VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

10 CFR 50.55a

May 5, 2004

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D. C. 20555

Serial No.	03-428A
NL&OS/GDM	R1
Docket No.	50-281
License No.	DPR-37

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 2
FOURTH INTERVAL INSERVICE INSPECTION PROGRAM
REQUEST FOR ADDITIONAL INFORMATION

In a letter dated August 25, 2003 (Serial No. 03-428) Virginia Electric and Power Company (Dominion) submitted the inservice inspection (ISI) program for the fourth inservice inspection interval for Surry Unit 2 for Class 1, 2, and 3 components and component supports. During the course of their review, the NRC staff identified the need for additional information to facilitate the completion of their review. On January 28, 2004, the Surry NRC Project Manager provided the staff's questions associated with the overall fourth interval ISI program submittal and the relief requests included therein. These questions and our proposed responses were discussed during a conference call held on February 12, 2004. At the conclusion of the conference call, we agreed to provide a written response to the NRC's questions. Accordingly, Dominion's response to the staff's questions is provided in the attachment.

If you have any questions or require additional information, please contact Mr. Gary Miller at (804) 273-2771.

Very truly yours,

LNHartz

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachment

Commitments made in this letter: None

A047

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Attachment

Response to NRC Request for Additional Information

Fourth Interval Inservice Inspection Program
Surry Power Station Unit 2

Virginia Electric and Power Company
(Dominion)

ATTACHMENT

Response to NRC Request for Additional Information
Fourth Interval Inservice Inspection Program

Surry Power Station Unit 2

1. *Relief Request on the Section XI Code Edition to be used for the Fourth 10-Year Inservice Inspection Interval*

   Virginia Electric Power Company (VEPCO, the licensee) states that the fourth 10-year inservice inspection (ISI) interval at the Surry Power Station, Unit 2 (Surry 2) is scheduled to begin on May 10, 2004. The ISI program has been developed in accordance with the 1998 Edition through the 2000 Addenda of ASME Section XI. VEPCO cites 10 CFR 50.55a(g)(4)(iv) which allows the use of later editions, addenda or portions thereof, of ASME Section XI that have been incorporated by reference in paragraph 10 CFR 50.55a(b), subject to approval by the NRC staff. VEPCO has requested approval to use the 1998 Edition with 2000 Addenda of ASME Section XI for the fourth 10-year ISI interval.

   Paragraph 10 CFR 50.55a(g)(4)(ii) states that inservice examination of components during successive inspection intervals must comply with the latest edition and addenda that have been incorporated by reference in paragraph (b) of the section 12 months prior to the start of the inspection interval, subject to limitations in (b). Since the fourth interval at Surry 2 begins May 10, 2004, the latest Code referenced in 10 CFR 50.55a(s)(3)(b) 12 months prior to that date is the 2000 Addenda (this is found in the 2003 revision of CFR). Explain why NRC staff approval is required for the use of ASME Section XI, 1998 Edition with 2000 Addenda.

   Dominion Response

   We agree that NRC staff approval is not required to use ASME Section XI, 1998 Edition with 2000 Addenda. As the fourth interval at Surry Unit 2 begins May 10, 2004, the latest Code referenced in 10 CFR 50.55a(s)(3)(b) 12 months prior to that date is the 2000 Addenda. Therefore, Surry Unit 2 will use ASME Section XI, 1998 Edition with 2000 Addenda in accordance with regulatory requirements.

2. *Request for Relief CMP-001. Examination Category B-D, Item B3.120, Full Penetration Welded Nozzles in Vessels, Pressurizer Surge Nozzle Inside Radius Section*

   A. *VEPCO stated that any ultrasonic examination of the pressurizer surge nozzle could only be described as "best effort," and that a remote visual examination, conducted from the inside of the pressurizer, has very little probability of success. Similar statements were made when requesting the same relief during previous* *inspection intervals. Given that advancements in ultrasonic testing and remote visual technologies have been made since the previous request, describe what steps have been taken by the licensee to improve the level of inspection for the pressurizer surge nozzle. In addition, provide detailed drawings that show cross-sectional view of the surge nozzle, thermal sleeve, and basket diffuser. The staff requests that these drawings include a list of the material specifications, dimensions of the components, and clearly indicate the interferences on the outside of the vessel for performing ultrasonic examination.*

B. VEPCO has provided a basis to support a determination of hardship; however, further information is needed in order for the NRC staff to arrive at reasonable assurance of continued structural integrity for this component. VEPCO states that the calculated cumulative usage factors for operational and design transients in the surge nozzle inner radius are 0.29 and 0.11, for inside and outside surfaces, respectively, and that these values are less than the design limit and provide insight into the potential for failure in this region. Please elaborate on what insights may be derived from the analyses, primarily from the point of view of expected degradation mechanisms and the probability of failure that these mechanisms present, based on operational considerations.

C. VEPCO has stated that the alternative to volumetric examinations will be the Code-required visual VT-2 examinations performed in conjunction with system leakage tests during each refueling outage. Please describe if any augmentation of the visual VT-2 examination will be employed specifically for the surge nozzle, if the Code-required volumetric examination is eliminated.

D. VEPCO also states that Technical Specifications (TS) surveillance requirements related to reactor coolant leak rates and containment atmospheric radioactivity will be satisfied. However, based on recent industry events such as the primary coolant leak at VC Summer, it is unclear whether simply meeting TS is sufficient to indicate that a significant leak is occurring. Please describe any other alternatives the licensee has considered to indicate that a leak associated with the pressurizer surge nozzle may be occurring. In your response, specifically address whether VEPCO has considered any special instrumentation for this region for monitoring potential leakage from the pressurizer surge nozzle or for detecting the containment atmospheric radioactivity levels in the vicinity of the pressurizer surge nozzle.

<u>Dominion Response</u>

We request that Relief Request CMP-001 be withdrawn at this time with the intent of resubmitting a revised relief request in the near future after completion of further research and the receipt of additional design data and drawings from the NSSS supplier.

3. *Request for Relief CMP-002, Examination Category C-G, Item C6.10, Pressure Retaining Welds in Pumps and Valves, Pump Casing Welds*

Please provide drawings of the pumps that detail the location of welds, pump assemblies, and other support components, that limit remote visual examination of the inside surface of the subject welds. In VEPCO's alternative, it is stated that remote visual VT-1 examination of the inside surface of the welds will be performed if the pump is disassembled for maintenance. Please discuss whether the disassembly of these pumps is expected to occur during the fourth inspection interval at Surry 2. Also, describe any inspection history that may have occurred in previous intervals, and the results of these inspections. VEPCO also states that some welds are partially accessible. Provide a list of partially accessible welds and the expected completion percentages for these welds. Please discuss the degradation mechanisms expected to occur to the pump casings, and how the proposed alternative reasonably ensures the structural integrity of these pressure boundaries.

Dominion Response

We request that Relief Request CMP-002 be withdrawn. Upon further investigation of code requirements for this Category C-G inspection and review of IWC-1220 Code Exemption changes, this relief is not required. The code required examinations will be performed upon disassembly of the pumps for maintenance.

4. *Request for Relief CMP-003, Appendix I, Article I-2000, Calibration Blocks for Ultrasonic Examination*

VEPCO has provided a few examples of how the existing calibration blocks at Surry 2 deviate from Code requirements. Please provide a comprehensive list of the calibration blocks found to be out of compliance with the Code, describe the features that make these blocks noncompliant, and discuss how these features will affect the ultrasonic calibrations and examinations performed at Surry 2.

Dominion Response

This relief request is being withdrawn based on the NRC's previous response to a similar relief request for North Anna Unit 2. In the safety evaluation included in the letter dated June 12, 2002, for the North Anna Unit 2 relief request, the NRC stated that, "the ASME Code already provides a means of considering the use of alternative calibration blocks under the provisions of IWA-2240. Thus, the licensee's implementation of IWA-2240 regarding the application of alternative calibration blocks obviate the need for this relief request."

5. <u>*Request for Relief CMP-004, IWA-2600, Weld Reference System*</u>

Please state the specific Code requirement for which an alternative is proposed. Also, describe in detail the hardship or unusual difficulty that would be incurred, if required to meet the Code requirements. As an alternative, further describe the licensee's existing system and how this system provides an acceptable level of quality and safety.

<u>Dominion Response</u>

IWA-2620, "Piping," refers to the requirements in III-4300. III-4330, "Reference System," in the 1999 Addenda states "Circumferential and longitudinal welds requiring volumetric examination shall be marked once before or during the preservice examination to establish a reference." The original construction code at Surry did not require preservice marking. Rather than attempting to go back and stamp every Section XI weld in the plant (approximately 6000 welds), which would be a considerable hardship, Surry is instead marking the welds when performing the first inspection requirement and has been doing so since the beginning of the third inservice inspection interval. With the exception of weld additions due to code changes and updates, the weld population to be inspected should be appropriately marked at this time. However, weld selections are sometimes exchanged for other suitable choices due to changes in exposure control, accessibility or implementation of new programs such as the Risk-Informed Inservice Inspection program. Thus, the possibility exists that an unmarked weld may be encountered. At this point in the program, marking all welds in the ISI population at this point before an inspection requirement becomes due would not provide any additional, useful information and would create a significant and unnecessary burden of labor and radiation exposure to obtain information for weld locations that may never require examination.

Surry is required by procedure to mark reference points upon inspection and establish the datum point during preservice examination for new welds. A permanent datum point is denoted by the capital letter "T", with the cross of the "T" located at the zero reference point with the leg of the "T" lying on the weld centerline pointing in the 7 direction. Four scans made for UT examination are denoted as follows: 2-downstream of weld, 5-upstream of weld, 7-clockwise to system flow and 8-counterclockwise to system flow. The location of indications is reported with relation to the datum point or other identifiable reference point.

6. <u>*Request for Relief CMP-005, IWA-2600 Weld Reference System, Automated Reactor Vessel Examinations*</u>

A. *Please state the specific Code requirement for which an alternative is proposed. Also, describe in detail the hardship or unusual difficulty that would be incurred, if required to meet the Code requirements. Include information pertaining to making permanent location markers on the inner surface of the vessel, as well as, for future* *examinations or characterization of detected flaws, whether any examinations may be necessary from the outer surface of the vessel that would require location markers.*

B. VEPCO stated that the automated tool establishes a zero point on the reactor pressure vessel (RPV) during each examination, and that electronic encoders on the automated system provide sufficient repeatability. Please describe the tolerances, or location error, that may be expected with the automated system.

C. Clarify whether VEPCO will implement the methods in recently published ASME Code Case N-613-1, or other Code Cases that address proposed reductions of examination volumes, during the fourth inspection interval at Surry 2. If so, discuss how the location accuracy of the automated Inspection tool, combined with the validity of as-built drawings of the RPV welds, will ensure that 100% of the reduced volumes are being inspected.

Dominion Response

A. The Code requirement for which the alternative is proposed is IWA-2620. Due to the extreme high dose involved in performing NDE inspections on the reactor vessel, an automated tool will be utilized that permits examination personnel to monitor work from a remote location. This advanced ultrasonic examination technology does not have the capability of permanently marking the welds. However, a repeatable reference system will be established for the examinations using permanent vessel landmarks, such as numbered reactor vessel bolt holes and known equipment setup orientation. The response to Item B below provides additional detail.

B. Our presently secured contractor, who will be performing our reactor vessel ten-year inspection on Surry Unit 1 during the Fall 2004 refueling outage using the automated tool, provided the information below. Dominion expects as good or better accuracy and tolerances in future ten-year inspections.

Positioning Accuracy

The positional accuracy of the tool by design is specified as ± 6.35mm. This accuracy was demonstrated to be within ± 6mm in a series of dry and wet tests. This accuracy is sufficient to ensure the proper placement of the probe package on the component surface and to meet the required tolerances on defect location.

To provide this positioning accuracy during the inspection process, a pre-examination calibration process is conducted. The pre-examination calibration process for each robot includes an axis zeroing step in the equipment check-out procedure and an in-vessel environmental check. The intent of these checks is to ensure that the nozzle azimuth and radial position from the vessel centerline as specified in the robot controller are consistent with the actual vessel. This is essentially a calibration of the pre-site vessel and robot model to the actual vessel conditions. This calibration is repeated every time the robot arm is re-installed and after trouble-shooting of the robot controller has occurred.

Known landmarks, within the reactor vessel and on the manipulator, are used to establish the in-vessel environmental check. For the upper robot and the Zone 1 and Zone 2 scans of the nozzle to safe end weld, a nozzle azimuthal reference includes the top dead center of the nozzle. Radial position references include the corner of an outlet nozzle protrusion and the center of an inlet nozzle inner radius.

The end effector on the robot is moved with respect to these references. Positional differences between the robot readings and the known vessel location are input into the manipulator model. This process adjusts the robot and model to the vessel. Acceptable tolerances on azimuth and elevation are ±5mm.

Positioning Repeatability

The positional repeatability of the tool by design is specified as ± 6.35mm. This repeatability was demonstrated to be within ± 6mm in a series of dry and wet tests. This accuracy is sufficient to re-locate a detected flaw and to re-scan a specified region of the reactor vessel.

To provide this repeatability accuracy during the inspection process, a pre-examination calibration process is conducted. The pre-examination calibration process for each robot includes an axis zeroing step in the equipment check-out procedure and an in-vessel environmental check. The intent of these checks is to ensure that the nozzle azimuth and radial position from the vessel centerline as specified in the robot controller are consistent with the actual vessel. This is essentially a calibration of the pre-site vessel and robot model to the actual vessel conditions. This calibration is repeated every time the robot arm is re-installed and after trouble-shooting of the robot controller has occurred.

C. As discussed above the positioning accuracy and repeatability of the vessel inspection tool is very good and would support use of Code Case N-613-1 reduced volume examinations. Dominion has not determined at this time if use of Code Case N-613-1 will be necessary in the fourth inspection interval. A request to use this Code Case would be separately submitted later, if determined necessary, or its use would be made in accordance with regulatory requirements if the Code Case is subsequently included in Regulatory Guide 1.147.

7. *Request for Relief CMP-006, Examination Categories B-B, B-D and C-A, Pressure Retaining Welds on the Regenerative Heat Exchanger*

A. *VEPCO has requested relief from Examination Category B-B, B-D and C-A requirements for welds on Regenerative Heat Exchanger 2-CH-E-3. The welds associated with each of these categories have been identified in CMP-006 Section I and Figure CMP-006-1. Figure CMP-006-1 also shows Class 2 nozzle-to-shell welds 1-05, 1-07, 1-10, 1-12, 1-14, and 1-16, The information provided in the request suggests that these are Category C-B welds and would also be subject to the examination requirements specified in Table IWC-2500-1. However, VEPCO has not included these Class 2 nozzle-to-vessel welds as being within the scope of the request, nor identified any dose burden associated with the examination requirements for these welds. Please identify the Code Examination Category(s) and examinations being performed for nozzle-to-shell welds 1-05, 1-07, 1-10, 1-12, 1-14, and 1-16 and provide additional explanation regarding the radiation dose burden associated with these welds.*

B. *Provide detailed drawings that show cross-sectional views of the nozzle-to-vessel, head and shell welds included in this request. The staff requests that the drawings include a list of the materials' specifications, dimensions of the components, and clearly indicate interferences for performing ultrasonic examinations.*

C. *The Class 1 welds on 2-CH-E-3 can be subject to thermal fatigue loading associated with the loss and subsequent re-initiation of letdown/charging. Design basis transient loadings for these conditions and assumed transient cycle occurrences are generally evaluated as part of ASME Section III Class 1 fatigue analyses. Please identify the design basis cumulative fatigue usage factors associated with Category B-B and B-D welds and discuss the plant's operating occurrences for these design basis type events and their magnitude relative to the design basis transient profiles assumed in the regenerative heat exchanger design report.*

Dominion Response

Revision of this relief request to address the NRC's questions will require further research and will likely require Dominion to contact the heat exchanger manufacturer to obtain more detailed design information. In addition, Dominion is currently monitoring the development of a Code Case relative to this issue. Consequently, we request that Relief Request CMP-006 be withdrawn at this time with the intent of submitting a revised relief request at a later date.

8. *Request for Relief SPT-001, Examination Category B-P, System Leakage Tests for Class 1 Pressure Retaining Components*

   A. *System leakage tests are required to be performed at a pressure corresponding to nominal operating pressure. VEPCO has elected to perform these during the return-to-power sequence at the end of each refueling outage. Because of the sub-atmospheric design of containment at Surry 2, these examinations are performed just prior to reactor start-up by personnel wearing self-contained breathing apparatus (SCBA). The Code requires a maximum examination distance (six feet - Table IWA-2210-1) for direct visual VT-2 examinations. With respect to the contents of Relief Request No. SPT-001 clarify whether VEPCO is proposing to establish a new maximum distance for direct VT-2 examinations or to perform the system leakage tests without erection of temporary scaffolding, or both.*

   B. *If VEPCO confirms that it is proposing to establish a new maximum distance for direct VT-2 examinations, VEPCO states that the visual VT-2 examination maximum distance has been qualified at Surry 2 to extend to nine-feet, nine-inches. Please cite Code references for allowing this new distance qualification. In addition, further describe the qualification process, including all parameters and limitations used for this qualification (e.g., the minimum lighting conditions required, the visual standard used, pass/fail criteria, etc.).*

   C. *On the matter of temporary scaffolding, VEPCO states that any components which cannot be accessed from permanent structures, or with ladders, will be deemed "inaccessible," and the surrounding area (including the floor or equipment surfaces located underneath these components) will be examined for evidence of leakage. This is allowed by Code under IWA-5241(b) and IWA-5242(b), for non-insulated and insulated components, respectively. The NRC staff understands that it may not be feasible, given seismic constraints and personnel hardship considerations, to erect temporary scaffolding to facilitate VT-2 examinations in all areas. Based on the previous discussion, NRC staff approval concerning the installation of temporary scaffolding may not be required, however, the NRC staff requests that VEPCO provide further clarification on the following items:*

      a) *Based on direct visual VT-2 access limitations, clarify what percentage of all components in Class 1 systems will be considered "inaccessible."*

      b) *Describe the primary system locations that are inaccessible for direct visual VT-2 examination.*

      c) *Confirm for the subject system leakage tests, that 10-minute (uninsulated components) and 4-hour (insulated components) hold times will be applied prior to performing the visual VT-2 examinations.*

Dominion Response

A. Dominion will comply with the direct VT-2 maximum distance of 6 feet in accordance with table IWA-2210-1. The distance from 6 feet to 9 feet - 9 inches was qualified as a remote visual examination with our Authorized Nuclear Inservice Inspector (ANII) using no visual aids per IWA-2210(c). (Note: the examination is performed using air masks preventing the use of binoculars and other such equipment). The qualification made use of a near-distance vision test chart per IWA-2210(b) and a light meter verifying 15 foot-candles. The relief request only pointed out that the examiner could effectively examine out to 9 feet, 9 inches using direct and remote means. The concern is that to meet the direct or remote criteria while wearing air masks, the installation of temporary scaffolding would be required in some instances.

B. As noted in the response to 8.A above, Dominion is not proposing to establish a new maximum distance for direct VT-2 examinations.

C.(a) and (b)

Station drawings 11548-FM-1E, F, and G are provided in Enclosure 1 as references to aid the discussion of distances from the Class 1 components to the examiner. The Class 1 components are primarily located within containment, which is sub-atmospheric during the Class 1 system leakage test. Systems that include Class 1 components are reactor coolant, safety injection, charging, sampling, and residual heat removal. The locations where temporary scaffolding would be needed for a direct VT-2 exams are the containment basement, the containment loop room, the pressurizer (bottom location), the pressurizer (upper location), and the reactor head area. The percentage estimates assume an average 5 feet eye distance from floor combined with the 6' direct visual requirement.

The containment basement floor is located at elevation –27'7" (drawings 1E, 1F, and 1G). The pipe support racks are located between elevations –18'7" and –6'5" (drawings 1E, 1F, and 1G with elevations listed on 1E). The components being examined are almost entirely located in the pipe racks or the basement overhead. As such, these components for the most part would exceed the 6 feet direct visual criteria for a VT-2 exam. The affected components are associated with the reactor coolant, safety injection, charging, sampling, and residual heat removal systems. In the aggregate, the estimated inaccessibility of these components is > 90%.

The containment loop rooms are located between elevations –3'6" and gratings located at 16'0" and 20'0" (drawings 1E, 1F, and 1G). Limitations exist when examining the components from above (i.e., from the grating) in the reactor coolant pump cubicle due to distance and obstructions and also in meeting the 6 feet direct VT-2 examination requirement from below in the loop room. The affected components include portions of the steam generator, reactor coolant pump, and piping components associated with the reactor coolant, safety injection, charging, and sampling systems. In the aggregate, the estimated inaccessibility of these components is >50%.

The pressurizer (lower portion, drawing 1E) is located between elevations 18'4" and 44'0." Additionally, piping components associated with pressurizer spray are located in this area. As these components are essentially vertical, approximately half of the surface areas would exceed the 6 feet direct visual VT-2 examination criterion resulting in an estimated inaccessibility of >50%.

The pressurizer (upper portion, drawing 1E) is located between elevations 47'4" and 68'0." Additionally, safety valve and power operated relief valve piping and level instrumentation are located in this area. A ladder exists in this area allowing access to the top of the pressurizer; however, the sub-atmospheric conditions and personnel safety concerns preclude the use of the ladder during the system leakage test. The piping is located at the top of the pressurizer and would exceed the 6 feet direct visual VT-2 examination requirement. Approximately, one-third of the pressurizer exceeds the 6 feet visual VT-2 requirement resulting in an estimated inaccessibility of >33%.

The reactor vessel head is examined from the 47' foot elevation level during the system leakage test. The reactor head is located between elevations 18'4" (drawing 1F) and 23'7" (drawing 1E). The examination vantage point is not directly in-line with the vessel head from the 47' level; consequently, the exam must be performed at an angle and approximately 25 to 30 feet from the head. Therefore, reactor vessel head inaccessibility is 100%.

It should be noted that the areas discussed above would be examined indirectly looking for leakage at low points of vertical runs and on the floor beneath, in addition to the extended visual exams discussed above. Additionally, the reactor vessel head receives augmented examinations as directed by recently imposed NRC requirements.

(c) The 10 minute (uninsulated) and 4-hour (insulated) hold times will be applied prior to performing the visual VT-2 examinations.

9. *Request for Relief SPT-002, Examination Category B-P, System Leakage Tests for Class 1 Small Diameter Vent and Drain Piping*

VEPCO states that the proposed alternative includes approximately 20 connections to the reactor coolant system, all 1-inch or less in diameter, and among these connections are system vent, drain, sample, and instrumentation lines. However, in the basis for relief, VEPCO refers to vent and drain configurations with double isolation. It is unclear whether sample or instrumentation lines, which may not have double isolation, should be included in the proposed alternative, Please list each *component item for which this alternative is intended. For each item included in the list, provide a discussion of the function of this item and indicate whether a double isolation valve configuration is present and whether the first isolation valve is normally configured in the closed position. Also, indicate whether there exists any Inconel 600 materials in the subject connections and clarify whether this relief request proposal includes penetrations of primary system vessels, such as the reactor pressure vessel, steam generators, or pressurizer. In addition, further describe the burden associated with the performance of pressurization and visual VT-2 examination of the non-isolable portions of these connections.*

Dominion Response

The table below provides the detailed information requested by the NRC for Class 1 small diameter vent and drain piping.

SEG	MARK NUMBER / DESCRIPTION	FUNCTION	DOUBLE ISOLATION VALVE	1st VALVE NORMALLY CLOSED	PENETRATES PRIMARY SYS VESSEL	INCONEL 600
1	2-RC-154 on 3/4" line	Vent	No	Yes	No	No
2	2-RC-48 on 3/4" line to blank flange	High Point Vent	No	Yes	No	No
3	2-RC-80 on 3/4" line to blank flange	High Point Vent	No	Yes	No	No
4	2-RC-156 on 3/4" line	Stand Pipe Vent	No	Yes	No	No
5	3/4" line between 2-RC-104 and 2-RC-173	Stand Pipe	No	Yes	No	No
6	1" line between 2-RC-36, 2-RC-186, 2-RC-FNG-545A	Reactor Vessel Vent	No	Yes	No	No
7	3/4" line between 2-RC-105, 2-RC-106 and 2-RC-157	Vessel Level Stand Pipe	No	Yes	No	No
8	3/4" line downstream of 2-RC-106	Test Connection	Yes	Yes	No	No
9	3/4" line downstream of 2-RC-103	Test Connection	No	Yes	No	No
10	3/4" line between 2-RC-170 and 2-RC-138	Stand Pipe Vent to Pressurizer	No	Yes	No	No
11	3/4" line between 2-SI-411 and 2-SI-412	Test Connection	Yes	Yes	No	No
12	3/4" line between 2-SI-414 and 2-SI-415	Test Connection	Yes	Yes	No	No
13	3/4" line between 2-SI-417 and 2-SI-418	Test Connection	Yes	Yes	No	No

To test the configurations included in the table above, an operator would need to open the normally closed valve after reaching test pressure and temperature or reconfigure the valve closed if left open during test pressurization at the start. In either case the operator would be changing valve position while the reactor coolant system was in a high temperature (>500°F) and high-pressure condition (>2200 psig.) with the associated hazards. The test is also performed while the containment is sub-atmospheric, which would require the operator to wear a self-contained breathing apparatus. The test requires valve manipulations under the associated elevated containment air temperature and humidity conditions. Alternatively, a test rig from the backside of the connection could test the areas in question. This type of testing can be performed while the containment is atmospheric and at normal environmental conditions; however, this alternative requires more set-up and preparation time, which would subject test and support personnel to increased personnel exposure. (A similar approach was used at North Anna and resulted in personnel exposure of approximately 1.5 man-rem.) Additionally, this testing would not test the end cap or end flange connection, which would be re-installed following removal of the test connection.

10. *Request for Relief SPT-003. Examination Category B-P, System Leakage Tests for Pressure Retaining Partial Penetration Welds on the Reactor Pressure Vessel, Bottom Mounted Instrumentation Nozzles*

A. *In lieu of the visual VT-2 required to be performed at normal operating pressure, VEPCO has proposed to examine the bottom mounted instrument (BMI) nozzles on the reactor pressure vessel during each refueling outage, but when the system is not at normal operating pressure (i.e., during cold shutdown) and to use evidence of leakage or corrosion (e.g., indications of the presence of boric acid residue) as the basis for detecting active leakage at these BMI locations. VEPCO has also summarized the environmental conditions (e.g., temperature, confined spaces, limited airflow, and sub-atmospheric conditions) that would be experienced by personnel if the personnel were required to perform these examinations at normal operating pressure. However, no radiation exposure levels have been discussed. Please provide estimates for personnel radiation exposure, if required to perform the examinations per the Code requirements.*

B. *VEPCO's alternative in Relief Request SPT-003 is to examine the BMI areas during each refueling outage, but only when containment is at atmospheric conditions (i.e., during cold shutdown). Please state whether a direct or remote visual VT-2 "bare-metal" examination of the reactor pressure vessel BMI penetrations will be performed during each refueling. If a direct visual examination is not performed, describe the parameters under which the remote examinations will be performed, including the type and rigor of the examinations, extent of components that will be examined, and the evaluation criteria to be used. State how the presence of boric acid and corrosion products from other sources will be differentiated from active leak(s) at BMI penetrations.*

Dominion Response

A. Radiation exposure is not a primary concern for this relief. The estimated dose for a ten-minute inspection would be 8.3 mrem. The primary consideration is the hardship involved with the physical constraints created by the self-contained breathing apparatus as discussed in the relief.

B. In a letter dated September 22, 2003 (Serial No. 03-459), Dominion responded to NRC Bulletin 2003-02, "Leakage from Reactor Pressure Vessel Lower Head Penetrations and Reactor Coolant Pressure Boundary Integrity," for Surry Units 1 and 2. The response for Surry Unit 2 that was provided in Attachment 1 of the submittal was the required thirty-day response for units that had a scheduled Fall 2003 outage. In summary, Dominion's response for Surry Unit 2 included a commitment to perform a 360-degree, bare-metal visual examination of the fifty lower reactor pressure vessel (RPV) head bottom-mounted instrumentation (BMI) penetration nozzles during the Surry Unit 2 Fall 2003 refueling outage. This inspection was performed during the refueling outage with no indication of boric acid leakage detected at any of the BMI nozzles nor was any indication of head wastage observed. These results, and the inspection techniques used, are documented in Dominion's letter to the NRC dated January 30, 2004 (Serial No. 03-459B). If evidence of boric acid deposits had been identified on any of the BMI penetration nozzles, the finding would have been entered into the corrective action program for tracking, cause determination and disposition/resolution of the condition.

   Bare-metal VT-2 visual examinations of the BMI penetration nozzles will be performed during each refueling outage for Surry Unit 2. Relief Request SPT-003 has been revised to include this requirement and supercedes the previously provided relief request. Revision 1 of Relief Request SPT-003 is included in Enclosure 2 for NRC review and approval. The bare-metal visual examination of the BMI penetration nozzles discussed above will allow the examiner to perform a much more thorough and effective examination versus the specified code examination, which would require the visual inspection to be performed while wearing a full face, self-contained breathing apparatus due to sub-atmospheric conditions.

Enclosure 1

Drawings

11548-FM-1E
11548-FM-1F
11548-FM-1G

Surry Power Station Unit 2

Virginia Electric and Power Company
(Dominion)

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D-03

Enclosure 2

Relief Request SPT–003, Revision 1

Surry Power Station Unit 2

Virginia Electric and Power Company
(Dominion)

RELIEF REQUEST SPT-003, REVISION 1

I. IDENTIFICATION OF COMPONENTS

System: Reactor Coolant (RC)

Components: Partial Penetration Welds at the Bottom of the Reactor Vessel

II. CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1998 Edition with Addenda up to and including the 2000 Addenda, Category B-P, Item No. B15.10, requires a visual (VT-2) examination of the bottom of the reactor vessel during the system leakage test of IWB-5220.

III. BASIS FOR RELIEF

To meet the Section XI pressure and temperature requirements for the system leakage test of the reactor vessel, the SPS 2 reactor containment is required to be at subatmospheric pressure. Station administrative 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 these 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 alternate examination would take place. The proposed alternate examination would be performed under conditions that are safer and allow for a more thorough examination.

RELIEF REQUEST SPT-003, REVISION 1

IV. ALTERNATE PROVISIONS

Technical Specifications have surveillance requirements that monitor leakage and radiation levels. The applicable Technical Specification requirements will be satisfied through the fourth inservice inspection interval. Furthermore, 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. A bare-metal VT-2 visual examination for evidence of boric acid leakage/corrosion will be conducted each refueling outage on the bottom of the reactor vessel when the containment is at atmospheric conditions.

The monitoring methods of the station and the VT-2 visual examination of the area each refueling outage provide 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, results in a hardship without a compensating increase in quality and safety over the proposed alternative. Therefore, approval of this request for relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii).

(Note: A similar relief request was approved for North Anna Unit 1 for the third inservice inspection interval under TAC No. MA5750. Requests for relief were also approved for North Anna Unit 2, third inservice inspection interval under TAC No. MB2280; and for Surry Units 1 and 2, third inservice inspection intervals under TAC Nos. MB1083 and MB1084, respectively.)