August 11, 2004

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, UNIT 1 - AMERICAN SOCIETY OF MECHANICAL ENGINEERS INSERVICE INSPECTION PROGRAM FOURTH 10-YEAR INTERVAL REQUESTS FOR RELIEF (TAC NOS. MB7762, MB7763, AND MB9528)

Dear Mr. Christian:

By letter dated December 12, 2002, as supplemented by letters dated December 5, 2003, and February 13, April 20, and June 29, 2004, Virginia Electric and Power Company (VEPCO) requested relief from certain American Society of Mechanical Engineers (ASME) requirements for the fourth 10-year Inservice Inspection Interval (ISI) at Surry Power Station, Unit 1. As part of these relief requests, VEPCO sought to use the 1998 Edition through the 2000 Addenda of the ASME Code, Section XI instead of the 1995 Edition through the 1996 Addenda. In its submittals dated December 5, 2003, and June 29, 2004, VEPCO withdrew Relief Requests CMP-001 through -005, CC-001 through -005, and SPT-002. Subsequently, the Nuclear Regulatory Commission (NRC) staff has completed its review of Relief Requests SPT-003 and SPT-004, and our evaluations and conclusions are contained in the enclosed Safety Evaluation.

The NRC staff has approved VEPCO's proposal to implement the 1998 Edition through the 2000 Addenda of the ASME Code, with the conditions stated in the enclosed Safety Evaluation, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(4)(iv) for the fourth 10-year ISI at Surry, Unit 1. In addition, the NRC staff has reviewed Relief Request SPT-004 and has concluded that complying with the requirements of Section XI of the ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff has concluded that VEPCO's proposed alternative provide reasonable assurance of system leakage integrity. Therefore, Relief Request SPT-004 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval at Surry, Unit 1. Finally, the NRC staff has reviewed Relief Request SPT-003 and has concluded that VEPCO's proposed alternative provides an acceptable level of quality and safety. Therefore, Relief Request SPT-003 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval at Surry, Unit 1.

This completes the NRC staff's activities associated with TAC Nos. MB7762, MB7763, and MB9528. The NRC staff is continuing to review Relief Requests SPT-001 and R-001 under TAC Nos. MB7760 and MB7770.

Sincerely,

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-280

Enclosure: As stated

cc w/encl: See next page

This completes the NRC staff's activities associated with TAC Nos. MB7762, MB7763, and MB9528. The NRC staff is continuing to review Relief Requests SPT-001 and R-001 under TAC Nos. MB7760 and MB7770.

Sincerely,

/RA/

Mary Jane Ross-Lee, Acting Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-280

Enclosure: As stated

cc w/encl: See next page

Distribution

PUBLIC	SCoffin	EDunnington	RidsRgn2MailCenter
RidsNrrDlpmLpdii	GHill	SMonarque	PDII-1 R/F
RidsOgcRp	ACRS		

ADAMS ACCESSION NO. ML042250379 NRR-028

OFFICE	PM/PDII-1	LA/PDII-2	SC/EMCB	OGC	SC/PDII-1
NAME	SMonarque	EDunnington	TChan	HMcGurren	MJRoss-Lee(A)
DATE	7/28/04	7/21/04	7/21/04	8/04/04	8/11/04

OFFICIAL RECORD COPY

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO THE FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

SURRY POWER STATION, UNIT 1

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-280

1.0 INTRODUCTION

By letter dated December 12, 2002, as supplemented by letters dated December 5, 2003, and February 13, April 20, and June 29, 2004, Virginia Electric and Power Company (the licensee) requested relief from certain American Society of Mechanical Engineers (ASME) Code requirements and requirements of Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.55a(g)(4)(ii) for the fourth 10-year Inservice Inspection Interval (ISI) at Surry Power Station, Unit 1 (SPS 1). Specifically, the licensee requested to use the 1998 Edition through the 2000 Addenda of the ASME Code, Section XI instead of the 1995 Edition through the 1996 Addenda. In addition, the licensee requested NRC staff approval of Relief Requests SPT-003 and SPT-004. Relief Request SPT-003 proposed an alternative to the required system leakage testing requirements for approximately 20 small diameter Class 1 reactor coolant system (RCS) pressure boundary vent, drain, and sample and instrumentation connections, and Relief Request SPT-004 proposed an alternative to the required system leakage testing of the reactor pressure vessel lower head.

By letters dated December 5, 2003, and June 29, 2004, the licensee withdrew Relief Requests CMP-001 through -005, CC-001 through -005, and SPT-002.

2.0 REGULATORY EVALUATION

The ISI of the ASME Boiler and Pressure Vessel Code (Code) Class 1, Class 2, and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by 10 CFR 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). As stated in 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the 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 pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for

Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI Code of record for the fourth 10-year interval at SPS 1 is the 1995 Edition through the 1996 Addenda. The fourth 10-year ISI interval at SPS 1 began on October 14, 2003, and ends on December 13, 2013. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

3.0 TECHNICAL EVALUATION:

The licensee proposes to use the 1998 Edition of the ASME Code Section XI through the 2000 Addenda.

3.1 System/Components for Which Relief is Requested

Components inspected as required by the 1995 Edition of the ASME Section XI Code with the 1996 Addenda as modified by 10 CFR 50.55a for the fourth 10-year ISI Interval at SPS 1.

3.2 Code Requirements

10 CFR 50.55a(g)(4)(ii) states: “Inservice examination of components and system pressure tests conducted during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (b) of this section 12 months before the start of the 120-month inspection interval, subject to the limitations and modifications listed in paragraph (b) of this section.”

For SPS 1, the ASME Code, Section IX incorporated by reference in 10 CFR 50.55a(b)(2) 12 months prior to the start of the licensee’s fourth 120-month inspection interval is the 1995 Edition through the 1996 Addenda of the ASME Section XI Code. A revision of 10 CFR 50.55a(b)(2) endorsed the 1998 Edition through the 2000 Addenda and became effective on October 28, 2002.

3.3 Proposed Alternative and Basis for Use

By letter dated December 12, 2002, the licensee proposed to use the 1998 Edition through the 2000 Addenda of the ASME Code, Section XI, as modified by 10 CFR 50.55a. After discussion with the NRC staff, the licensee provided a supplemental letter dated February 13, 2004. In this letter, the licensee committed to the following two conditions when using the 1998 Edition of the ASME, Section XI through the 2000 Addenda as modified by 10 CFR 50.55a:

1) Surry Unit 1 will not invoke paragraph IWA-4340, which allows a defect to remain in a component provided that the defect can be eliminated from the pressure boundary by modification, without prior NRC approval, and

2) Surry Unit 1 will perform pressure tests for Class 1, 2 and 3 mechanical joints undergoing repair and replacement activities in accordance with paragraph IWA-4540(c) included in the 1998 version of the ASME Section XI Code.

3.4 Staff Evaluation

As stated in 10 CFR 50.55a(g)(4)(ii), it is required that ISI, during a given interval, be performed in accordance with the requirements of the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the 120-month inspection interval subject to the limitations and modifications listed therein. As stated in 10 CFR 50.55a(g)(4)(iv), inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in 10 CFR 50.55a(b) and subject to Commission approval. The Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the SPS 1 fourth interval expiration date of October 13, 2003, was the 1995 Edition of the ASME Code through the 1996 Addenda. The 1998 Edition through 2000 Addenda was endorsed by a subsequent revision of 10 CFR 50.55a and became effective on October 28, 2002. Subsequent to the endorsement of the 1998 Edition through the 2000 Addenda in 10 CFR 50.55a, the NRC staff has taken issue with the following two items discussed in the paragraphs below.

Provisions outlined in IWA-4340 were added to the 2000 Addenda in order to provide requirements for the mitigation of defects by "modification." Paragraph IWA-4340 allows a defect to remain in a component provided that the defect can be eliminated from the pressure boundary by "modification." It is the NRC staff's determination that the scope of the activity envisioned or permitted by this subarticle does not provide limitations on the applicability of its provisions to specific ASME Classes or components. As written, this provision could be used in applications with widely varying safety significance and levels of difficulty in implementation. A proposed rule published in the *Federal Register,* Vol. 69, on January 7, 2004 (69 FR 879) seeks to prohibit the use of IWA-4340 when using the 2001 Edition through the 2003 Addenda of the Code. Subsequently, by letter dated February 13, 2004, the licensee agreed to the following condition when using the 1998 Edition though the 2000 Addenda:

> Surry Unit 1 will not invoke paragraph IWA-4340, which allows a defect to remain in a component provided that the defect can be eliminated from the pressure boundary by modification, without prior NRC approval.

The requirements to pressure test Class 1, 2 and 3 mechanical joints undergoing repair and replacement activities were deleted in the 1999 Addenda of ASME Code, Section XI. Therefore, pressure testing of mechanical joints is no longer required by Section XI when performing IWA-4000 repair and replacement activities. It is the NRC staff's determination that there is no justification for eliminating the requirements for pressure testing Class 1, 2, and 3 mechanical joints. The pressure testing of mechanical joints affected by repair and replacement activities is necessary to ensure and verify structural and leakage integrity of the pressure boundary. A proposed rule published in the *Federal Register* Vol. 69, on January 7, 2004 (69 FR 879) seeks to retain the pressure testing requirements in IWA-4540(c) of the 1998 Edition when using the 2001 Edition through the 2003. Subsequently, by letter dated February 13, 2004, the licensee agreed to the following condition when using the 1998 Edition though the 2000 Addenda:

Surry Unit 1 will perform pressure tests for Class 1, 2 and 3 mechanical joints undergoing repair and replacement activities in accordance with paragraph IWA-4540(c) included in the 1998 version of the ASME Section XI Code.

3.5 Conclusion

The NRC staff concludes that the licensee's request to use the 1998 Edition through the 2000 Addenda of the ASME Code, Section IX, with the conditions stated above, in lieu of the 1995 Edition through the 1996 Addenda as required by 10 CFR 50.55a(b)(2) provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(4)(iv), the NRC staff authorizes the use of the 1998 Edition of ASME Section XI through the 2000 Addenda as modified by 10 CFR 50.55a with the following conditions that the licensee agreed to in its letter dated February 13, 2004:

1) Surry Unit 1 will not invoke paragraph IWA-4340, which allows a defect to remain in a component provided that the defect can be eliminated from the pressure boundary by modification, without prior NRC approval, and

2) Surry Unit 1 will perform pressure tests for Class 1, 2 and 3 mechanical joints undergoing repair and replacement activities in accordance with paragraph IWA-4540(c) included in the 1998 version of the ASME Section XI Code.

Approval is granted for the fourth 10-year ISI interval at SPS 1, which began on October 14, 2003, and ends on December 13, 2013. As a result of public comments, should the final rule reflect restrictions or modifications relative to the two conditions that are different to what the licensee has committed, the licensee will only be bound to the requirements as stated in the final rule. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

4.0 TECHNICAL EVALUATION - SPT-003

By letter dated December 12, 2002, as supplemented by letter dated December 5, 2003, the licensee requested approval of Relief Request SPT-003 in order to not test approximately 20 small diameter ($\leq$ 1 inch) vent, drain, and sample lines as part of the RCS pressure test. As an alternative, the licensee proposed to perform an alternative examination each refueling outage.

4.1 System/Components for Which Relief is Requested

ASME Section XI, 1998 Edition through the 2000 Addenda, Examination Category B-P, Items B15.50 and B15.70, Class 1, Pressure-Containing Components. A list of the specific Class 1 RCS pressure boundary vent, drain, sample, and instrumentation lines and connections are shown in the following table.

	DRAWING/ COORDINATES	LINE NUMBER	BOUNDED BY	FUNCTION
1	11448-CBM-083B-3-3/E-7	3/4-RC-165-1502	Flange to # 5	Vent
2	11448-CBM-083B-3-3/E-7	3/4-RC-172-1502	Flange to #6	Vent
3	11448-CBM-083B-3-3/E-7	3/4/-RC-163-1502	Flange to #8	Vent
4	11448-CBM-086A-3-1/D-4	3/4-RC	1-RC-154 to flange	Drain
5	11448-CBM-086A-3-1/F-5	3/4-RC-165-1502	1-RC-9 to #1	Vent
6	11448-CBM-086A-3-2/F-5	3/4-RC-172-1502	1-RC-48 to #2	Vent
7	11448-CBM-086A-3-2/D-4	3/4-RC	1-RC-84 to Flange	Drain
8	11448-CBM-086A-3-3/F-6	3/4-RC-163-1502	1-RC-80 to #3	Vent
9	11448-CBM-086A-3-3/D-7	3/4-RC	1-RC-156 to Flange	Drain
10	11448-CBM-086A-3-3/C-4	3/4-RC-161-1502	1-RC-103 to Flange	Drain
11	11448-CBM-088C-3-1/E-7	3/4-CH	1CH-366 to Flange	Drain
12	11448-CBM-088C-3-2/C-8	3/4-CH	1-CH-325 to Flange	Drain
13	11448-CBM-088C-3-2/C-7	3/4-CH	1-CH-335 to Flange	Drain
14	11448-CBM-088C-3-2/C-6	3/4-CH	1-CH-351 to Flange	Drain
15	11448-CBM-089B-3-4/F-8	3/4-SI-401-ICN9	1-SI-421 to 1-SI-422	Test Connection
16	11448-CBM-089B-3-4/E-8	3/4-SI-403-ICN9	1-SI-424 to 1-SI-425	Test Connection
17	11448-CBM-089B-3-4/D-8	3/4-SI-405-ICN9	1-SI-427 to 1-SI-428	Test Connection

4.2 Code Requirements

ASME Section XI, 1998 Edition through the 2000 Addenda, Examination Category B-P, Items B15.50 and B15.70, Class 1, Pressure-Containing Components, require system leakage testing and associated VT-2 visual examination of all class 1 pressure-retaining piping and valves. IWB-5222(b) requires that “The pressure-retaining boundary during the system leakage test conducted at or near the end of each ISI interval shall extend to all Class 1 pressure-retaining components within the system boundary.”

4.3 Licensee’s Proposed Alternative

As an alternative to the Section XI requirement that once per interval a system leakage test be performed of the normally isolated portions of the subject Class 1 RCS pressure boundary vent, drain, sample, and instrumentation connections, the following is proposed:

1. The RCS vent, drain, instrumentation, and sample connections will be visually examined for leakage and any evidence of past leakage, with the isolation valves in

the normally closed position each refueling outage during the ASME Section Xl Class 1 System Leakage Test (IWB-5220).

2. During operation the RCS will be monitored for leakage and radiation levels in accordance with the requirements of the applicable Technical Specifications.

3. These alternative provisions will only be applied to the inservice testing performed to meet the requirements of Category B-P.

The proposed alternative examination requirements will ensure that the overall level of plant quality and safety will not be compromised. Therefore, approval to use the stated alternative examination requirements is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

4.4 Licensee's Basis for Use

These piping segments are equipped with either two valves, or a valve and end cap that provide for double isolation of the reactor coolant system (RCS) pressure boundary. For each pipe segment, the inboard (i.e. closer to the primary loop piping) or first isolation valve is maintained closed during normal operation; thus, the piping outboard of the first isolation valve is not normally pressurized.

The proposed alternative provides an acceptable level of safety and quality based on the following:

1. ASME Section Xl Code, 1998 Edition with addenda up to and including the 2000 Addenda, paragraph IWA-4540, provides the requirements for hydrostatic pressure testing of piping and components after repairs by welding to the pressure boundary. lWA-4540(b)(6) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Visual examination of these $\leq$1 inch diameter RCS vent/drain/sampling connections once each I0-year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME Xl Code.

2. The non-isolable portion of the RCS vent and drain connections will be pressurized and visually examined as required. Only the isolable portion of these small diameter vent and drain connections will not be pressurized.

3. These piping connections are typically socket welds that received a surface examination after installation.

4. The piping and valves are nominally heavy wall. These piping components and associated piping are towards the free end of a cantilever configuration (stub end isolated by either a valve or a flange). There is no brace or support for this portion of the pipe. Consequently, this portion does not experience any thermal loading.

5. This portion of the line is isolated during normal operation and does not experience pressure loading unless there is a leak at the first isolation valve.

6. The valves do not have an extension operator, so the rotational accelerations at the valve do not produce significant stress.

7. The stresses towards the free end of the cantilever due to other types of loading are only a small fraction of the applicable Code allowable.

The Technical Specifications (TS) require RCS leakage monitoring during normal operation. Should any of the TS leakage limits be exceeded, then SPS 1 is required to identify the source of the leakage and restore the RCS boundary.

During the 1998 North Anna Unit 1 refueling outage, similar piping segments were pressurized by the connection of a test rig. The dose associated with this testing was 1.5 man-rem. It is expected that conditions at SPS Unit 1 would yield comparable exposure results if the testing were performed.

4.5 Staff Evaluation

The Code requires that all Class 1 small diameter ($\leq$1 inch) components (including vent and drain, sample, and instrumentation connections) within the RCS boundary undergo a system leakage test once each refueling outage. The licensee has proposed an alternative to the system leakage test requirements for the subject RCS components. The proposed alternative is to visually examine the RCS components for evidence of leakage each refueling outage during the RCS leakage test with the isolation valves in the normally closed position. Furthermore, during normal plant operations, the licensee would monitor the RCS system for leakage and radiation levels in accordance with the TS.

RCS leakage tests are conducted with the isolation valves in their normally closed position. This means that the portions of those small diameter piping and connections located outboard of the first isolation valve, which are part of the Class 1 pressure boundary, will not meet the Code requirements for testing. As such, the licensee has provided an acceptable basis for not including these unpressurized segments of the piping system as part of the Code-required pressure tests. The affected piping segments typically have socket-welded connections and the welds received a surface examination after installation. Unless the first isolation valve leaks during operation, the affected segments are normally not pressurized. This means that these segments are much less susceptible to corrosion mechanisms that affect the portions of the RCS normally pressurized during operation. In addition, the affected segments are not likely to be subjected to thermal loadings and stresses associated with remote operation operators due to the segment configurations.

For the licensee to perform the Code-required test, it would be necessary to pressurize the affected segment by connecting a test rig, or opening the first valve at normal system operating pressure, thereby eliminating the double isolation from the RCS boundary. Pressurization by this method would cause safety concerns for the personnel performing the examination due to the close proximity to the primary RCS piping. The licensee states that testing by this method would expose plant personnel to an estimated 1.5 man-rem.

The licensee proposed to visually examine the isolation valves for evidence of present and past leakage during the system leakage test that is performed each refueling outage. Also, the RCS vent and drain connections will be visually examined with the isolation valves in the normally

closed position during the 10-year ISI pressure test. The proposed alternative examination will provide reasonable assurance of system leakage integrity and thus provide an acceptable level of quality and safety.

4.6 Conclusion

The NRC staff concludes that the licensee's proposed alternative is acceptable and provides an acceptable level of quality and safety. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval for SPS 1. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

5.0 TECHNICAL EVALUATION - SPT-004 REV.1

In its submittal dated April 20, 2004, the licensee requested approval of Relief Request SPT-004 Rev. 1 to use an alternative that satisfied the visual examination requirements related to the reactor pressure vessel (RPV) lower head penetrations.

5.1 System/Components for Which Relief is Requested

ASME Section XI, 1998 Edition through the 2000 Addenda, Examination Category B-P, Item B15.10, Class 1, Pressure-Retaining Boundary, Reactor Vessel, Partial Penetration Welds at the Bottom of the Reactor Vessel.

5.2 Code Requirements

ASME Section XI, 1998 Edition through the 2000 Addenda, Examination Category B-P, Item B15.10, Class 1 requires a VT-2 visual examination during system leakage testing of the pressure-retaining boundary of the reactor vessel prior to startup following a reactor fueling outage.

5.3 Proposed Alternative

The licensee proposes to perform a VT-2 bare-metal visual examination each refueling outage for evidence of boric acid leakage/corrosion on the bottom of the reactor vessel when the containment is at atmospheric conditions. Additionally, the licensee stated that it will continue to perform the TS Surveillance Requirements that monitor RCS leakage and radiation levels. The licensee indicated that the incore sump room has a level alarm in the control room requiring operator action. In the event of a leak, these actions would identify any integrity concerns associated with this area.

The licensee stated that the combination of monitoring methods and VT-2 bare-metal visual examination performed each refueling outage provides an acceptable level of quality and safety. Because of the burden and potential safety challenges caused by the subatmospheric conditions of the containment, the Code-required examinations at the bottom of the reactor vessel during system leakage tests result in a hardship without a compensating increase in quality and safety over the proposed alternative. Therefore, the licensee seeks approval of this request for relief in accordance with 10 CFR 50.55a(a)(3)(ii).

5.4 Licensee's Basis for Relief

To meet the Section XI pressure and temperature requirements for the system leakage test of the reactor vessel, the SPS 1 reactor containment is required to be at subatmospheric pressure. Station administration procedures require that self-contained breathing apparatus must be worn for containment entries under these conditions. This requirement significantly complicates the visual (VT-2) examination of the bottom of the reactor vessel during testing. Access to the bottom of the reactor vessel requires the examiner to descend several levels by ladder and navigate the entrance leading to the reactor vessel. In addition to the physical constraints, the examiner must contend with extreme environmental conditions: elevated air temperatures due to reactor coolant at temperatures above 500 degrees F and limited air circulation in the vessel cubicle. Also, the limited capacity of the breathing apparatus further encumbers the performance of the examination.

These factors increase the safety hazard associated with the examination. As a minimum, the examiner is forced to perform the examination under considerable physical burden. To place the examiner under this increased risk and burden is not justifiable. This combination of conditions does not exist during the refueling outage when the proposed alternative examination would take place. The proposed alternative examination would be performed under conditions that are safer and allow for a more thorough examination.

5.5 Staff Evaluation

System leakage tests are performed on the RPV each refueling outage in accordance with ASME Code, Section XI, Table IWA-2500-1, Examination Category B-P, Item B15.10. The Code requires that a VT-2 visual examination shall be conducted by examining the accessible external exposed surfaces of pressure-retaining components for evidence of leakage. The Code-required VT-2 visual examination can be performed on bare metal surfaces or insulation-covered surfaces. During system leakage tests, the VT-2 visual examination is conducted in accordance with IWA-5211. System leakage tests are conducted at a test pressure not less than the nominal operating pressure associated with 100-percent rated reactor pressure.

The licensee is proposing to perform an alternative inspection method on the same inspection frequency as required in Table IWB-2500-1. This proposed inspection will look for evidence of leakage similar to what would be performed under the Code requirements but while SPS 1 is in a refueling outage and the containment is at atmospheric conditions. In addition, the licensee has committed to performing the VT-2 bare-metal visual inspection on the RPV lower head surface instead of performing the Code-required inspections.

The NRC staff agrees that the environment in the containment during system leakage tests makes inspecting the RPV lower head penetrations very difficult. The licensee's proposed alternative would allow inspectors to perform an inspection in an environment where they are not encumbered by high temperatures and subatmospheric conditions that would be present while the plant is at nominal operating temperature, thus allowing time for a more thorough inspection. Based on the NRC staff's experience from recent RPV penetration nozzle leakage discovered at other plants, the NRC staff believes that a bare-metal visual examination will

reveal a leaking RPV penetration nozzle before a Code-required VT-2 examination that is normally conducted with the insulation in place. Performing the Code-required inspection would result in a hardship for the licensee without a compensating increase in the level of quality and safety. The NRC staff has determined that the proposed alternative examination would provide reasonable assurance of system leakage integrity.

5.6 Conclusion

The NRC staff concludes that conducting a Code-required VT-2 examination of the lower RPV head when the containment is at sub-atmospheric conditions would result in hardship without a compensating increase in the level of quality and safety. The licensee's proposed alternative of performing a VT-2 bare-metal visual examination when the containment is under atmospheric conditions provides reasonable assurance of system leakage integrity. Therefore, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the fourth 10-year ISI interval for SPS 1. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

6.0 REFERENCES

1. Letter dated December 12, 2002, L.N. Hartz (Virginia Electric and Power Company, Vice President, Nuclear Engineering Services) to U.S. Nuclear Regulatory Commission, containing *Surry Power Station Unit 1, Fourth Interval Inservice Inspection Program Relief Request*.

2. Letter dated December 5, 2003, L.N. Hartz (Virginia Electric and Power Company, Vice President, Nuclear Engineering Services) to U.S. Nuclear Regulatory Commission, containing *Surry Power Station Unit 1, Fourth Interval Inservice Inspection Program Request For Additional Information.*

3. Letter dated February 13, 2004, L.N. Hartz (Virginia Electric and Power Company, Vice President, Nuclear Engineering Services) to U.S. Nuclear Regulatory Commission, containing *Surry Power Station Unit 1, Fourth Interval Inservice Inspection Program Commitments.*

4. Letter dated April 20, 2004, L.N. Hartz (Virginia Electric and Power Company, Vice President, Nuclear Engineering Services) to U.S. Nuclear Regulatory Commission, containing *Surry Power Station Unit 1, Fourth Interval Inservice Inspection Program Request for Additional Information*.

5. Letter dated August 21, 2001, U.S. Nuclear Regulatory Commission to D.A. Christian (Virginia Electric and Power Company, Sr. Vice President- Nuclear), containing *Surry Power Station Units 1 and 2, Inservice Inspection Program Relief Request.*

Principal Contributor: Robert Davis

Date: August 11, 2004

cc:

Ms. Lillian M. Cuoco, Esq.
Senior Counsel
Dominion Resources Services, Inc.
Building 475, 5th Floor
Rope Ferry Road
Rt. 156
Waterford, Connecticut 06385

Mr. Richard H. Blount, II
Site Vice President
Surry Power Station
Virginia Electric and Power Company
5570 Hog Island Road
Surry, Virginia 23883-0315

Senior Resident Inspector
Surry Power Station
U. S. Nuclear Regulatory Commission
5850 Hog Island Road
Surry, Virginia 23883

Chairman
Board of Supervisors of Surry County
Surry County Courthouse
Surry, Virginia 23683

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, Virginia 23218

Robert B. Strobe, M.D., M.P.H.
State Health Commissioner
Office of the Commissioner
Virginia Department of Health
P.O. Box 2448
Richmond, Virginia 23218

Office of the Attorney General
Commonwealth of Virginia
900 East Main Street
Richmond, Virginia 23219

Mr. Chris L. Funderburk, Director
Nuclear Licensing & Operations Support
Dominion Resources Services, Inc.
Innsbrook Technical Center
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Mr. Jack M. Davis
Site Vice President
North Anna Power Station
Virginia Electric and Power Company
P. O. Box 402
Mineral, Virginia 23117-0402

Mr. William R. Matthews
Vice President - Nuclear Operations
Virginia Electric and Power Company
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, Virginia 23060-6711