth 10-year ISI interval.

The NRC staff is closing out TAC Nos. MC5586, MC5587, MC5596, MC5597, MC5598, and MC5600 with this letter.

Sincerely,

/RA/
Evangelos Marinos, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosure: As stated

cc w/ encl: See next page

The NRC staff is closing out TAC Nos. MC5586, MC5587, MC5596, MC5597, MC5598, and MC5600 with this letter.

Sincerely,

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUESTS SPT-004 THROUGH SPT-007

SURRY POWER STATION, UNITS 1 AND 2

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NOS. 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated January 10, 2005, as supplemented by letter dated July 19, 2005, Virginia Electric and Power Company (VEPCO, the licensee) submitted Relief Requests SPT-005, SPT-006, and SPT-007 for Surry, Unit 1 and SPT-004, SPT-005, and SPT-006 for Surry, Unit 2 for the fourth 10-year inservice inspection (ISI) interval. In its letter dated July 19, 2005, VEPCO withdrew Relief Requests SPT-007 for Surry, Unit 1 and SPT-006 for Surry, Unit 2. The Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), has evaluated the information provided by the licensee. As a result, the NRC staff adopts the evaluations and recommendations for granting or authorizing relief contained in PNNL's Technical Letter Report (TLR), included as Attachment 1 of this Safety Evaluation (SE). Attachment 2 of this SE lists each relief request and the status of approval.

2.0 REGULATORY REQUIREMENTS

ISI of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the *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). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the Director of the Office of Nuclear Reactor Regulation, 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 pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 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 ASME Code of Record for Surry, Units 1 and 2 is the 1998 Edition of ASME Section XI, through and including the 2000 Addenda. The fourth 10-year ISI interval program at Surry, Unit 1 started on October 14, 2003, and is scheduled to end on October 13, 2013. Likewise, the fourth 10-year ISI interval program at Surry, Unit 2 started on May 10, 2004, and is scheduled to end on May 9, 2014.

3.0 TECHNICAL EVALUATION

The NRC staff, with technical assistance from its contractor, PNNL, has reviewed and evaluated the information provided by the licensee in its letter dated January 10, 2005. In its application for the fourth 10-year ISI Program Plan, the licensee proposed Requests for Relief SPT-005, SPT-006, and SPT-007 for Surry, Unit 1 and Requests for Relief SPT-004, SPT-005, and SPT-006 for Surry, Unit 2. In response to a NRC request for additional information, the licensee withdrew Requests for Relief SPT-007 and SPT-006 for Surry, Units 1 and 2, respectively, and provided additional information in its letter dated July 19, 2005. The NRC staff adopts the evaluations and recommendations for authorizing relief contained in PNNL's TLR included as Attachment 1 of this SE. Attachment 2 of this SE lists each relief request and the status of approval.

For Requests for Relief SPT-005 and SPT-006 for Surry, Unit 1 and Requests for Relief SPT-004 and SPT-005 for Surry, Unit 2, the ASME Code, Section XI, Examination Category B-P, Item B15.50, requires that a system leakage test, as defined in Paragraph IWB-5220, be performed on Class 1 piping systems following each reactor refueling outage prior to plant start-up. Paragraph IWB-5220 requires the system leakage test to be conducted at a pressure not less than the pressure corresponding with 100 percent rated reactor power. The pressure retaining boundary during Class 1 system leakage tests shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation start-up. Additionally, when the system leakage test is conducted at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 components within each system boundary.

The licensee requested relief from performing ASME Code-required system leakage tests at normal reactor coolant system (RCS) pressure of approximately 2235 pounds per square inch gauge on certain piping segments. As an alternative the licensee proposed to perform the system leakage tests at pressures less than those specified by ASME Code. To perform the ASME Code-required system leakage test at normal RCS pressure 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 operational design requirements by potentially defeating the RCS boundary double isolation, which is mandated when fuel is present in the reactor vessel.

Therefore, the NRC staff determined that having the licensee perform the ASME Code-required system leakage tests at normal RCS pressure would be a significant hardship without a compensating increase in quality and safety. Furthermore, the NRC staff determined that the licensee's proposed alternative to perform the system leakage tests at pressures less than

those specified by ASME Code provides reasonable assurance leak tightness and structural integrity of the piping segments.

4.0 CONCLUSION

The Surry, Unit 1 Requests for Relief SPT-005, and SPT-006 and Surry, Unit 2 Requests for Relief SPT-004 and SPT-005 to the ASME Code requirements have been reviewed by the NRC staff with the assistance of its contractor, PNNL. The TLR provides PNNL's evaluation of these requests for relief. The NRC staff has reviewed the contractor's TLR and adopts the evaluations and recommendations for authorizing the Surry, Unit 1 Requests for Relief SPT-005 and SPT-006 and Surry, Unit 2 Requests for Relief SPT-004 and SPT-005 for the fourth 10-year ISI interval.

For the Surry, Unit 1 Requests for Relief SPT-005, and SPT-006 and Surry, Unit 2 Requests for Relief SPT-004, and SPT-005 the NRC staff concluded that the ASME Code-required examinations would cause significant hardship without a compensating increase in the level of quality and safety and the licensee's proposed alternative provides reasonable assurance of leak tightness and structural integrity of the piping segments. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval.

All other requirements of the ASME Code, Sections III and XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Attachments: 1. TLR, Pacific Northwest National Laboratory
2. Summary of Relief Requests

Principal Contributor: T. McLellan

Date: November 1, 2005

TECHNICAL LETTER REPORT

ON FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

REQUESTS FOR RELIEF

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION)

SURRY POWER STATION, UNITS 1 AND 2

DOCKET NUMBERS 50-280 AND 50-281

1.0 INTRODUCTION

By letter dated January 10, 2005, the licensee, Virginia Power and Electric Company (Dominion), submitted several requests from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*. The requests are for the fourth 10-year inservice inspection (ISI) interval at Surry Power Station, Units 1 and 2 (Surry 1-2). In response to a U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information (RAI), the licensee withdrew Request for Reliefs SPT-007 and SPT-006 for Surry 1-2 respectively and submitted further information by letter dated July 19, 2005. Pacific Northwest National Laboratory (PNNL) has evaluated the requests for relief and supporting information submitted by the licensee in Section 3.0 below.

2.0 REGULATORY REQUIREMENTS

Inservice inspection (ISI) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by Title 10 of the Code of Federal Regulation (10 CFR) 50.55a(g), except where specific relief has been granted by the NRC 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 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 Code of record for the

Surry, Units 1 and 2, fourth 10-year interval ISI programs, which began on October 14, 2003 and May 10, 2004, respectively, is the 1998 Edition of ASME Section XI, through and including the 2000 Addenda.

3.0 TECHNICAL EVALUATION

The information provided by Virginia Electric and Power Company (Dominion) in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below.

3.1 Request for Relief SPT-005, (TAC MC5586), Surry-1, ASME Code, Section XI, Examination Category B-P, All Pressure Retaining Components

ASME Code Requirement: [ASME Code, Section XI,] Examination Category B-P, Item B15.50, requires that a system leakage test, as defined in Paragraph IWB-5220, be performed on Class 1 piping systems following each reactor refueling outage prior to plant start-up. [ASME Code, Section XI,] Paragraph IWB-5220 requires the system leakage test to be conducted at a pressure not less than the pressure corresponding with 100% rated reactor power. The pressure retaining boundary during [ASME Code Class 1] system leakage tests shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation start-up. Additionally, when the system leakage test is conducted at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 components within each system boundary.

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 (SI) and Residual Heat Removal (RHR) Systems that connect to the Reactor Coolant System (RCS) (see Table 3.1 below for descriptions of the piping segments included in this alternative). The licensee's alternative is to perform the system leakage 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 SPT-005, Surry 1

Segment Description	NPS Diameter (inches)	Segment Length (feet)
Between valves 1-SI-109, 1-SI-107 and 1-SI-HCV-1850B	From 1-SI-109 to 1-SI-107 is 12 inch From RCS loop to 1-SI-HCV-1850B is 3/4 inch	12 8
Between valves 1-SI-130, 1-SI-128,1-SI-HCV-1850D and 1-RH-MOV-1720A	From 1-SI-130 to 1-SI-128 is 12 inch From RCS loop to 1-SI-HCV-1850D is 1 inch and 3/4 inch From RCS loop to 1-RH-MOV-1720A is 10 inch	90 1 to 3 127
Between valves 1-SI-147, 1-SI-145, 1-SI-HCV-1850F and 1-RH-MOV-1720B	From 1-SI-147 to 1-SI-145 is 12 inch From RCS loop to 1-SI-HCV-1850F is 3/4 inch From RCS loop to 1-RH-MOV-1720B is 10 inch	91 6 25

Licensee Basis for Relief (as stated):

Normal reactor coolant pressure at 100% rated power is approximately 2235 psig [pounds per square inch gauge]. The piping in question is separated from this reactor coolant pressure by a single check valve, and as such does not normally see this pressure. Part of the area in question is pressurized during normal operation to approximately 660 psig from the passive safety injection accumulators.
The accumulator pressure is monitored from the control room throughout the operating cycle. An external pressurization source would be necessary to meet the requirement of normal reactor coolant pressure. Since the check valve would be part of the test boundary, a pressure differential would be required between the reactor coolant system and the area in question to maintain check valve closure. Maintaining the differential pressure and ensuring no test fluid intrusion into the reactor coolant system (reactivity control issue) is considered unusually difficult to meet with no compensating increase in quality or level of safety when considering the alternative below.

Licensee's Proposed Alternative Examination (as stated):

The areas in question are examined (VT-2) each refueling as part of the normal Class 1 system leakage test (normal valve line-up) for evidence of leakage. Except for the areas between [valves] 1-SI-108 and 1-SI-HCV-1850B, 1-SI-129 and 1-SI-HCV-1850D, and 1-SI-146 and 1-SI-HCV-1850F, the areas will be examined at that time at the safety injection accumulator nominal operating pressure. Additionally, except for the areas between [valves] 1-SI-108 and 1-SI-HCV-1850B, 1-SI-129 and 1-SI-HCV-1850D, and 1-SI-146 and 1-SI-HCV-1850F, the components are under safety injection accumulator pressure during normal operation. Through-wall leakage for these areas would be identified by the control room through their monitoring of the pressure on the safety injection accumulators. The areas between [valves] 1-SI-108 and 1-SI-HCV-1850B, 1-SI-129 and 1-SI-HCV-1850D, and 1-SI-146 and 1-SI-HCV-1850F would also be examined at or near the end of the interval by using an external pressurization source, or by opening the isolation valves separating the lines from the safety injection accumulator pressure. The test pressure would again correspond to the safety injection accumulator nominal operating pressure.

In accordance with 10 CFR 50.55a(a)(3)(ii) Dominion requests that the tests described above be alternatively performed for the fourth inspection interval.

Evaluation: The ASME Code requires that a system leakage test be performed at the end of each refueling outage, and when performed at or near the end of the interval, the test must include all Class 1 components within the RCS boundary. The system leakage test must be performed at a test pressure not less than the nominal operating RCS pressure corresponding with 100% rated reactor power. However, several SI and RHR piping line segments are connected to the RCS through self-actuating check 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 operational design requirements by potentially defeating the RCS boundary double isolation, which is mandated when fuel is present in the reactor vessel. In addition, pressurizing these segments to normal RCS pressures may result in test fluid intrusion into the RCS system boundary, which could become a reactivity control issue. 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 that time, the subject safety injection system (SI) 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 SI and RHR connections on each of the primary coolant loops. The piping segments are fabricated of austenitic stainless steel and range in diameter from 3/4-inch to 12-inch nominal pipe size (NPS) (see Table 3.1 for specific sizes). 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 SI piping segments connecting to the main RCS loops, the self-actuating isolation check valves are designed to prevent back-flow of primary coolant into the respective high and low pressure SI 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 segments connected to the RCS upstream of the first SI check valve. The RHR system has a maximum design pressure of approximately 600 psig, therefore, it is only operated during normal shutdown and start-up sequences, and is not designed to experience normal RCS operating pressure. The first motor operated valve is closed and locked prior to the RCS pressure exceeding 465 psig, thus the RHR piping segments between the first SI check valve and the locked RHR motor operated valve 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 to perform the

system-hydrostatic tests at approximately 660 psia, the normal operating pressure of the SI system and accumulators.

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 is also required to meet 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.

3.2 Request for Relief SPT-006, (TAC MC5587), Surry-1, ASME Code, Section XI, Examination Category B-P, All Pressure Retaining Components

ASME Code Requirement: [ASME Code, Section XI,] Examination Category B-P, Item B15.50, requires that a system leakage test, as defined in Paragraph IWB-5220, be performed on Class 1 piping systems following each reactor refueling outage prior to plant start-up. [ASME Code, Section XI,] Paragraph IWB-5220 requires the system leakage test to be conducted at a pressure not less than the pressure corresponding with 100% rated reactor power. The pressure retaining boundary during [Class 1] system leakage tests shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation start-up. Additionally, when the system leakage test is conducted at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 components within each system boundary.

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 Class 1 components and piping between valves 1-RH-MOV [motor operated valve] -1700 and 1-RH-MOV-1701 in the suction line of the residual heat removal (RHR) system. The licensee's alternative is to perform the system leakage test at a pressure less than that specified by ASME Code, based on the hardship that would be incurred if the ASME Code-required pressure is imposed.

Licensee Basis for Relief (as stated):

Normal reactor coolant pressure at 100% rated power is approximately 2235 psig. The piping in question is separated from this reactor coolant pressure by a single closed valve, and as such does not normally see this pressure. Opening valve

1-RH-MOV-1700 is prevented by a pressure interlock, which prevents opening, when pressure in the reactor coolant system is above 465 psig. The interlock protects the low pressure RHR system from being over-pressurized by the higher pressure reactor coolant system. There is no other valve that would allow pressurization of the area from an external source. The system design prevents Code compliance in this area and it is therefore considered a hardship, since a plant modification would be needed, to meet code requirements.

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

The area in question is examined (VT-2) each refueling as part of the normal Class 1 system leakage test (normal valve line-up) for evidence of leakage. Additionally, it is proposed that the area be examined as part of the Class 2 system leakage test pressure boundary using the Class 2 test requirements associated with the adjoining Class 2 piping. This would result in an additional pressure test each period at the Class 2 RHR system nominal operating pressure with the associated visual VT-2 examination.

It is requested per 10 CFR 50.55a(a)(3)(ii) that the tests described above for the fourth inspection interval be alternatively performed.

<u>Evaluation:</u> The ASME Code requires that a system leakage test be performed at the end of each refueling outage, and when performed at or near the end of the interval, the test must include all Class 1 components within the RCS boundary. The system leakage test must be performed at a test pressure not less than the nominal operating RCS pressure corresponding with 100% rated reactor power. However, an RHR suction piping segment located between two inter-locked motor controlled valves does not allow normal RCS pressure to be used to pressurize this segment. In order to test the subject piping segment 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 these valves or employ hydrostatic pumps connected directly to the piping segment. Either of these options would conflict with 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 that time, the subject RHR piping segment is isolated from the RCS by two, inter-locked motor operated control valves. The subject components are fabricated of austenitic stainless steel and consist of a thirty-two foot long segment of 14-inch NPS diameter piping, and motor operated valves 1-RH-MOV-1700 and 1-RH-MOV-1701. This segment, including the motor operated control valves, are considered part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2.

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

required by technical specifications to be closed and locked prior to the RCS pressure exceeding 465 psig, therefore the RHR piping segment cannot be pressurized during a normal RCS pressure test sequence. In order to subject the identified piping segment to RCS pressure, the first motor operated 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 would require modifications to the existing pressure boundary. Either of these methods would result in a significant hardship for the licensee.

The licensee has proposed to perform the pressure tests and accompanying visual VT-2 examinations of the subject piping segment along with the remaining Class 2 (downstream of valve 1-RH-MOV-1701) portions of the system during the RHR system walk-downs at the beginning of each refueling outage. The test pressure will be approximately 300-350 psig, which is the maximum normal operating pressure of the RHR system. In addition, this piping segment will continue to receive a visual VT-2 examination in conjunction with the Class 1 portions of the RCS, although not at RCS pressure.

The licensee's proposal represents the highest test pressure that can be obtained without significant plant modifications and is intended to test the subject piping segment 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. 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.

3.3 <u>Request for Relief SPT-007, (TAC MC5596), Surry-1, ASME Code, Section XI, Examination Category B-P, All Pressure Retaining Components</u>

Note: As a result of an NRC Request for Additional Information (RAI), the licensee elected to withdraw this request for relief.

3.4 <u>Request for Relief SPT-004, (TAC MC5600), Surry-2, Examination Category B-P, All Pressure Retaining Components</u>

<u>ASME Code Requirement:</u> [ASME Code, Section XI,] Examination Category B-P, Item B15.50, requires that a system leakage test, as defined in Paragraph IWB-5220, be performed on Class 1 piping systems following each reactor refueling outage prior to plant start-up. [ASME Code, Section XI,] Paragraph IWB-5220 requires the system leakage test to be conducted at a pressure not less than the pressure corresponding with 100% rated reactor power. The pressure retaining boundary during [Class 1] system leakage tests shall correspond to the reactor coolant boundary, with all valves in

the position required for normal reactor operation start-up. Additionally, when the system leakage test is conducted at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 components within each system boundary.

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 (SI) and Residual Heat Removal (RHR) Systems that connect to the Reactor Coolant System (RCS) (see Table 3.4 below for descriptions of the piping segments included in this alternative). The licensee's alternative is to perform the system leakage 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 3.4 - Piping Segments in Request for Relief SPT-004, Surry-2

Segment Description	NPS Diameter (inches)	Segment Length (feet)
Between valves 2-SI-109, 2-SI-107 and 2-SI-HCV-2850B	From 2-SI-109 to 2-SI-107 is 12 inch From RCS loop to 2-SI-HCV-2850B is 3/4 inch	92 2
Between valves 2-SI-130, 2-SI-128 and 2-SI-HCV-2850D and 2-RH-47	From 2-SI-130 to 2-SI-128 is 12 inch From RCS loop to 2-SI-HCV-2850D is 3/4 inch From RCS loop to 2-RH-47 is 10 inch	86 3 9
Between valves 2-SI-147, 2-SI-145 and 2-SI-HCV-2850F and 2-RH-MOV-2720B	From 2-SI-147 to 2-SI-145 is 12 inch From RCS loop to 2-SI-HCV-2850F is 3/4 inch From RCS loop to 2-RH-MOV-2720B is 10 inch	94 3 23

Licensee Basis for Relief (as stated):

Normal reactor coolant pressure at 100% rated power is approximately 2235 psig. The piping in question is separated from this reactor coolant pressure by a single check valve, and as such does not normally see this pressure. Part of the area in question is pressurized during normal operation to approximately 660 psig from the passive safety injection accumulators. The accumulator pressure is monitored from the control room throughout the operating cycle. An external pressurization source would be necessary to meet requirement of normal reactor coolant pressure. Since the check valve would be part of the test boundary, a pressure differential would be required between the reactor coolant system and the area in question to maintain check valve closure. Maintaining the differential pressure and ensuring no test fluid intrusion into the reactor coolant system (reactivity control issue) is considered unusually difficult to meet with no compensating increase in quality or level of safety when considering the alternative below.

Licensee's Proposed Alternative Examination (as stated):

The areas in question are examined (VT-2) each refueling as part of the normal Class 1 system leakage test (normal valve line-up) for evidence of leakage. Except for the areas between, 2-SI-108 and 2-SI-HCV-2850B, 2-SI-129 and 2-SI-HCV-2850D, and 2-SI-146 and 2-SI-HCV-2850F, the areas will be examined at that time at the safety injection accumulator nominal operating pressure. Additionally, except for the areas between, 2-SI-108 and 2-SI-HCV-2850B, 2-SI-129 and 2-SI-HCV-2850D, and 2-SI-146 and 2-SI-HCV-2850F, the components are under safety injection accumulator pressure during normal operation. Through-wall leakage for these areas would be identified by the control room through their monitoring of the pressure on the safety injection accumulators. The areas between 2-SI-108 and 2-SI-HCV-2850B, 2-SI-129 and 2-SI-HCV-2850D, and 2-SI-146 and 2-SI-HCV-2850F would also be examined at or near the end of the interval by using an external pressurization source, or by opening the isolation valves separating the lines from the safety injection accumulator pressure. The test pressure would again correspond to the safety injection accumulator nominal operating pressure.

In accordance with 10 CFR 50.55a(a)(3)(ii) Dominion requests that the tests described above be alternatively performed for the fourth inspection interval.

Evaluation: The ASME Code requires that a system leakage test be performed at the end of each refueling outage, and when performed at or near the end of the interval, the test must include all Class 1 components within the RCS boundary. The system leakage test must be performed at a test pressure not less than the nominal operating RCS pressure corresponding with 100% rated reactor power. However, several SI and RHR piping line segments are connected to the RCS through self-actuating check 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 operational design requirements by potentially defeating the RCS boundary double isolation, which is mandated when fuel is present in the reactor vessel. In addition, pressurizing these segments to normal RCS pressures may result in test fluid intrusion into the RCS system boundary, which could become a reactivity control issue. 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 that time, the subject safety injection system (SIS) segments are isolated from the RCS. These segments are described in Table 3.4, and primarily consist of limited runs of piping between the first and second isolation valves in the SI and RHR connections on each of the primary coolant loops. The piping segments are fabricated of austenitic stainless steel and range in diameter from 3/4-inch to 12-inch

NPS (see Table 3.4 for specific sizes). 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 SI piping segments connecting to the main RCS loops, the self-actuating isolation check valves are designed to prevent back-flow of primary coolant into the respective high and low pressure SI 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 segments connected to the RCS upstream of the first SI check valve. The RHR system has a maximum design pressure of approximately 600 psig, therefore, it is only operated during normal shutdown and start-up sequences, and is not designed to experience normal RCS operating pressure. The first motor operated valve is closed and locked prior to the RCS pressure exceeding 465 psig, thus the RHR piping segments between the first SI check valve and the locked RHR motor operated valve 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 to perform the system hydrostatic tests at approximately 660 psia, the normal operating pressure of the SI system and accumulators.

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 is also required to meeting the hold times for insulated (4 hours) and non-insulated (10 minutes) components, as shown in [ASME Code, Section XI,] 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.

3.5 Request for Relief SPT-005, (TAC MC5597), Surry-2, ASME Code, Section XI, Examination Category B-P, All Pressure Retaining Components

ASME Code Requirement: [ASME Code, Section XI,] Examination Category B-P, Item B15.50, requires that a system leakage test, as defined in Paragraph IWB-5220, be performed on Class 1 piping systems following each reactor refueling outage prior to plant start-up. [ASME Code, Section XI,] Paragraph IWB-5220 requires the system leakage test to be conducted at a pressure not less than the pressure corresponding with 100% rated reactor power. The pressure retaining boundary during [Class 1] system leakage tests shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation start-up. Additionally, when the system leakage test is conducted at or near the end of each inspection interval, the pressure retaining boundary shall extend to all Class 1 components within each system boundary.

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 Class 1 components and piping between valves MOV-2700 and MOV-2701 (RHR suction). The licensee's alternative is to perform the system leakage test at a pressure less than specified by ASME Code, based on the hardship that would be incurred if the ASME Code-required pressure is imposed.

Licensee Basis for Relief (as stated):

Normal reactor coolant pressure at 100% rated power is approximately 2235 psig. The piping in question is separated from this reactor coolant pressure by a single closed valve, and as such does not normally see this pressure. Opening valve MOV-2700 is prevented by a pressure interlock, which prevents opening, when pressure in the reactor coolant system is above 465 psig. The interlock protects the low pressure RHR system from being over-pressurized by the higher pressure reactor coolant system. There is no other valve that would allow pressurization of the area from an external source. The system design prevents Code compliance in this area and it is therefore considered a hardship, since a plant modification would be needed, to meet code requirements.

Licensee's Proposed Alternative Examination (as stated):

The area in question is examined (VT-2) each refueling as part of the normal Class 1 system leakage test (normal valve line-up) for evidence of leakage. Additionally, it is proposed that the area be examined as part of the Class 2 system leakage test pressure boundary using the Class 2 test requirements associated with the adjoining Class 2 piping. This would result in an additional pressure test each period at the Class 2 RHR system nominal operating pressure with the associated visual VT-2 examination.

It is requested per 10 CFR 50.55a(a)(3)(ii) that the tests described above for the fourth inspection interval be alternatively performed.

<u>Evaluation:</u> The ASME Code requires that a system leakage test be performed at the end of each refueling outage, and when performed at or near the end of the interval, the test must include all Class 1 components within the RCS boundary. The system leakage test must be performed at a test pressure not less than the nominal operating RCS pressure corresponding with 100% rated reactor power. However, an RHR suction piping segment located between two inter-locked motor controlled valves does not allow normal RCS pressure to be used to pressurize this segment. In order to test the subject piping segment 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 these valves or employ hydrostatic pumps connected directly to the piping segment. Either of these options would conflict with 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 that time, the subject RHR piping segment is isolated from the RCS by two, inter-locked motor operated control valves. The subject components are fabricated of austenitic stainless steel and consist of a thirty-three foot long segment of 14-inch NPS diameter piping, and motor operated valves MOV-2700 and MOV-2701. This segment, including the motor operated control valves, are considered part of the reactor coolant pressure boundary, as defined in 10 CFR 50.2.

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 required by technical specifications to be closed and locked prior to the RCS pressure exceeding 465 psig, therefore the RHR piping segment cannot be pressurized during a normal RCS pressure test sequence. In order to subject the identified piping segment to RCS pressure, the first motor operated 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 would require modifications to the existing pressure boundary. Either of these methods would result in a significant hardship for the licensee.

The licensee has proposed to perform the pressure tests and accompanying visual VT-2 examinations of the subject piping segment along with the remaining Class 2 (downstream of valve MOV-2701) portions of the system during the RHR system walk-downs at the beginning of each refueling outage. The test pressure will be approximately 300-350 psig, which is the maximum normal operating pressure of the RHR system. In addition, this piping segment will continue to receive a visual VT-2

examination in conjunction with the Class 1 portions of the RCS, although not at RCS pressure.

The licensee’s proposal represents the highest test pressure that can be obtained without significant plant modifications and is intended to test the subject piping segment 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. 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.

3.6 Request for Relief SPT-006, (TAC MC5598), Surry-2, ASME Code, Section XI, Examination Category B-P, All Pressure Retaining Components

Note: As a result of an NRC RAI, the licensee elected to withdraw this request for relief.

4.0 CONCLUSIONS

Pacific Northwest National Laboratory has reviewed the licensee's submittal and concludes, for Requests for Relief SPT-005, and SPT-006 for Surry, Unit 1, that compliance with the ASME Code requirements would result in a significant hardship or unusual difficulty with no compensating increase in quality or safety. The results from the alternative pressure tests proposed by the licensee will be sufficient to produce detectable leakage from significant service-induced sources, should these exist. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Requests for Relief SPT-005 and SPT-006 be authorized for the fourth 10-year interval at Surry Power Station, Unit 1.

Similarly, for Requests for Relief SPT-004, and SPT-005 for Surry, Unit 2, it has been shown 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 will be sufficient to produce detectable leakage from significant service-induced sources, should these exist. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Requests for Relief SPT-004 and SPT-005 be authorized for the fourth 10-year interval at Surry Power Station, Unit 2.

As a result of an NRC Request for Additional Information (RAI), the licensee elected to withdraw Requests for Relief SPT-007 for Surry 1, and SPT-006 for Surry 2.

SURRY POWER STATION, UNITS 1 AND 2
Fourth 10-Year ISI Interval

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	PNNL TLR Sec.	System or Component	Exam. Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Disposition
SPT-005 (Surry 1)	3.1	Safety Injection Piping	B-P	B15.50	100% of Class 1 pressure-retaining boundary is required to be pressure tested at nominal RCS pressure for 100% rated power	Visual VT-2	Perform leakage tests in specific piping segments at lower than RCS pressure	Authorized 10 CFR 50.55a(a)(3)(ii)
SPT-006 (Surry 1)	3.2	Residual Heat Removal Piping	B-P	B15.50	100% of Class 1 pressure-retaining boundary is required to be pressure tested at nominal RCS pressure for 100% rated power	Visual VT-2	Perform leakage tests between valves 1-RH-MOV-1700 and -1701 along with other Class 2 portions of this piping at lower than RCS pressure	Authorized 10 CFR 50.55a(a)(3)(ii)
SPT-007 (Surry 1)	3.3	All Pressure Retaining Components	B-P	B15.50 B15.70	100% of pressure-retaining boundary is required to be tested following each refueling outage	Visual VT-2	Conduct pressure testing and VT-2 examinations at beginning of refueling outages	Withdrawn by licensee in letter dated July 19, 2005
SPT-004 (Surry 2)	3.4	Safety Injection Piping	B-P	B15.50	100% of Class 1 pressure-retaining boundary is required to be pressure tested at nominal RCS pressure for 100% rated power	Visual VT-2	Perform leakage tests in specific piping segments at lower than RCS pressure	Authorized 10 CFR 50.55a(a)(3)(ii)
SPT-005 (Surry 2)	3.5	Residual Heat Removal Piping	B-P	B15.50	100% of Class 1 pressure-retaining boundary is required to be pressure tested at nominal RCS pressure for 100% rated power	Visual VT-2	Perform leakage tests between valves MOV-2700 and -2701 along with other Class 2 portions of this piping at lower than RCS pressure	Authorized 10 CFR 50.55a(a)(3)(ii)
SPT-006 (Surry 2)	3.6	All Pressure Retaining Components	B-P	B15.50 B15.70	100% of pressure-retaining boundary is required to be tested following each refueling outage	Visual VT-2	Conduct pressure testing and VT-2 examinations at beginning of refueling outages	Withdrawn by licensee in letter dated July 19, 2005

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