VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

December 12, 2002

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

Serial No. 02-642
NL&OS/GDM R1
Docket Nos. 50-280
License Nos. DPR-32

Gentlemen:

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 1
FOURTH INTERVAL INSERVICE INSPECTION PROGRAM

Pursuant to 10 CFR 50.55a(g), Virginia Electric and Power Company (Dominion) submits the inservice inspection (ISI) program for the fourth inservice inspection interval for Surry Unit 1 for Class 1, 2, and 3 components and component supports. The Inservice Inspection Plan, included as Attachment 1 to this letter, describes the programmatic aspects of inservice examinations of components and component supports and system pressure tests. In letters dated October 31, 1997 and August 13, 1998 (Serial Nos. 97-640 and 98-421, respectively), Dominion submitted a risk-informed ISI program for Class 1, 2, and 3 piping for Surry Unit 1. The fourth interval ISI program examination requirements for Class 1 and 2 piping (Examination Categories B-F, B-J, C-F-1, and C-F-2) comply with the NRC approved risk-informed ISI program.

The Surry Unit 1 ISI Schedule Summary identifies the frequency and types of examinations to be performed during the fourth inservice inspection interval and is included as Attachment 2 for your information. The summary schedule presents the anticipated examinations for the interval based on ISI category and item numbers. This schedule should be considered tentative since over the duration of the interval some of the detailed information provided in the schedule summary will likely change. Such changes are required in response to specific issues such as plant configuration or risk-informed ISI updates.

The System Pressure Test Plan is not included in this submittal. It will be separately submitted in subsequent correspondence.

This program has been written in accordance with the requirements of the 1998 Code Edition, with addenda up to and including the 2000 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code (98-2000 Code). The fourth inservice inspection interval will begin on October 14, 2003, and Surry Unit 1 will begin implementation of the fourth interval inservice inspection program on that date. Please note that the incorporation of the 98-2000 Code into 10 CFR 50.55a occurred on October 28, 2002.

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As allowed by 10 CFR 50.55a(g)(4)(iv), Surry Unit 1 will implement the inservice examination of components and system pressure test requirements set forth in the 98-2000 Code. Surry Unit 1 will also comply with the limitations and modifications listed in 10 CFR 50.55a(b) related to the implementation of the 98-2000 Code. The referenced paragraph subjects this action to the approval of the Nuclear Regulatory Commission and this letter hereby requests that approval. Dominion has decided to exercise this option to permit the eventual update of both Surry units to the same edition of the ASME Section XI Code. (The fourth interval inservice inspection program update for Surry Unit 2 will be determined by the edition of the Section XI Code referenced in 10 CFR 50.55a(b) on May 10, 2003.)

The implementation of the 98-2000 Code will be limited for Surry Unit 1 because it was not designed to completely meet the detailed inservice inspection examination and system pressure test requirements of Section XI due to its vintage. Therefore, Dominion is proposing alternatives and/or requesting relief from certain inspection and testing requirements. The proposed alternatives and relief requested from specific Section XI requirements are provided in Attachment 1. Previously submitted similar relief requests are identified along with the NRC approval reference, where applicable.

The Surry Unit 1 fourth interval inservice inspection program and associated relief requests have been reviewed and approved by the Station Nuclear Safety and Operating Committee.

If you have any questions or require additional information, please contact us.

Very truly yours,

Leslie N. Hartz
Vice President – Nuclear Engineering

Attachments

Commitments made in this letter: None

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Attachment 1

<u>Fourth Interval Inservice Inspection Program</u>
<u>Surry Power Station Unit 1</u>

Virginia Electric and Power Co.
(Dominion)

VIRGINIA ELECTRIC AND POWER COMPANY
(DOMINION)
SURRY POWER STATION, UNIT 1

INSERVICE INSPECTION PLAN FOR
COMPONENTS AND COMPONENT SUPPORTS

THE FOURTH INSERVICE INSPECTION INTERVAL
OCTOBER 14, 2003 – DECEMBER 13, 2013

REVISION 0
OCTOBER 2002

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As allowed by Title 10, Code of Federal Regulations, Section 50.55a (10 CFR 50.55a), paragraph (g)(4)(iv), as revised October 28, 2002, the Surry Power Station, Unit 1 (SPS 1), "Inservice Inspection Plan for Components and Component Supports - Fourth Inservice Inspection Interval" (ISI Plan) has been prepared to meet the requirements of American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section XI, Rules For Inservice Inspection of Nuclear Power Plant Components (Section XI), 1998 Edition up to and including the 2000 Addenda (hereafter this document may be identified as either the "98-2000 Code" or as the "Code."). This updated program also complies with the supplemental requirements of 10 CFR 50.55a effective on October 28, 2002 regarding the use of the 98-2000 Code. The program commences on October 14, 2003 and is scheduled to extend through December 13, 2013. Extension of the interval end date by 2 months is to resolve anticipated refueling outage schedule conflicts with the end dates of the second and third periods. In cases where the requirements of the Code have been determined to be impractical, requests for relief were prepared and submitted to the Nuclear Regulatory Commission (NRC) as allowed by 10 CFR 50.55a(g)(5)(iii).

If the ISI Plan proposes to utilize an alternative to specific requirements of the Code, then a request for relief to use the proposed alternative has been prepared and submitted to the NRC as allowed by 10 CFR 50.55a(a)(3)(i) or (ii). The most significant alternative to the ISI Plan is the use of risk-informed selection and examination criteria as alternatives to the selection and examination requirements of Table IWB-2500-1, B-F, "Pressure Retaining Dissimilar Metal Welds In Vessel Nozzles"; Table IWB-2500-1, B-J, " Pressure Retaining Welds in Piping"; Table IWC-2500-1, C-F-1, "Pressure Retaining Welds In Austenitic Stainless Steel Or High Alloy Piping"; and Table IWC-2500-1, C-F-2, "Pressure Retaining Welds In Carbon Or Low Alloy Steel Piping". Dominion Letter 97-640 submitted this alternative to the NRC on October 31, 1997. Dominion Letter 98-421, dated August 13, 1998, submitted revisions to the initial alternative proposal. The NRC approved the alternative, as revised, on December 16, 1998 (TAC No. MA0125), Dominion Letter Number 98-736.

This Inservice Inspection (ISI) Program is divided into the "Inservice Inspection Plan for Components and Component Supports" (ISI Plan), the "Inservice Inspection Program Plan for Pumps and Valves," and the "Containment Inservice Inspection Program." The Inservice Inspection Program Plan for Pumps and Valves is submitted separately from the ISI Plan. The Containment Inservice Inspection Program contains the requirements applicable to Section XI, Subsections IWE, "Requirements For Class MC Components Of Light Water Cooled Power Plants," and IWL, "Requirements For Class CC Concrete Components of Light-Water Cooled Power Plants," and related supplemental requirements of the September 9, 1996 revised rule. The SPS 1 Containment Inservice Inspection Program was implemented in accordance with the revised rule. It has separate interval dates from the ISI Plan and does not require update at this time.

The ISI Plan, contained within, provides an overview and summary of the SPS 1 inservice inspection program for Subsections IWA, IWB, IWC, IWD, and IWF of the 98-2000 Code. Therefore, the ISI Plan is applicable to the components (including their supports) which are classified as ASME Code Class 1, Class 2, and Class 3. The boundaries of the ISI Plan, component classifications, and the employment of specific programmatic exemptions identified in IWB-1220, IWC-1220, IWD-1220, IWF-1230, and elsewhere in the Code are shown on the ISI Classification Boundary Drawings (CBB/CBM's). This is accomplished by the use of symbols as well as text on the drawings. The symbols used on the CBB/CBM's are defined on the Legends and Symbols Drawing, 11448-CB-L&S-4.

The "Inservice Inspection Schedule for Components and Component Supports" (ISI Schedule) is provided as a separate document from the ISI Plan. The ISI Schedule details the examination category and item number for each component, the examination and test requirements, and the examination methods. It also provides the schedule for the examinations. Another document separate from the ISI Plan is the System Pressure Test Plan (SPT Plan). It details the implementation of the system pressure test program. The ISI Plan, the ISI Schedule and the SPT Plan, jointly, meet the documentation requirements of the Code, IWA-2420, for the Surry Unit 1 Fourth Inservice Inspection Interval.

SURRY POWER STATION UNIT 1, INTERVAL 4
RELIEF REQUEST/CORRESPONDENCE SUMMARY

Component and NDE Relief Requests
CMP-001 - Requests the elimination of the volumetric examination of the nozzle inside radius area associated with the surge line nozzle of the pressurizer. The examination is replaced with a VT-2 visual examination.
CMP-002 - Requests permission to perform a visual examination of the inside surface of the inaccessible welds of the Outside Recirculation Spray Pump Casings and Low Head SI Pump Casings, if the pumps are opened for maintenance. When the pumps are in their service configuration, the casing welds are inaccessible from both the inside and the outside.
CMP-003 - Requests permission to continue to use existing calibration blocks for components not being examined to the requirements of Appendix VIII.
CMP-004 - Proposes to continue to use the weld reference system established in the third inservice inspection interval as opposed to the weld reference system required by Section XI.
CMP-005 - Proposes to continue to use the electronic weld reference system associated with a reactor vessel inspection tool as opposed to establishing the weld reference system required by Section XI.
System Pressure Test Relief Requests
SPT-001 - Requests permission to perform the Class 1 System Leakage Test from existing structures, ladders and platforms as opposed to using scaffolding to gain access to within 6 feet of the test surface.
SPT-002 - Proposes to conduct the hydrostatic test of Class 3 components to the requirements of Code Case N-498-1, "Alternative Rules for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems," without adopting the Code Case. Code Case N-498-1 allows the hydrostatic pressure test to be performed at normal operating pressure and temperature. The 98-2000 Edition of Section XI incorporates the reduced pressure testing requirements for Class 1 and 2 components, thus making the use of the Code Case unnecessary for those components.
SPT-003 - Requests permission to not test the approximately 20 small diameter ($\leq$ 1" NPD) vent, drain and sample lines as part of the reactor coolant system pressure test. As an alternative, an examination will be performed each refueling outage of the small piping lines looking for evidence of leakage.
SPT-004 - Requests permission to use alternative requirements to satisfy the VT-2 visual examination requirements related to the penetrations in the bottom of the reactor vessel. Specifically, an examination will be performed when the containment is at atmospheric conditions.
Component Support Relief Request
CS-001 - Requests permission to continue to use Technical Specifications for the examination and testing of snubbers as opposed to OM-4 or ISTD. ISTD is very similar to Technical Specifications for examination and testing but would require a Technical Specification change to use.
Miscellaneous Documents

Partial Coverage Relief Requests
Code Cases - Relief Requests
CC-001 - Proposes to use the reporting requirements of Code Case N-532, "Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000," as an alternative to the NIS-1 and NIS-2 reports required by Section XI.
CC-002 - Requests the use of Code Case N-533-1, "Alternative Requirements for VT-2 Visual Examinations of Class 1, 2, and 3, Insulated Pressure-Retaining Bolted Connections." This Code Case allows the bolted connections in systems borated for the purpose of controlling reactivity to be examined during the normal refueling outage in lieu of when the connections are at operational pressure.
CC-003 - Requests permission to use Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections." This Code Case allows an evaluation to be performed to determine if a bolt must be removed for examination.
CC-004 - Requests permission to use Code Case N-597, "Requirements for Analytical Evaluation of Pipe Thinning." This will allow a pipe with flow assisted corrosion (FAC) degradation found to be unacceptable under the current Code minimum thickness requirements to remain in service for an additional period of time if the requirements of the Code Case are met.
CC-005 - Requests the deferral of 100% of the reactor vessel shell-to-flange weld to the third period by use of Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel"
Risk Informed - ISI - Relief Requests
R-001 - Requests permission to not perform the required volumetric examination of high safety-significant (HSS) socket welds and branch connections A VT-2 visual examination will be performed as an alternative.

SECTION 1 INSERVICE INSPECTION PLAN FOR COMPONENTS AND COMPONENT SUPPORTS

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1.0 Inservice Inspection Plan - General

1.1 General Information

Surry Power Station Unit 1 (SPS 1) is located on the James River in Surry County, Virginia. The plant employs a three loop Pressurized Water Reactor (PWR) Nuclear Steam Supply System provided by Westinghouse Electric Corporation.

1.2 Fourth Inservice Inspection Interval

As allowed by Title 10, Code of Federal Regulations, Part 50, Section 50.55a (10 CFR 50.55a), paragraph (g)(4)(iv), the SPS 1 fourth inservice inspection interval ISI Plan was updated to the 1998-2000 Section XI Code. This is the latest edition and addenda of Section XI incorporated into 10 CFR 50.55a. It was incorporated into the regulation on October 28, 2002. The SPS 1 update to the 98-2000 Code also complies with the supplemental requirements contained within the October 28, 2002 update of the 10 CFR 50.55a regarding the implementation of the 98-2000 Section XI Code. The fourth inservice inspection interval begins on October 14, 2003 and will extend through December 13, 2013. The fourth inservice inspection interval is scheduled for a period of 10 years and two months. The dates for the inspection periods are scheduled as follows:

Period 1: October 14, 2003 through October 13, 2006
Period 2: October 14, 2006 through December 13, 2010
Period 3: December 14, 2010 through December 13, 2013

(Note: The successive inservice inspection interval is planned to begin on October 14, 2013. See IWA-2430(d).)

The inservice inspection interval is extended two months beyond the standard 10 years in order to avoid a potential scheduling conflict between the last examination outages of the second and third periods and their respective period and interval end dates.

1.3 Inservice Inspection Plan Description

1.3.1 Scope of the ISI Plan

The ISI Plan contained herein addresses the examination and testing of Class 1, 2, and 3 components and the associated component supports. Applicable requirements of Subsections IWA, IWB, IWC, IWD, IWF and the associated Mandatory

Appendices of the Code are contained in the ISI Plan, ISI Schedule, and the SPT Plan. This document is not intended to provide specific information on the implementation of the ISI program. The intent of this document is to provide information on the scope of the SPS 1 ISI Plan (e.g., its boundary and compliance with Section XI), provide alternatives to the requirements of Section XI where appropriate, and identify those Section XI requirements determined to be impractical. Requests for relief for the impractical requirements have been developed per 10 CFR 50.55a(g)(5)(iii). Alternatives have been proposed in accordance with 10 CFR 50.55a(a)(3)(i) or (ii).

The ISI Schedule and the SPT Plan provide information on the specific components selected for examination and test, including the category and item number, and the examination or test to be performed. Unless stated otherwise, the extent of examination will be 100% of the Code required examination or test surface, volume or area.

1.3.2 Inspection Plan and Implementation Schedule

The inspection plan required by IWA-2420 (a) and the implementing schedule required by IWA-2420 (b) detail the requirements to be addressed by the overall ISI program for components and component supports. The ISI Plan, ISI Schedule, and the SPT Plan together, provide the required information.

1.3.3 Inspection Program Employed

The ISI Plan for SPS 1 utilizes the interval format of Inspection Program B, as defined in IWA-2432.

1.3.4 Classification and Identification of Components

The construction permit for SPS 1 was issued on June 25, 1968. At that time, the ASME Boiler and Pressure Vessel Code covered only pressure vessels. Piping, pumps and valves were built primarily to the rules of USAS B31.1. Essentially, Surry Power Station was designed and constructed prior to the origination of the ASME Code classifications named Class 1, 2, and 3. Therefore, the system classifications used as a basis for the Inservice Inspection and Testing Programs are based on the guidance set forth in 10 CFR 50 and Regulatory Guide 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear

Power Plants." Pursuant to 10 CFR 50.55a paragraph (g)(1), inservice inspection requirements of Section XI of the ASME Code are then assigned to these components, within the constraints of existing plant design.

Classification Boundary Drawings (CBB/CBM's) documenting the system classifications were developed to aid in the review and implementation of the subject programs. A list of the CBB/CBM drawings follows in paragraph 1.7.

1.3.5 Components Exempt from Examination

The application of the exemptions allowed by IWB-1220, IWC-1220, IWD-1220, and IWF-1230, and other paragraphs of the Code is also detailed on the 11448-CBB/CBM drawings. This is accomplished by the use of symbols as well as text on the drawings. The legend for the symbols is provided on drawing 11448-CBM-L&S-4. Additionally, the ISI Schedule and the SPT Plan provide the Code Classification in accordance with the information provided on the CBB/CBM drawings.

1.3.6 Requests For Relief

Where the requirements of ASME Section XI have been determined to be impractical, requests for relief have been developed in accordance with 10 CFR 50.55a(g)(5)(iii). Additionally, where alternatives to the requirements of Section XI are proposed, they will be developed in accordance with the requirements of 10 CFR 50.55a(a)(3)(i) and/or (ii). These requests for approval to use alternative requirements will also be presented in the sets of requests for relief . There are six sets of requests for relief:

- a CMP series for components and nondestructive examination requirements,
- a SPT series for system pressure testing requirements,
- a CS series for component support requirements,
- a PRT series for partial examinations,
- a CC series for Code Cases that are part of the ISI Plan but not approved by the NRC for industry use, and
- a R series for risk-informed ISI requirements.

The partial examination requests for relief are necessary because SPS 1 was not designed/constructed considering access for examination. Consequently, it may not be possible to obtain 100% of each required examination or test. Therefore, requests for relief asking approval for partially completed examinations or tests will be submitted. These requests for relief will be developed and submitted as each partial examination/test is completed. (Note: Code Case N-460 will be utilized to the extent allowed to accept partial examinations of Class 1 and Class 2 welds.)

1.3.7 ASME Section XI Code Cases

As allowed by 10 CFR 50.55a and/or USNRC Regulatory Guide 1.147, Revision 12, certain Code Cases have been incorporated into this program. Additionally, certain Code Cases that have been approved by the Code, but have not yet been approved by the NRC, are also incorporated into the ISI Plan. Approval to use these unapproved Code Cases has been requested as part of the request for relief process. All Code Cases utilized as part of this ISI plan are listed in Section 6.

1.3.8 Exclusion of Subsections IWE and IWL

Subsections IWE, "Requirements For Class MC Components of Light-Water Cooled Power Plants," and IWL, "Requirements For Class CC Concrete Components of Light-Water Cooled Power Plants," have not been included in this ISI Plan. The programs to satisfy these requirements have been developed and are being implemented as required by the regulations. The implementation of SPS 1 IWE and IWL programs is being controlled by other ISI programs exclusive of this plan. No submittal has been made of the IWE and IWL programs as allowed by the regulations.

1.3.9 Pump and Valve Inservice Testing Program

Subsections IWP and IWV, Inservice Testing of Pumps and Inservice Testing of Valves in Nuclear Power Plants are no longer included in the Section XI Code and accordingly are not included in this program. This inservice testing program, addressing pumps and valves, is prepared and submitted in accordance with the requirements of 10 CFR 50.55a. However, this action is taken separately from the ISI Program.

1.4 Augmented Examinations

Augmented examinations resulting from commitments made to the NRC may or may not involve components of the ISI Plan. The implementation of augmented examinations is in accordance with the applicable commitments and is achieved by programs and procedures maintained by either the SPS 1 staff or the corporate base engineering staff. However, some augmented examinations are part of the ISI Schedule for the purposes of scheduling and tracking the completion of the involved examinations. The inclusion of an augmented examination into the ISI Schedule does not make the examination part of the ISI Plan. Any augmented examination included in the ISI Schedule is identified as an augmented examination. Since the augmented examinations performed to satisfy the regulatory commitments are outside of Section XI Code requirements, the Authorized Nuclear Inservice Inspector is not involved with the augmented examinations.

1.5 Risk-Informed ISI Plan

The most significant alternative to the ISI Plan is the use of risk-informed selection and examination criteria as alternatives to the selection and examination requirements of Table IWB-2500-1, B-F, "Pressure Retaining Dissimilar Metal Welds In Vessel Nozzles"; Table IWB-2500-1, B-J, "Pressure Retaining Welds in Piping"; Table IWC-2500-1, C-F-1, "Pressure Retaining Welds In Austenitic Stainless Steel Or High Alloy Piping"; and Table IWC-2500-1, C-F-2, "Pressure Retaining Welds In Carbon Or Low Alloy Steel Piping." Dominion Letter 97-640 to the NRC submitted this alternative, known as the Risk-Informed Inservice Inspection (RI-ISI) Program (RI-ISI Plan), on October 31, 1997. Dominion Letter 98-421, dated August 13, 1998, revised the original submittal by providing an update of the Risk-Informed Inservice Inspection Plan. The NRC approved the revised alternative program on December 16, 1998 (TAC No. MA0125), Dominion Letter Number 98-736. These documents and others related to the development and approval of the RI-ISI alternative by the NRC are retained within the records retention program for SPS 1. In addition, copies of the documents have been gathered into a basis document for easy reference. The title of this document is, *"Risk Informed Inservice Inspection Program Basis, Surry Power Station Unit 1"*.

Components selected for examination by the RI-ISI Plan are identified, scheduled for examination, and performance tracked as part of the ISI Schedule. The RI-ISI Plan does not provide alternatives to the pressure test program.

1.6 Repair/Replacement Program

Repair/Replacement activity will be in accordance with Surry Power Station Unit 1 administrative procedures as well as Dominion corporate administrative procedures. Together, these sets of procedures assure compliance with Section XI. Pressure testing required for Section XI repair/replacement activity will be conducted in accordance with the repair/replacement program.

1.7 Inservice Inspection Drawings

The following is a list of the Classification Boundary Drawings. The boundaries of the ISI Plan, component classifications, and the employment of specific programmatic exemptions identified in IWB-1220, IWC-1220, IWD-1220, IWF-1230, and elsewhere in the Code are shown on the ISI Classification Boundary Drawings (CBB/CBM series). This is accomplished by the use of symbols as well as text on the drawings. The symbols used on these drawings are defined on the Legends and Symbols Drawing, 11448-CBM-L&S-4.

Two additional sets of drawings are used to implement the ISI Plan - the 11448-WMKS series and the 11448-SPB/SPM series. The 11448-WMKS series identifies the components in the program and their location in the plant. The locations of welds and supports are shown on these drawings. The ISI Schedule identifies which WMKS is associated with each component in the program. The 11448-SPB/SPM series identifies the extent of the test zones used by the SPT Plan to complete the required surveillance system pressure tests.

SURRY POWER STATION UNIT 1, INTERVAL 4
ISI CLASSIFICATION BOUNDARY DRAWINGS

11448-CB-L&S-4, SH-001,	LEGEND & SYMBOLS
11448-CBB-006A-4, SH-001,	AIR COOLING & PURGING SYSTEM
11448-CBB-041A-4, SH-001,	CHILLED WATER SYSTEM
11448-CBB-041A-4, SH-002,	CHILLED WATER SYSTEM
11448-CBB-041A-4, SH-003,	CHILLED WATER SYSTEM
11448-CBB-047B-4, SH-001,	FIRE PROTECTION SYSTEM
11448-CBM-064A-4, SH-001,	MAIN STEAM SYSTEM
11448-CBM-064A-4, SH-002,	MAIN STEAM SYSTEM
11448-CBM-064A-4, SH-003,	MAIN STEAM SYSTEM
11448-CBM-064A-4, SH-004,	MAIN STEAM SYSTEM
11448-CBM-064B-4, SH-001,	STEAM GENERATOR NITROGEN CONN SYSTEM
11448-CBM-066A-4, SH-002,	AUX STEAM & AIR REMOVAL SYSTEM
11448-CBM-066B-4, SH-001,	AUX STEAM SYSTEM - PRIMARY PLANT
11448-CBM-067A-4, SH-001,	CONDENSATE SYSTEM
11448-CBM-067A-4, SH-002,	CONDENSATE SYSTEM
11448-CBM-067B-4, SH-001,	CONDENSATE SYSTEM
11448-CBM-067B-4, SH-002,	CONDENSATE SYSTEM
11448-CBM-068A-4, SH-001,	FEEDWATER SYSTEM
11448-CBM-068A-4, SH-002,	FEEDWATER SYSTEM
11448-CBM-068A-4, SH-003,	FEEDWATER SYSTEM
11448-CBM-068A-4, SH-004,	FEEDWATER EMERGENCY MAKE-UP SYSTEM
11448-CBM-069A-4, SH-002,	MOISTURE SEPARATOR & HP HTR DR & V
11448-CBM-070A-4, SH-001,	LP HEATER DRAINS & VENTS
11448-CBM-070A-4, SH-003,	LP HEATER DRAINS & VENTS
11448-CBM-071A-4, SH-001,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071A-4, SH-002,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071A-4, SH-003,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071A-4, SH-004,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071B-4, SH-001,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071B-4, SH-002,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071D-4, SH-001,	CIRCULATING AND SERVICE WATER SYSTEM
11448-CBM-071D-4, SH-002,	CIRCULATING & SERVICE WATER SYSTEM
11448-CBM-072A-4, SH-001,	COMPONENT COOLING WATER SYSTEM

11448-CBM-072A-4, SH-002,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072A-4, SH-003,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072A-4, SH-004,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072A-4, SH-005,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072A-4, SH-006,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072A-4, SH-007,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072B-4, SH-001,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072B-4, SH-002,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072B-4, SH-003,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072C-4, SH-001,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072C-4, SH-002,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072C-4, SH-003,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072C-4, SH-004,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072D-4, SH-001,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072D-4, SH-002,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072D-4, SH-003,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072E-4, SH-001,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072E-4, SH-002,	COMPONENT COOLING WATER SYSTEM
11448-CBM-072G-4, SH-001,	COMPONENT COOLING WATER SYSTEM
11448-CBM-074A-4, SH-001,	VACUUM PRIMING SYSTEM
11448-CBM-075C-4, SH-001,	COMPRESSED AIR SYSTEM
11448-CBM-075G-4, SH-001,	COMPRESSED AIR SYSTEM
11448-CBM-075J-4, SH-001,	CONTAINMENT INSTRUMENT AIR SYSTEM
11448-CBM-079A-4, SH-002,	BORON RECOVERY SYSTEM
11448-CBM-079C-4, SH-001,	BORON RECOVERY SYSTEM
11448-CBM-079D-4, SH-001,	BORON RECOVERY SYSTEM
11448-CBM-081A-4, SH-001,	FUEL PIT SYSTEMS
11448-CBM-082A-4, SH-001,	SAMPLING SYSTEM
11448-CBM-082B-4, SH-002,	SAMPLING SYSTEM
11448-CBM-083A-4, SH-001,	VENT AND DRAIN SYSTEM
11448-CBM-083A-4, SH-002,	VENT AND DRAIN SYSTEM
11448-CBM-083B-4, SH-001,	VENT AND DRAIN SYSTEM
11448-CBM-083B-4, SH-003,	VENT AND DRAIN SYSTEM
11448-CBM-084A-4, SH-001,	CONTAINMNET SPRAY SYSTEM
11448-CBM-084A-4, SH-002,	CONTAINMENT SPRAY SYSTEM
11448-CBM-084A-4, SH-003,	CONTAINMENT SPRAY SYSTEM
11448-CBM-084B-4, SH-001,	RECIRCULATION SPRAY SYSTEM
11448-CBM-084B-4, SH-002,	RECIRCULATION SPRAY SYSTEM
11448-CBM-085A-4, SH-001,	CONT VACUUM & LEAKAGE MONITOR SYSTEM
11448-CBM-085A-4, SH-002,	CONT VACUUM & LEAKAGE MONITOR SYSTEM
11448-CBM-086A-4, SH-001,	REACTOR COOLANT SYSTEM, LOOP-A
11448-CBM-086A-4, SH-002,	REACTOR COOLANT SYSTEM, LOOP-B
11448-CBM-086A-4, SH-003,	REACTOR COOLANT SYSTEM, LOOP-C

11448-CBM-086B-4, SH-001,	REACTOR COOLANT SYSTEM
11448-CBM-086B-4, SH-002,	REACTOR COOLANT SYSTEM
11448-CBM-086C-4, SH-001,	REACTOR COOLANT SYSTEM
11448-CBM-086C-4, SH-002,	REACTOR COOLANT SYSTEM
11448-CBM-087A-4, SH-001,	RESIDUAL HEAT REMOVAL SYSTEM
11448-CBM-087A-4, SH-002,	RESIDUAL HEAT REMOVAL SYSTEM
11448-CBM-088A-4, SH-001,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088A-4, SH-002,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088A-4, SH-003,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088A-4, SH-004,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088B-4, SH-001,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088B-4, SH-002,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088C-4, SH-001,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-088C-4, SH-002,	CHEMICAL & VOLUME CONTROL SYSTEM
11448-CBM-089A-4, SH-001,	SAFETY INJECTION SYSTEM
11448-CBM-089A-4, SH-002,	SAFETY INJECTION SYSTEM
11448-CBM-089A-4, SH-003,	SAFETY INJECTION SYSTEM
11448-CBM-089B-4, SH-001,	SAFETY INJECTION SYSTEM
11448-CBM-089B-4, SH-002,	SAFETY INJECTION SYSTEM
11448-CBM-089B-4, SH-003,	SAFETY INJECTION SYSTEM
11448-CBM-089B-4, SH-004,	SAFETY INJECTION SYSTEM
11448-CBM-090C-4, SH-001,	CONTAINMENT HYDROGEN ANALYZER SYSTEM
11448-CBM-118A-4, SH-002,	REACTOR CAVITY PURIFICATION SYSTEM
11448-CBM-124A-4, SH-001,	STEAM GEN BLOWDOWN RECIRC & XFER SYSTEM
11448-CBM-124A-4, SH-002,	STEAM GEN BLOWDOWN RECIRC & XFER SYSTEM
11448-CBM-124A-4, SH-003,	STEAM GEN BLOWDOWN RECIRC & XFER SYSTEM
11448-CBM-124A-4, SH-004,	STEAM GEN BLOWDOWN RECIRC & XFER SYSTEM
11448-CBM-124B-4, SH-001,	STEAM GENERATOR BLOWDOWN SYSTEM
11448-CBM-130A-4, SH-001,	RAD MONITOR SYSTEM CIRC & SERV WTR
11448-CBM-130B-4, SH-001,	RAD MONITOR CONT PARTICULATE SYSTEM

SECTION 2 REQUESTS FOR RELIEF - COMPONENTS

Section 2.1 Nondestructive Examination

RELIEF REQUEST CMP-001

I. IDENTIFICATION OF COMPONENTS

Nozzle Inner Radius Section (Pressurizer Surge Nozzle)

Weld #	Component #	Drawing #	Class
23NIR	1-RC-E-2	11448-WMKS-RC-E-2	1

II. CODE REQUIREMENTS

Category B-D, Item B3.120 requires the volumetric examination of the nozzle inside radius section of the pressurizer surge nozzle. (Note: In accordance with 10 CFR 50.55a revision effective October 28, 2002, the Code reference is to the 1998 Edition.)

III. BASIS OF REQUEST FOR RELIEF

The Surry Unit 1 pressurizer surge line nozzle is integrally cast into the bottom pressurizer lower head. The nozzle is located under the pressurizer skirt and is surrounded by 78 heater penetrations. Interference from the heater penetrations and heater cables, as well as the location of the nozzle under the pressurizer skirt restricts the access to the nozzle. This limits the examiner's ability to manipulate the search unit to examine the nozzle inner radius.

The only viable ultrasonic technique currently available to examine nozzle inner radii involves the fabrication of calibration blocks that closely simulate the O.D. and I.D. nozzle geometry. This is necessary so that search units can be produced that will interrogate the inner radius section at precise angles. Also, in order to obtain meaningful results, the nozzle material grain structure must be such that an adequate signal-to-noise ratio can be obtained over a long metal path distance.

Integrally cast nozzles contain limitations such as an irregular O.D. profile, a rough surface condition, and an attenuating grain structure. The irregular surface condition causes the beam angle to vary from point to point around the nozzle. The attenuating grain structure results in a low signal-to-noise ratio at the nozzle inner radius. Limited access to the nozzle as well as the limitations imposed by the material conditions, area dose rates, and the complicated nature of the examination technique would make evaluation of any indications very difficult.

Access to the SPS 1 pressurizer surge line nozzle is obstructed by insulation and the cables for the pressurizer heaters. Removal of the insulation and cables would be difficult as well as labor and time intensive. The exposure of the workers and technicians needed to perform the tasks is a real and relevant concern. It is almost certain that some, and possibly all, heater cables would have to be disconnected so that the cables can be pulled back to allow access for removing insulation and doing the exam. It is also likely that some cable or heater damage would occur during removal. If it is assumed that all 78 heater cables have to be disconnected and pulled back, the dose estimate is 55.773 man-rem. While actions would be taken to provide protection against the radiation, the large dose rate gradients in the under-pressurizer area present an unusual challenge. Temporary shielding is considered impractical in this situation because placement of the shielding material would obstruct and potentially preclude accessibility to the examination surface.

Other personnel safety concerns potentially involved with this examination include the increased risk for an unplanned exposure event and prevention of contamination with personnel wedged between the surge line and the exposed portion of the pressurizer heaters. Other issues include actual accessibility after removal of the various forms of interference and the likelihood of difficulties in replacing the insulation to its original configuration. Furthermore, the amount of examination coverage would be dependent on the overall accessibility obtained.

In conjunction with license renewal, Westinghouse performed an evaluation to address the impact of operational transients for SPS 1 to account for insurge/outsurge transients in addition to design transients in the pressurizer lower head. The results of the evaluation show that the Cumulative Usage Factor (CUF) for the nozzle inner radius are 0.29 (inside surface) and 0.11 (outside surface). These CUFs are considerably less than the design limit of 1.0 and provide insight into the potential for failure in this area. Additionally, Dominion is unaware of an industry failure involving the inside radius section of the surge-line nozzle in a Westinghouse design pressurizer.

There are several uncertainties regarding an alternative examination of the inside surface of the pressurizer surge line area by use of a remote visual tool. Such an examination requires that a boroscope be fed through the manway and down through openings in the heater support baffles. Adding to the difficulty in performing such an exam, there is a perforated basket diffuser covering the surge nozzle opening on the inside of the pressurizer. The boroscope would need to be positioned through the support plates, and then threaded through a perforation in the basket diffuser, if possible, to the pressurizer surge line area. (See Figure.) This examination will be partially obscured by the thermal sleeve, which extends beyond the inside radius area into the volume of the pressurizer. These obstructions would need to be overcome several times in order to achieve the required examination coverage. Furthermore, the resulting examination would only be of the cladding that covers the inside radius of the nozzle, which is considered to be only marginally beneficial in determining the structural integrity of the nozzle. Additionally, performing the visual inspection requires opening the RCS and establishing access and foreign material exclusion controls. The boroscope itself has the potential to become lodged inside the perforated basket diffuser or behind a pressurizer heater support plate. This inspection effort and the significant potential risk associated with it are not commensurate with the limited benefit that may be obtained from the inspection.

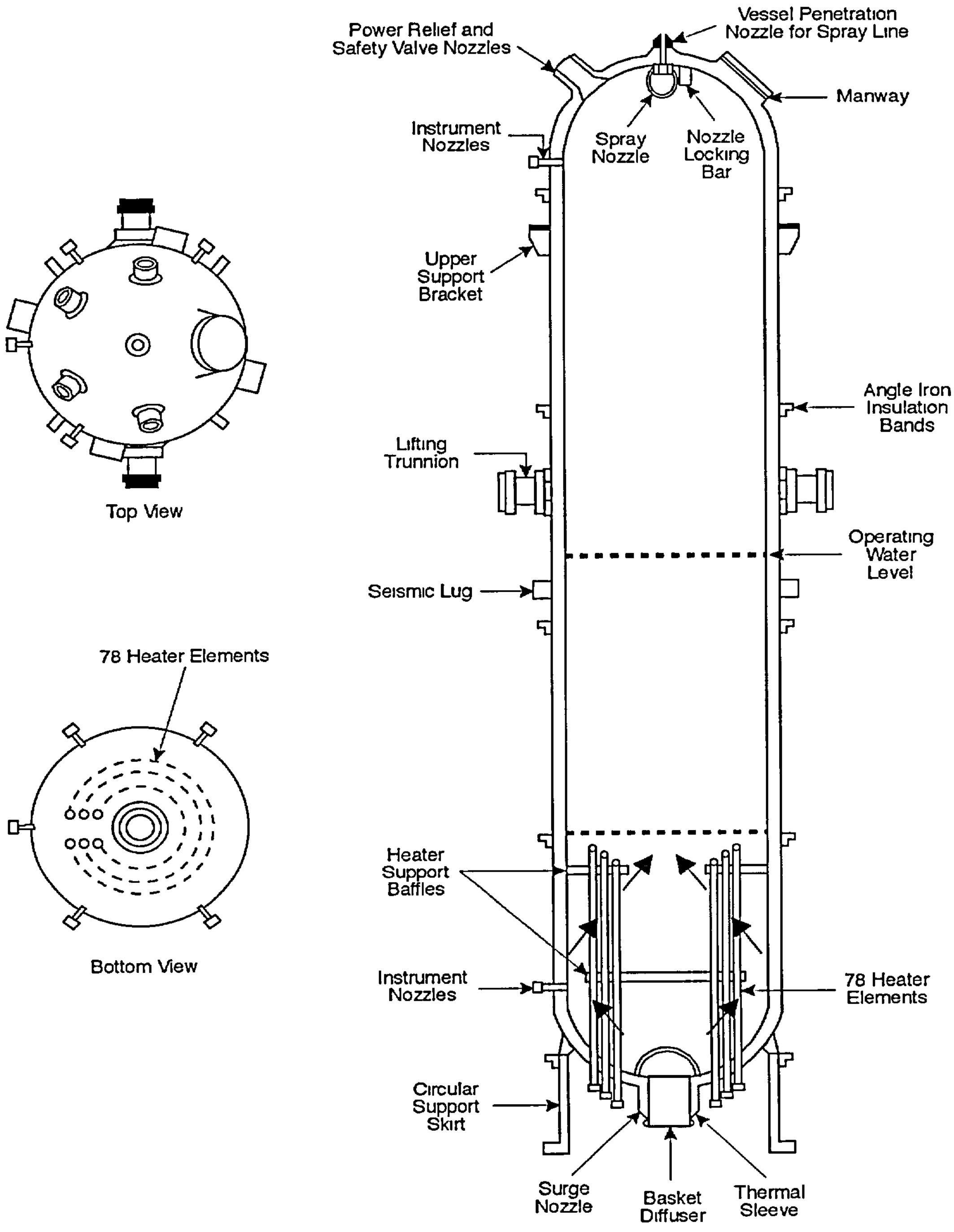
Any ultrasonic examination on this nozzle could only be described as "best effort." The benefit gain would not be commensurate with the difficulty and anticipated exposure estimate of 55.773 man-rem to perform this examination. An alternative examination employing a remote visual technology has very little if any reasonable probability of success. As such, we are applying for relief per 10 CFR 50.55a(a)(3)(ii) since compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

IV. ALTERNATE REQUIREMENTS

The pressurizer surge line nozzle-to-vessel inner radius section will be examined as part of the normally scheduled Class 1 system leakage test each refueling. In addition, the surveillance requirements of Technical Specifications that determine the reactor coolant system leak rate and the containment atmosphere radioactivity will be satisfied. These programs ensure that the overall level of plant quality and safety will not be compromised.

(Note: A similar relief for North Anna Power Station Unit 1 was granted for use during the second inservice inspection interval, TAC No. M71066; and during the third inservice inspection interval, TAC NO. MA5750. This relief request was also approved for North Anna Power Station Unit 2 for the second inservice inspection interval, TAC NO. M79147; and for the third inservice inspection interval, TAC NO. MB2280. Similar relief was also granted for Surry Power Station Unit 1, third inservice inspection interval, TAC NO. M87312; Surry Power Station Unit 2, third inservice inspection interval, TAC NO. M89085; Byron Station Units 1 and 2, second inservice inspection interval, TAC Nos. M94830 and M94831; Haddam Neck Plant, third inservice inspection interval; TAC NO. M80457; and Beaver Valley by letter dated 10/8/97.)

FIGURE CMP-001-1



PRESSURIZER

RELIEF REQUEST CMP-002

I. IDENTIFICATION OF COMPONENTS

Systems: Outside Recirculation Spray (RS) and Safety Injection (SI)

Components: Pump casing welds identified below

Component	Weld	Drawing #	Class
1-RS-P-2A	2-01	11448-WMKS-RS-P-2A	2
1-RS-P-2A	2-02	11448-WMKS-RS-P-2A	2
1-RS-P-2A	2-03	11448-WMKS-RS-P-2A	2
1-RS-P-2A	2-04	11448-WMKS-RS-P-2A	2
1-RS-P-2A	0-12*	11448-WMKS-RS-P-2A	2
1-RS-P-2B	2-01	11448-WMKS-RS-P-2B	2
1-RS-P-2B	2-02	11448-WMKS-RS-P-2B	2
1-RS-P-2B	2-03	11448-WMKS-RS-P-2B	2
1-RS-P-2B	2-04	11448-WMKS-RS-P-2B	2
1-RS-P-2B	0-12*	11448-WMKS-RS-P-2B	2
1-SI-P-1A	2-01	11448-WMKS-SI-P-1A	2
1-SI-P-1A	2-02	11448-WMKS-SI-P-1A	2
1-SI-P-1A	2-03	11448-WMKS-SI-P-1A	2
1-SI-P-1A	2-04	11448-WMKS-SI-P-1A	2
1-SI-P-1A	0-13*	11448-WMKS-SI-P-1A	2
1-SI-P-1B	2-01	11448-WMKS-SI-P-1B	2
1-SI-P-1B	2-02	11448-WMKS-SI-P-1B	2
1-SI-P-1B	2-03	11448-WMKS-SI-P-1B	2
1-SI-P-1B	2-04	11448-WMKS-SI-P-1B	2
1-SI-P-1B	0-13*	11448-WMKS-SI-P-1B	2

(* Welds 0-12 on 1-RS-P-2A, 2B and 0-13 on 1-SI-P-1A, 1B have not been verified to exist. However, large areas of the pump casing surface are inaccessible or extremely difficult to inspect due to surrounding areas of interference. The construction technique used on the pump casing in areas that are accessible for inspection and as documented in previously identified inaccessible areas would indicate these welds do exist. When these pumps are removed for maintenance in the future, an inspection of the areas from the ID of the pump will be performed to confirm the presence or absence of the welds and this relief request will be amended at that time if required.)

RELIEF REQUEST CMP-002

II. CODE REQUIREMENTS

Category C-G, Item C6.10, Pump Casing Welds, requires that a surface examination be performed on 100% of the welds each interval. The examination can be limited to one pump casing in a system if multiple pumps of similar design, size, function, and service are involved. Examination Category C-G further states that the examination may be performed from either the inside or outside surface of the component.

III. BASIS OF REQUEST FOR RELIEF

These pumps are vertical, two-stage, centrifugal pumps, with an extended shaft and casing to allow suction from the containment sump. The welds (or portion of welds) identified above are part of the associated pump casings that are embedded within the concrete building structure. This concrete embedment makes the welds inaccessible from the outside. Access to the inside of the pump casings is limited by physical size of the casing (27-1/4 inch inside diameter for most of its approximately 49 foot length), as well as the pump shaft and the pump shaft support obstructions which are within the pump casing when the pump is assembled.

The pump assembly is of significant weight (approximately 9,000 lbs.) and extends essentially the full length of the pump casing. The removal of the pump from the pump casing to gain access for examination would be a significant undertaking because disassembly of the pump would be required. It is considered to be impractical because it would allow access for surface examination to only a small portion of the overall weld area that is inaccessible when the pump assembly is in place. Except for those welds that are partially accessible, the welds of interest are located on the lower portion of the casing, i.e., the furthest area away from the opening created by the disassembly. It is also possible that such pump disassembly/assembly, specifically for examination, may negatively contribute to the overall performance of the pumps.

IV. ALTERNATE REQUIREMENTS

In accordance with 10 CFR 50.55a(a)(3)(ii) the following alternative is proposed. A surface examination of the accessible portions of the welds will be performed to the extent and frequency described in IWC-2500. A remote visual examination (VT-1) of the inside diameter of the pump casing welds will be performed only if the pump is disassembled for maintenance. The remote VT-1 examination will be to the extent allowed by the maintenance activity and will meet the requirements of IWA-2210, "Visual Examinations," and IWA-2211, "VT-1 Examination."

RELIEF REQUEST CMP-002

(Note: A similar relief request was approved for North Anna Unit 1, third inservice inspection interval under TAC NO. MA5750; for North Anna Unit 2, third inservice inspection interval under TAC NO, MB2280; for Surry Unit 2, third inservice inspection interval under TAC NO. M89085; and for Surry Unit 1, third inservice inspection interval under TAC NO. M87312.)

RELIEF REQUEST CMP-003

I. IDENTIFICATION OF COMPONENTS

Ultrasonic calibration blocks for vessels greater than 2" in thickness that are not required to be examined in accordance with Appendix VIII to ASME Section XI.

Ultrasonic calibration blocks for piping, and vessels less than or equal to 2" in thickness, that are not required to be examined in accordance with Appendix VIII to ASME Section XI.

II. CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1998 Edition with addenda up to and including the 2000 Addenda, provides requirements for fabrication of ultrasonic calibration blocks - specifically, Article I-2000 of Appendix I and related Supplements.

III. BASIS OF REQUEST FOR RELIEF

Surry Power Station was essentially constructed prior to the issue and adoption of the current requirements of ASME Section XI. Therefore, the original ultrasonic calibration blocks used for SPS 1 were fabricated before the current guidelines of ASME Section XI were developed and approved. Meeting the requirements of Article I-2000 of Appendix I for the calibration blocks as specified in the most recently approved Section XI Code identified above would require new calibration blocks to be fabricated.

The existing calibration blocks have been used historically to examine the above components at SPS 1, and they are generally in compliance with the current requirements of the ASME Code. An example of the variations in design include the blocks for piping and vessels $\leq$ 2 inches that do not meet the recommended design specified by the Code for a thickness less than 1 inch. The notches are not staggered. Also, the notches in some of the piping blocks are located one (1) "t" (or thickness) from the end of the block instead of 1 1/2" as specified. Another example includes the vessel calibration blocks used for the steam generator primary side tubesheet-to-head weld, and pressurizer welds in that they are partially clad instead of fully clad as specified. These variations in design are not significant and do not pose a threat to the quality of the resultant examinations.

Furthermore, using the existing calibration blocks for the components identified in Section I allows correlation of ultrasonic data from the examinations of the previous inservice inspection intervals. It is considered important to maintain the repeatability of the examinations as much as possible by maintaining the use of the existing calibration blocks. Additionally, it is expected that the cost of obtaining fully compliant calibration blocks will result in expenditures not commensurate with the little or no improvement in safety that could be obtained from their use.

IV. ALTERNATIVE PROVISIONS

In accordance with 10 CFR 50.55a(a)(3)(ii) the following alternative is proposed. The existing calibration blocks will be used to perform examinations during the fourth inservice inspection interval in lieu of the current code requirements for calibration blocks. This alternative will be applicable to examinations not subject to the requirements of Appendix VIII to the Section XI Code in accordance with the regulation.

(Note: A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program under TAC NO. MA5750; for Surry Unit 1, third inservice inspection interval, under TAC NO. M87312; and for Surry Unit 2 third inservice inspection interval, under TAC NO.M89085).

RELIEF REQUEST CMP-004

I. IDENTIFICATION OF COMPONENTS

ISI Class 1 and 2 piping, vessel, and component welds normally examined from their outside surface. Excluded from this request for relief are all welds normally examined from their inside surface by use of the automated reactor vessel examination tool. This exclusion includes the reactor vessel nozzle to piping weld, ISI Class R-A.

II. CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1998 Edition with addenda up to and including the 2000 Addenda, IWA-2600, "Weld Reference System."

III. BASIS OF REQUEST FOR RELIEF

The original construction codes used at Surry Power Station, dated from the late 1960's and did not require that a weld reference system be established. Establishment of a weld reference system cannot be practically attained within the scope and schedule of existing outages. During the third inservice inspection interval, the implementation of this Section XI Code requirement was also considered to be impractical and a request for relief was submitted (SR-006 of the third inservice inspection interval ISI Program). The alternative provisions proposed in this request for relief are consistent with those proposed and accepted by the NRC for the third inservice inspection interval (Reference: NRC letter dated 07/19/95, TAC NO. M87312). Consistent with the commitments made in the third inservice inspection interval, this alternative reference system was established within the plant on those welds examined as part of the third inservice inspection interval. Continued use of the alternative reference system is reasonable because it provides an acceptable level of quality and safety. To reject the alternative reference system already in use would require the plant to establish either the system required by the Section XI Code in IWA-2600 or some other alternative system yet to be developed. In either case, significant effort would be expended to achieve compliance with the requirements of IWA-2600 (or as modified) without any justifiable gain in quality or safety. Therefore, continued use of the alternative reference system presented below is requested under the provisions of 10 CFR 50.55a(a)(3)(ii).

IV. ALTERNATE PROVISIONS

SPS 1 will use weld isometrics drawings (the WMKS series) to provide a detailed identification of location of each weld requiring examination as part of the fourth inservice inspection interval. For any weld volumetrically examined as part of the fourth inservice inspection interval that did not require volumetric examination as part of the third inservice inspection interval, the proposed alternative reference system will establish a permanent reference point indicating a zero point and direction of examination. The volumetric examination of other welds will use the points of reference established in the third inservice inspection interval, which are consistent with the stated proposal for the fourth inservice inspection interval.

Where surface examination is specified, Section XI requires that 100% of the selected weld or area be examined. Unlike the performance of a volumetric examination, there is no need to indicate the direction of examination (or scan) to assure uniformity in reporting results. In these cases no marks are placed on the weld or area. In some cases, only a portion of a weld may be examined as part of a period examination. This usually involves a large weld that is divided into thirds, with 1/3 being done each period. In these cases, the weld is required to have both a surface and volumetric examination. Therefore, a reference point is marked on the weld to assist with the volumetric examination.

Welds accepted for continued service that contain volumetric indications accepted under the criteria of IWX-3500 or IWX-3600 shall be marked to ensure the relocation of the indication, using appropriate reference marks. All reference marks will be permanently fixed on the weld.

The location of accepted surface indications is documented on a map of the weld or surface that permits accurate identification of areas on the examination surface. The map contains sufficient indicators (e.g., reference points, orientation, and/or proximity to other welds) to positively identify the weld or area in question and the examination starting point. The starting point of the map is determined from the instructions provided for determining the location of the zero reference point associated with a volumetric examination. The examination record will provide information as to the location of the surface indication on the weld examination map.

(A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program by letter dated April 25, 2000, under TAC NO. MA5750; for North Anna Unit 2, the third inservice inspection interval, under TAC NO. MB2223; and Surry Unit 2, third inservice inspection interval, under TAC NO. M89085.)

RELIEF REQUEST CMP-005

I. IDENTIFICATION OF COMPONENTS

Pressure retaining welds in the reactor vessel (Examination Category B-A), the reactor vessel nozzle area (Examination Category B-D), and the dissimilar metal welds joining the reactor vessel nozzles to the reactor coolant loop piping (Examination Category R-A) examined by the automated reactor vessel examination tool.

II. CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1998 Edition with addenda up to and including the 2000 Addenda, IWA-2600, "Weld Reference System."

III. BASIS OF REQUEST FOR RELIEF

The original construction requirements of the Surry Power Station did not require establishing a reference system for the reactor vessel and associated dissimilar metal welds as now required by IWA-2600. An automated examination tool now accomplishes these examinations. The automated examination tool establishes its reference point using an existing zero reference on the reactor vessel. This point allows the device to repeat examination locations without the necessity of any other reference systems. The tool determines its location by the use of an electronic encoder system, which provides for sufficient repeatability. Electronic encoding systems have been in use for the reactor vessel examinations performed for Dominion and the industry for over a decade. Dominion has not identified any concern regarding the use of the system from its staff, the vendor, the ANII, or the regulator. Additionally, Dominion is unaware of an industry concern with this type of location/reference system. It is Dominion's position that the electronic referencing system used by the automated reactor vessel examination tool provides an acceptable level of quality and safety. This alternative system can locate welds with sufficient repeatability for future examinations. Therefore, it will satisfy the objectives of IWA-2600.

The examinations performed by the automatic tool are conducted from the inside of the reactor vessel. Establishing the reference system required by IWA-2600 on the inside of an operational reactor vessel is a significant hardship that will provide no increase in quality or safety. Therefore, approval of this proposed alternative reference system is requested under the provisions of 10 CFR 50.55a(a)(3)(ii).

IV. ALTERNATE PROVISIONS

The automated reactor vessel examination tool will continue to establish its reference system based upon the existing zero reference and the electronic encoding system designed into the tool. No other system is planned or deemed necessary.

(A similar relief request was approved for North Anna Unit 1, third inservice inspection interval, under TAC NO. MA5750; for North Anna Unit 2, the third inservice inspection interval, under TAC NO. MB2280; for Surry Unit 1, third inservice inspection interval, under TAC NO. M87312; and for Surry Unit 2, third inservice inspection interval, under TAC NO. M89085.)

SECTION 2 REQUESTS FOR RELIEF - COMPONENTS

Section 2.2 System Pressure Testing

RELIEF REQUEST SPT-001

I. IDENTIFICATION OF COMPONENTS

Class 1 pressure retaining components.

II. CODE REQUIREMENTS

Relief is requested from the following requirements of Section XI, 1998 Edition with addenda up to and including the 2000 Addenda as follows:

1) Table IWB-2500-1, Category B-P requires that all Class 1 pressure-retaining components receive a system leakage test each refueling outage. Note (2) of the table states, "The system leakage test (IWB-5220) shall be conducted prior to plant startup following (*each*) reactor refueling outage."

2) IWB-5221 (a) requires, "The system leakage test shall be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power."

3) IWA-2212 (b) by reference to Table IWA-2210-1 requires the "maximum examination distance (*as allowed by Table IWA-2210-1*) shall apply to the distance from the eye to the surfaces being examined." The maximum distance allowed by Table IWA-2210-1 is six feet.

III. BASIS OF REQUEST FOR RELIEF

The Class 1 system leakage test is performed at the end of a refueling outage as part of the start-up process. The SPS 1 design takes advantage of subatmospheric pressures within containment to mitigate the consequences of certain accident scenarios. The plant's Technical Specifications require subatmospheric conditions to exist within containment at the system leakage test conditions required by IWB-5221(a) for the Class 1 leakage test. The subatmospheric requirements create conditions that require the use of self-contained breathing apparatus (SCBA) with full-face respirators by anyone required to be in the containment.

The VT-2 visual examination procedure has been demonstrated using no visual aids to a distance of nine feet nine inches using a visual card that complies with the 1998 Edition, 2000 Addenda of the ASME Code. We have evaluated additional remote monitoring equipment and determined they are not practical for inspectors wearing full-face respirators and SCBA. The use of binoculars or a telescope is not feasible because the eyepiece cannot directly be placed to the inspector's eye.

To perform direct examination within the maximum distance requirements of IWA-2212(b) it would be necessary to erect scaffolding to access, within six feet, all surfaces that require examination. The use of scaffolding would only be allowed in containment during unit operation if it has been designed and erected to withstand the design seismic event without causing damage to safety related equipment. The installation of the scaffolding at the end of one outage, and then disassembly at the beginning of the next refueling outage only to start the installation process over at the end of that outage is impractical.

To leave the scaffolding in place until the Class 1 system leakage test is completed and then remove it before proceeding with startup is also impractical. Because of the subatmospheric containment, it would be necessary to either bring the reactor coolant system back to less than or equal to 350^0F and 450 psig; or alternatively, attempt to remove the scaffolding while contending with the subatmospheric conditions. The latter would involve personnel using self-contained breathing apparatus. It is doubtful this would be successful; but regardless of the potential of success, it would be an unreasonable burden for the personnel involved.

ASME Code Interpretation XI-1-98-06 is consistent with this relief request. XI-1-98-06 states:

> Subject: IWA-2210, IWA-2212, and IWA-5240; VT-2 Visual Examination Requirements (1992 Edition Through the 1995 Edition with the 1997 Addenda), Date Issued: January 16, 1998, File: IN97-034
>
> Question (1): Is it a requirement of IWA-2212(b) and Table IWA-2210-1 that all VT-2 examinations be conducted by direct examination?
> Reply (1): No.
>
> Question (2): When items subject to VT-2 examinations are inaccessible for direct examination because the distance requirement is exceeded, does IWA-2210 require a remote examination be performed?
> Reply (2): No. Alternatives are described in IWA-5241 and IWA-5242.
>
> Question (3): When performing a VT-2 visual examination on surrounding areas (including floor areas or equipment surfaces) per IWA-5241 (b) or IWA-5242(b), do the requirements of Table IWA-2210-1 apply to the surrounding area rather than the actual component?
> Reply (3): Yes.

IV. ALTERNATE PROVISIONS

SPS 1 requests approval in accordance with 10 CFR 50.55a(a)(3)(ii) to perform the Class 1 system leakage test without the erection of temporary scaffolding to satisfy the examination requirements of IWA-2212 (b). As an alternative, existing permanent structures, platforms or ladders will be used to the extent practical to gain access to the surface to be examined. The required visual examination will be performed from the access afforded by these structures, ladders or platforms to the extent practical. Any examination surface that cannot be accessed per the requirements of Table 2210-1 or to the maximum qualified distance will be considered "inaccessible". As such the surrounding area (including floor areas or accessible equipment surfaces located underneath the inaccessible components) will be examined for leakage as required by IWA-5241 (b) or IWA-5242 (b).

(Note: A similar request for relief was approved for North Anna Power Station Unit 2 for the third inservice inspection interval under TAC NO. MB2280.)

RELIEF REQUEST SPT-002

I. IDENTIFICATION OF COMPONENTS

Class 3 pressure retaining components and piping.

II. CODE REQUIREMENTS

Section XI, 1998 Edition with addenda up to and including the 2000 Addenda, Table IWD-2500-1, Category D-B, Item Number D2.20, requires a system hydrostatic test in accordance with IWD-5222 be performed once in the inservice inspection interval. IWD-5222 requires the test pressure to be greater than the normal operating pressure.

III. BASIS OF REQUEST FOR RELIEF

This request for relief asks that the test pressure be reduced to the normal operating pressure for the system hydrostatic test required of Class 3 components.

The Section XI Subcommittee determined that the over pressurization requirements of earlier Code editions were excessive and issued Code Case N-498 in various revisions to eliminate the over pressurization requirement. The NRC agreed with the position taken by Code Case N-498-1, "Alternative Rules for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems," dated May 11, 1994. Code Case N-498-1 is currently identified for use in the latest approved revision of Regulatory Guide 1.147, dated May 1999. However, Code Case N-498-1 addresses all three classes of components. Section XI, 1998 Edition with 2000 Addenda, has incorporated the reduced pressure requirements of the Code Case for Class 1 and 2 items, but failed to do so for Class 3 items. A later revision to Section XI (2001 Edition) eliminated the over pressurization testing requirements for Class 3 items.

To continue to perform the over pressurization testing of Class 3 components as part of the fourth inservice inspection interval is considered impractical as both the industry and the NRC have agreed that the benefit to safety does not merit the effort to perform the test at the elevated pressure. The alternative testing proposed in the following paragraph of this request for relief is the same as that required for Class 3 items by Code Case N-498-1. To propose these requirements outside of the Code Case allows the aspects of the Code Case to be applied to the Class 3 components only. It does not cause the need to request the use of only part of the Code Case or to correct what are now incorrect references to specific Section XI requirements contained within the Code Case. (For example, only Examination Category D-B now states system pressure testing requirements for Class 3 components.)

IV. ALTERNATE PROVISIONS

In accordance with 10 CFR 50.55a(a)(3)(i), SPS 1 proposes to use the following requirements as part of its fourth interval inspection ISI Plan for the system pressure testing of Class 3 components and piping as opposed to the requirements of Section XI referenced in Section II of this request.

1) A system pressure test shall be conducted at or near the end of the inservice inspection interval;

2) The boundary subject to test pressurization during the system pressure test shall extend to all Class 3 components included in those portions of systems required to operate or support the safety system function up to and including the first normally closed valve, including a safety or relief valve, or valve capable of automatic closure when the safety function is required;

3) Prior to performing the VT-2 visual examination, the system shall be pressurized to nominal operating pressure for at least 4 hours for insulated systems and 10 minutes for non-insulated systems. The system shall be maintained at nominal operating pressure during performance of the VT-2 visual examination; and

4) The VT-2 visual examination shall include all components within the boundary identified in paragraph IV., 2) above.

5) Test instrumentation requirements of IWA-5260 are not applicable.

(Note: A similar request for relief was approved for North Anna Power Station Unit 2, third inservice inspection interval, under TAC NO. MB2223.)

RELIEF REQUEST SPT-003

I. IDENTIFICATION OF COMPONENTS

Approximately 20, small diameter (≤ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent, drain, sample, and instrumentation connections.

II. IMPRACTICAL CODE REQUIREMENTS

Section XI 1998 Edition with addenda up to and including the 2000 Addenda, Examination Category B–P, Items B15.50 and B15.70 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 inservice inspection interval shall extend to all Class 1 pressure retaining components within the system boundary."

III. BASIS OF REQUEST FOR RELIEF

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 XI 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. IWA–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 ≤ 1 inch diameter RCS vent/drain/sampling connections once each 10–year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI 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.

IV. ALTERNATE PROVISIONS

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 XI 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).

(Note: The NRC approved a similar relief request for Edwin I. Hatch Plant, Units 1 and 2 under TAC Nos. MA2118 and MA2119, respectively. Also, the NRC approved a similar relief request for North Anna Power Station, Unit 1, third inservice inspection interval, under TAC NO. MA5750. Similar requests were approved for North Anna Power Station Unit 2, third inservice inspection interval, under TAC NO. MB2280; and Surry Power Station Units 1 and 2, third inservice inspection intervals, under TAC Nos. MA4979 and MA4980, respectively.)

RELIEF REQUEST SPT-004

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 1 reactor containment is required to be at sub-atmospheric 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 alternative examination would take place. The proposed alternate examination would be performed under conditions that are safer and allow for a more thorough examination.

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 VT-2 visual examination for evidence of boric acid leakage/corrosion will be conducted each refueling outage 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 sub-atmospheric 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 that unit's 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.)

SECTION 3 REQUESTS FOR RELIEF

COMPONENT SUPPORTS

RELIEF REQUEST CS-001

I. IDENTIFICATION OF COMPONENTS

Class 1, 2, and 3 Dynamic Restraints (Snubbers)

II. CODE REQUIREMENTS

The ASME B&PV Code, Section XI, 1998 Edition with addenda up to and including the 2000 Addenda, paragraphs IWF-5200 (a) and (b) and IWF-5300 (a) and (b) require the use of ASME/ANSI OM-1987, Part 4 (published in 1988) Code to perform the preservice and inservice examinations and tests of Class 1, 2, and 3 snubbers.

III BASIS OF REQUEST FOR RELIEF

This action proposes, as an alternative to the requirements of ASME/ANSI OM-1987, Part 4 (published in 1988), to use the existing Surry Power Station, Unit 1 (SPS 1), Technical Specification 4.17, "SHOCK SUPPRESSORS (SNUBBERS)", and specific paragraphs from ASME OMa-1996, Section IST, Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants."

Differences exist between the referenced Code requirements and SPS 1 Technical Specification 4.17, "SHOCK SUPPRESSORS (SNUBBERS)". ASME/ANSI OM-1987, Part 4 (published in 1988) (hereafter to be known as Part 4) contains requirements that were removed from the plant's Technical Specifications as recommended by Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions" (GL 90-09). The generic letter was issued on December 11, 1990 to reduce the burden placed upon licensees by the excessively restrictive inspection schedule then contained in the Technical Specifications. Section XI continues to require the excessively restrictive inspection schedule for snubbers by its requirement to use Part 4 in the preservice and inservice examination and testing of snubbers. A later revision of the Part 4 standard, ASME OMa-1996, Section IST, Subsection ISTD, "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants," (ISTD), changed the excessively restrictive requirements to essentially agree with the recommendations of GL 90-09. The rulemaking revising 10 CFR 50.55a, published in the Federal Register on September 22, 1999, recognized this fact and included in the rule a provision to allow the use of ISTD provided the licensee revises the applicable Technical Specification.

The current testing and examination requirements of Technical Specification 4.17 and the additional visual examination requirements of the approved Section XI edition have formed the basis of the Surry Unit 1 inservice examination/testing program for snubbers for the past ISI interval. The use of this program was approved by the NRC and is essentially the same as the program for examination and testing described in ISTD for inservice inspection. Therefore, SPS 1 continues to believe that the combination of Technical Specification 4.17 and Section XI, excluding paragraphs IWF-5200 and IWF-5300, is an alternative that provides an acceptable level of quality and safety for inservice examination and testing of snubbers. The continued administration of a program based on Technical Specifications and Section XI requires only minimal administrative change activity.

Revising the Technical Specifications to reference only the ISTD document is an administrative activity with little, if any, impact on safety or quality. Many of the requirements being removed from the Technical Specification are very similar if not identical to the requirements being added by ISTD. Additionally, the same is true of the reviews and revisions that would also be required of the involved procedures. SPS 1 determined that the proposed alternative approach avoids this unnecessary administrative impact and still provides a means to accomplish the inservice examination and testing intended by Section XI and regulation.

However, to satisfy the preservice examination and testing requirements intended by Section XI, additional examination and testing activity is required beyond the above proposal for inservice activities. Therefore, SPS 1 proposes that the inclusion of the examination and testing requirements contained in paragraphs ISTD 4, "Preservice Examination" (excluding paragraph 4.3) and ISTD 5, "Preservice Operability Testing" into the current snubber surveillance program provides an alternative with an acceptable level of safety and quality for the preservice examination and testing requirements. The inclusion of these requirements into the snubber surveillance program achieves the preservice inspection requirements of Section XI.

Paragraph 4.3 is not proposed for incorporation into the alternative examination program as it addresses requirements best suited for the initial heat up and cool down of the plant. It would be a hardship to try and impose these requirements on an operating plant such as SPS 1, which is constructed with a sub-atmospheric containment. As an alternative to the requirements of paragraph 4.3, SPS 1 will follow the guidance of IWF-2220(b) for systems that operate at a temperature greater than 200 degrees F. IWF-2220(b) requires the Owner to perform an additional preservice examination on the affected component supports during or following the subsequent system heatup and cooldown cycle unless determined unnecessary by evaluation. The examination may be performed either during operation or at the next refueling outage. This Section XI examination requirement has been accepted by regulation as providing acceptable quality and safety for supports, and SPS 1 believes it is acceptable as an alternative to paragraph 4.3 of ISTD. No other requirements of ISTD will be implemented as part of this alternative.

IV. ALTERNATIVE PROVISIONS

SPS 1 proposes as an alternative to Section XI requirements stated in Section II of this request, a snubber examination and testing program comprised of the following elements:

1) The continued implementation of the surveillance requirements of Technical Specification 4.17, "SHOCK SUPPRESSORS (SNUBBERS)", without change;

2) The implementation of the other applicable requirements of the ASME, Section XI Code, 1998 Edition with addenda up to and including the 2000 Addenda, unless specific approval has been obtained to do otherwise from the NRC pursuant to the requirements of 10CFR 50.55a and the plant's Technical Specifications. This will include using the VT-3 visual examination method in IWA-2213 for preservice and inservice examinations;

3) The preservice examination and testing requirements of ISTD paragraph 4, "Preservice Examination", excluding paragraph 4.3; and ISTD paragraph ISTD 5, "Preservice Operability Testing;" and

4) As an alternative to paragraph 4.3 of ISTD, for systems that operate at a temperature greater than 200 degrees F, SPS 1 will perform an additional preservice examination on the affected snubbers during or following the subsequent system heatup and cooldown cycle unless determined unnecessary by evaluation. This examination may be performed during operation of the plant or at the next refueling outage. This is consistent with the requirements of IWF-2200(b). No other requirements of ISTD will be implemented as part of this alternative.

SPS 1 submits that the above alternative program provides an acceptable level of quality and safety without the burden of administrative changes that add little or no value to quality or safety, or the hardship of performing snubber assessments under sub-atmospheric conditions. Therefore, having met the criteria of 10 CFR 50.55a(a)(3)(i), authorization to implement the proposed alternative program as part of the fourth inservice inspection interval is requested.

(Note: A similar request for relief was approved for North Anna Unit 2 for its third inservice inspection interval under TAC NO. MB2280.)

SECTION 4 MISCELLANEOUS DOCUMENTATION

(Reserved for use in a later revision.)

SECTION 5 REQUESTS FOR RELIEF

PARTIAL COVERAGE

No requests for relief involving partially completed examinations or tests have been prepared to date. These requests for relief will be prepared after the completion of each involved examination or test.

SECTION 6 CODE CASES

Section 6.1 Code Cases Utilized in the Fourth Inservice Inspection Interval

CODE CASES UTILIZED IN THE FOURTH INSERVICE INSPECTION INTERVAL

Case N-416-1	"Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3 Section XI, Division 1." (Approved with conditions by NRC as stated in Regulatory Guide 1.147, Rev. 12, May, 1999. The following condition is a requirement for use of CC N-416-1 and as such is part of the requirements of this ISI Plan: *"Code Case N-416-1 is acceptable subject to the following condition in addition to those conditions specified in the Code Case. Additional surface examinations should be performed on the root (pass) layer of butt and socket welds of pressure retaining boundary of Class 3 components when the surface examination method is used in accordance with Section III.")*
Case N-460	"Alternative Examination Coverage for Class 1 and Class 2 Welds."
Case N-513	"Evaluation Criteria for Temporary Acceptance of Flaws in Class 3 Piping." (10 CFR 50.55a places the following restrictions on the use on Code Case N-513: *1) The specific safety factors in paragraph 4.0 must be satisfied; 2) May not be applied to components other than pipe and tubing, such as pumps, valves, expansion joints, and heat exchangers; 3) May not be applied to leakage through a flange gasket; 4) May not be applied to threaded connections employing nonstructural seal welds for leakage prevention [through seal leakage is not a structural flaw, thread integrity must be maintained]; and 5) May not be applied to degraded socket welds.)*
Case N-522	"Pressure Testing of Containment Penetration Piping." (Code Case will be implemented with the following condition as specified in Regulatory Guide 1.147; *"The test should be conducted at the peak calculated containment pressure and the test procedure should permit the detection and location of through-wall leakage in containment isolation valves (CIVs) and pipe segments between the CIVs.")*
Case N-523-1	"Mechanical Clamping Devices for Class 2 and 3 Piping." (Note: the Code Case erroneously references Table NC-3321-2, rather than Table NC-3321-1.)

Case N-532*	"Alternative Requirements to Repair Replacement Documentation Requirements and Inservice Summary Report Preparation and by IWA-4000 and IWA-6000." (See Relief Request CC-001.)
Case N-533-1*	"Alternative Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections." (See Relief Request CC-002.)
Case N-566-1*	"Corrective Action for Leakage Identified at Bolted Connections." VT-1 examinations will be performed in lieu of the VT-3 examination on the removed bolting. (See Relief Request CC-003.)
Case N-597*	"Requirements for Evaluation of Pipe Wall Thinning." (See Relief Request CC-004.)
Case N-623*	"Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel." (See Relief Request CC-005.)

A Code Case identified with an asterisk (*) is not approved for use unless the NRC has granted the referenced request for relief to implement the Code Case.

SECTION 6 CODE CASES

Section 6.2 Requests for Relief

RELIEF REQUEST CC-001

I. IDENTIFICATION OF COMPONENTS

Class 1, 2, and 3 components and their supports

II. CODE REQUIREMENTS

Articles IWA-6000 and IWA-4000 of Section XI, 1998 Edition with addenda up to and including the 2000 Addenda, require the Owner to prepare and submit the "Form NIS-1 Owner's Report for Inservice Inspection" and "Form NIS-2 Owner's Report for Repair/Replacement Activity" for Class 1 and 2 components. Additionally, the Form NIS-2 is prepared and maintained in the Owner's records for Class 3 components.

III. BASIS OF REQUEST FOR RELIEF

SPS 1 proposes to implement as part of its fourth interval inspection ISI Plan Code Case N-532, "Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000," as an alternative to the Section XI Code requirements referenced above. SPS 1 has reviewed both the requirements of the Section XI Code and Code Case N-532 and has determined that the Code Case provides an adequate level of quality and safety as required by 10 CFR 50.55a(a)(3)(i). This conclusion is based on the following:

1) Code Case N-532 does not change or alter the Section XI requirement that each repair/replacement activity shall be reviewed by the Authorized Nuclear Inservice Inspector (ANII). This is a review performed by a qualified inspector who is knowledgeable in the requirements of Section XI. The objective of this review, i.e. an independent review by a party other than the Owner to assure compliance with Section XI requirements, is not changed by the Code Case. The NIS-2 submittal does not provide any information to verify that the repair/replacement activity was performed in accordance with Section XI other than some hydrostatic test parameters and the use of certain Code Cases. This is only a small amount of the information needed to verify Code compliance of a repair/replacement activity. Additionally, no information on repair/replacement activities of Class 3 components is required to be submitted. Therefore, the review by the ANII is the primary third party assurance activity associated with repair/replacement activities.

2) Code Case N-532 does not change the requirement that the documentation associated with a repair/replacement activity required by Section XI and the Owner's Quality Assurance Program be maintain at the

site for review by the NRC and/or representatives of the Owner or personnel from other appropriate organizations.

3) Code Case N-532 does not change the Owner's commitments to maintain an effective in-process quality assurance program to control applicable aspects of the repair/replacement activity in accordance with Appendix B to 10 CFR 50.

4) Code Case N-532 does require an abstract for repairs, replacements and corrective measures performed, even though the discovery of the flaw or relevant condition that necessitated the repair/replacement activity may not have resulted from an examination or test required by Section XI. This provides a significant improvement in reporting requirements associated with repair/replacement activities and should provide more meaningful information to the interested parties who do not have immediate access to the full body of documentation available at the site.

5) Code Case N-532 no longer requires that the examinations completed be reported examination by examination as required by Section XI. However, evidence of these examinations is maintained as required by Section XI and the records retention requirements of the Owner's quality assurance program. These records are available for review by the ANII, NRC, and representatives of other appropriate organizations. Code Case N-532 does require that a status report be provided by examination category. This "status report" provides a more meaningful and comprehensive assessment of the program's status than what is currently required.

6) Code Case N-532 does allow the report interval to be based on an inspection period as opposed to each fuel cycle. This would change the reporting frequency for SPS 1 from seven times in the inservice inspection interval to three. However, the NIS-1 and NIS-2 reports are not intended to provide timely notification of activities to regulators of industry events. Rather these currently required reports are summary reports of normal ISI activities. All other reporting commitments of SPS 1 remain unaffected by the use of this Code Case.

The use of Code Case N-532 was previously approved by the NRC for Wolf Creek Generating Station in a letter dated February 9, 1996. The safety evaluation included in the NRC approval letter noted a clarification to the term "corrective measures" as used in paragraph 2.0 (c) of the Code Case. It was noted that one definition of the term involves Code required activities such as repair and replacement. The other definition of the term involves maintenance activities such as tightening threaded fittings to eliminate leakage, torquing of fasteners to eliminate leakage at bolted connections, replacing valve packing due to unacceptable packing leakage, tightening loosened mechanical connections on supports, adjustment and realignment of supports, cleanup of corrosion on components from leakage, etc. It is our intent to use the same clarification proposed and accepted for Wolf Creek Generating Station, i.e., the first definition where the term "corrective measures" is a reference to repair/replacement activities. The term will not apply to normal maintenance activities that are not considered repair/replacement activities.

IV. ALTERNATE PROVISIONS

As an alternative to the requirements of IWA-6000 and IWA-4000, in accordance with 10 CFR 50.55a(a)(3)(i), SPS 1 proposes to use Code Case N-532, with the clarification of the term "corrective action" as discussed in Section III, in the implementation of its Section XI fourth interval inspection ISI Plan.

(Note: A similar relief request was approved for North Anna Unit 1 for the third interval inspection ISI Program and for the NAPS 2 third interval inspection ISI Program under TAC Nos. MA6878 and MB2280, respectively. Similar relief was also granted to the Surry Units 1 and 2 third interval ISI inspection programs under TAC Nos. MA6880 and MA6881, respectively.

RELIEF REQUEST CC-002

I. IDENTIFICATION OF COMPONENTS

Class 1, 2, and 3 bolted connections in systems borated for the purpose of controlling reactivity.

II. CODE REQUIREMENTS

Section XI 1998 Edition with addenda up to and including the 2000 Addenda, paragraph IWA-5242(a) states in part that "For systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure-retaining bolted connections for VT-2 visual examination..."

III. BASIS OF REQUEST FOR RELIEF

The ASME Section XI Subcommittee has approved and published Code Case N-533-1, "Alternative Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections," dated February 26, 1999. This Code Case provides an alternative to the requirements of IWA-5242(a) that allows for the removal of insulation and examination of the bolted connections without the connection being pressurized. It also requires a system pressure test and associated VT-2 examination without the removal of insulation. The method proposed by the Code Case is well suited to the examination of borated systems in that boron that leaks out of the system leaves a detectable residue. Therefore it will be detected by a VT-2 visual examination. Because the use of this Code Case is requested for borated systems, SPS 1 has determine that it provides an acceptable level of quality and safety.

The use of this Code Case is considered to be necessary by SPS 1, because the majority of involved bolted connections are located inside containment, and potentially in high radiation areas. Additionally, many of the connections are required to be assessed by the Class 1 system leakage test conducted as part of the start up testing following a refueling. During this test, SPS 1 is required by Technical Specifications to maintain sub-atmospheric conditions inside its containment. Because of the sub-atmospheric conditions, any personnel inside containment must wear a self-contained breathing apparatus. This would make the tear down and removal of scaffolding a task of significant hardship that provides no gain in safety, if the proposed alternative is utilized. (The scaffolding is required to gain access to the bolted connections and to reinstall the insulation after completion of the test.) Secondly, most of the involved connections are tested at elevated temperatures. The actions necessary to protect the personnel who must re-install insulation on these connections in systems containing fluids at substantially elevated temperatures and remove scaffolding in the proximity of the same connections represent another hardship with little or no improvement in safety.

IV. ALTERNATE PROVISIONS

SPS 1 requests approval, as allowed by 10 CFR 50.55a(a)(3)(i), to use Code Case N-533-1, "Alternative Requirements for VT-2 Visual Examinations of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections," dated February 26, 1999, as an alternative to the requirements of IWA-5242(a) that relate to bolted connections in systems borated for the purpose of controlling reactivity. In addition to complying with the requirements of paragraphs (a) and (b) of Code Case N-533-1, SPS 1 will also assure that a 4-hour hold time is achieved as part of the system leakage test required by paragraph (a) of the Code Case before the VT-2 visual examination associated with the system leakage test is performed.

(Note: A similar request for relief was approved for North Anna Power Station Unit 2 for its third inservice inspection interval under TAC NO. MB2280.)

RELIEF REQUEST-CC-003

I. IDENTIFICATION OF COMPONENTS

Pressure retaining bolted connections within the scope of ASME Section XI.

II. CODE REQUIREMENTS

Section XI 1998 Edition with addenda up to and including the 2000 Addenda, paragraph IWA-5250(a)(2) requires, in part, that "If leakage occurs at a bolted connection in a system borated for the purpose of controlling reactivity, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100."

III. BASIS OF REQUEST FOR RELIEF

Section XI requires the bolting to be removed and evaluated even if sufficient evidence exist to support the conclusion that the involved bolting has not been harmed by the leakage. Such factors as the time in service of the bolts or the susceptibility of the bolting material to corrosion by the leaking liquid may not be used to justify leaving bolting material in service without further examination. Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections," dated February 15, 1999, used in lieu of the Section XI requirements would allow greater flexibility and prudent decision making. Leaking conditions at a bolted connection may be an important factor in the degradation of bolting. However, the removal of bolting unnecessarily may result in damage of sound bolting, the exposure of personnel to radiation, and the expenditure of resources for no gain in safety. Code Case N-566-1 provides a basis for determining the acceptability of bolting based upon several factors including material, leaking medium, duration of the leak, general corrosion of the connection and the impact of such leakage on the system. An analysis to determine the need to remove a bolt for examination prior to any action to remove the bolting is required by Code Case N-566-1. This is an alternative to the requirements of Section XI that provides an acceptable level of quality and safety.

IV. ALTERNATE PROVISIONS

SPS 1 requests approval in accordance with 10 CFR 50.55a(a)(3)(i) to use Code Case N-566-1, "Corrective Action for Leakage Identified at Bolted Connections," dated February 15, 1999, as part of its fourth inservice inspection interval. In addition, as agreed with the Nuclear Regulatory Commission in gaining approval for a similar request for relief for North Anna Unit 2 third inservice inspection interval, if the evaluation determines that examination is required, a VT-1 visual examination will be performed on the removed bolting in lieu of the Code required VT-3 visual examination.

(Note: A relief request to use Code Case N-566-1 was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program, under TAC NO. MA5750; and for North Anna Unit 2, third inservice inspection interval, under TAC NO. MB2233.)

RELIEF REQUEST CC-004

I. IDENTIFICATION OF COMPONENTS

Class 2 and 3 carbon and low-alloy steel piping components subject to wall thinning as a result of flow-accelerated corrosion (FAC) phenomena. (Note: Corrosion phenomena other than FAC are not within the scope of this request for relief, even though they are within the scope on Code Case N-597.)

II. CODE REQUIREMENTS

ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition with addenda up to and including the 2000 Addenda, IWA-4611.1, provides the process for assessing a component for continued service after a defect has been removed. This paragraph stipulates that where the section thickness has been reduced below the minimum design thickness, the component shall be repaired. As an alternative, the component may be evaluated and accepted in accordance with the design rules of either the Construction Code or ASME Section III.

III. BASIS OF REQUEST FOR RELIEF

Code Case N-597, "Requirements for Analytical Evaluation of Pipe Wall Thinning," dated March 2, 1998, Section XI, Division 1, provides an acceptable approach for determining the structural integrity of components degraded by wall thinning as a result of flow-accelerated corrosion phenomena. The methodology specified in the code case is sufficiently conservative, and thus provides an adequate margin of safety for evaluating a component's structural integrity. Maintaining the option of using Code Case N-597 will facilitate the scheduling of the replacement of components degraded by wall thinning as a result of flow-accelerated corrosion phenomena. To be required to comply with IWA-4611.1 by performing unplanned, and therefore unscheduled repairs, in lieu of a determination of acceptance via Code Case N-597, is a hardship without a compensating increase in the level of quality and safety. Therefore, the option to implement Code Case N-597 is requested as allowed by 10 CFR 50.55a(a)(3)(ii).

IV. ALTERNATIVE PROVISIONS

As an alternative to the requirements of IWA-4611.1, we propose to establish the option of using ASME Code Case N-597, which permits carbon and low-alloy steel piping components subjected to wall thinning as a result of flow-accelerated corrosion phenomena to be evaluated and accepted in accordance with an alternative set of design rules to those contained in the Construction Code or ASME Section III. Further, procedural controls are currently in place in the Virginia Electric and Power Company (Dominion) Flow-Accelerated Corrosion Program that provide direction for calculating wear rates, forecasting remaining life, and conducting inspections of piping components susceptible to FAC. The methodology employed is consistent with EPRI's NSAC-202L-R2, "Recommendations for an Effective Flow-Accelerated Corrosion Program," and incorporates the pertinent NSAC-202L-R2 recommendations for calculating wear rates, forecasting remaining life, and conducting inspections as programmatic requirements.

For example, if the evaluation of ultrasonic test (UT) data for a component determines that significant wall thinning has occurred, additional data acquisition shall be performed (including UT grid refinement and/or scans) to further establish that the lowest wall thickness has been measured. After the additional UT data have been obtained, wear rates are calculated and the remaining wall thickness is projected for the end of the next fuel cycle. The projected remaining wall thickness is then analyzed in accordance with Code Case N-597, and a schedule for future component replacement is determined. The component will be inspected during each subsequent refueling outage until it is replaced consistent with the schedule determined by the Code Case N-597 evaluation.

Additionally, implementation of NSAC-202L-R2 recommendations for calculating wear rates, forecasting remaining life, and conducting inspections as programmatic requirements will be as follows: an NSAC recommendation identified by "shall" - is a mandatory requirement; and a recommendation identified by "should" - is a non-mandatory requirement; however, it is the preferred/desired method to be adhered to unless the FAC program administrator or management determines otherwise.

(Note: The NRC has previously approved the use of Code Case N-597 for use as part of the North Anna Unit 1, third interval program, under TAC NO. MB2223; North Anna Unit 2, third interval program, under TAC NO. MB2284; Surry Power Unit 1 third interval program, under TAC NO. MB2276; and Surry Unit 2, third interval program, under TAC NO. MB2277.)

RELIEF REQUEST CC-005

I. IDENTIFICATION OF COMPONENTS

Shell-to-Flange Weld of the Reactor Vessel, Category B-A, Item B1.30.

II. CODE REQUIREMENTS

Section XI of the ASME Boiler and Pressure Vessel Code, 1998 Edition with addenda up to and including the 2000 Addenda requires the following:

1) Note (3) to Table IWB 2500-1, Examination Category B-A states, "(3) When using Inspection Program B, the shell-to-flange weld examination may be performed during the first and third periods, in which case 50% of the shell-to-flange weld shall be examined by the end of the first period, and the remainder by the end of the third period. During the first period, the examination need only be performed from the flange face, provided this same portion is examined from the shell during the third period."

2) IWB-2420 (a) requires, "The sequence of component examinations which was established during the first inservice inspection interval shall be repeated during each successive inservice inspection interval to the extent practical."

III. BASIS OF REQUEST FOR RELIEF

On February 26, 1999 the ASME Boiler and Pressure Vessel Code approved Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel." Although this Code Case allows deferral of both the shell-to-flange and head-to-flange welds, this request for relief is applicable only to the shell-to-flange weld. The Code Case allows deferral of the shell-to-flange weld provided three conditions are met:

(a) No welded repair/replacement activities have ever been performed on the shell-to-flange weld. Compliance: No repair/replacement activity has been performed on the shell-to-flange weld.

(b) The shell-to-flange weld contains no identified flaws or relevant conditions that currently require successive inspections in accordance with IWB-2420 (b). Compliance: No successive inspections in accordance with IWB-2420 (b) are now or have ever been required on the shell-to-flange weld.

(c) The vessel is not in the first inservice inspection interval. Compliance: The reactor vessel will be in its fourth inservice inspection interval.

Therefore the conditions of Code Case N-623 have been satisfied.

The effect of IWB-2420 is to require the examinations of the shell-to-flange weld to be repeated on a 10-year schedule to the extent practical. If the examinations were divided between periods, as allowed by Note 3 to Table IWB 2500-1, Category B-A, this requirement would cause the initial schedule of each partial examination to be maintained. The technical justification of this requirement is to assure that components are not allowed to go excessive periods of time before reexamination. In anticipation of proposing this alternative, Surry Power Station plans to perform essentially a 100% examination of the shell-to-flange weld in the third period of the third inservice inspection interval. This will not only satisfy the examinations required by the third period but also repeat the examinations performed in the first period. It allows for the fourth interval deferral of all shell-to-flange weld examinations to the third period while still maintaining the objective of IWB-2420 (a).

Having satisfied the requirements of Code Case N-623 and having performed an essentially 100% examinations of the shell-to-flange weld in the third period of the third inservice inspection interval, SPS 1 proposes that the deferral of the examination of 100% of the shell-to-flange weld to the third period of the fourth inservice inspection interval provides an acceptable level of quality and safety. Approval for the deferral is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

IV. ALTERNATE PROVISIONS

As an alternative to the requirements of Note 3 to Table IWB-2500-1, Category B-A, all examinations of the shell-to-flange weld will be deferred to the third period of the fourth inservice inspection interval by the implementation of Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel," dated February 26, 1999.

(Note: A similar request for relief was approved for North Anna Power Station, Unit 2, third inservice inspection interval, under TAC NO. MB2280.)

SECTION 7 RISK INFORMED - ISI

REQUESTS FOR RELIEF

RELIEF REQUEST R-001

I. IDENTIFICATION OF COMPONENTS

ASME Class 1, 2, 3, and non-class socket weld connections and their branch connections, nominal pipe size 2 inches (NPS 2) and smaller, which are identified as being High Safety-Significant (HSS).

II. CODE REQUIREMENTS

As a pilot application, SPS Unit 1 was approved to use a Risk-Informed Inservice Inspection (RI-ISI) program on piping and dissimilar metal welds for the remainder of its operational life. Code Case N-577, Table 1, Examination Category R-A; and WCAP-14572, Rev. 1-NP-A, Table 4.1-1 both require examination of HSS components based upon the postulated failure mechanism for the element of piping being examined. The requirement does not account for the geometric limitations imposed by socket welds and their branch connections, NPS 2 and smaller, when volumetric examinations are specified. As such, the current requirement is considered impractical.

III. BASIS OF REQUEST FOR RELIEF

Certain socket weld connections and their branch connections, NPS 2 and smaller, for Surry Unit 1 have been identified as HSS and require volumetric examination for their postulated failure mechanism by WCAP-14572, Rev. 1-NP-A. These instances are associated with a potential thermal fatigue damage mechanism either caused by a postulated temperature stratification or as a default mechanism for segments selected for their consequence of failure with no assumed active mechanism occurring. Performing a volumetric examination on a socket weld connection or the branch connection NPS 2 and smaller provides little or no benefit due to limitations imposed by the joint configuration and the smaller pipe size.

The ASME Code Committee has recognized this problem and has revised Code Case N-577 to allow substitution of the VT-2 examination method for all damage mechanisms on socket weld connections selected as HSS. The revised version, N-577-1, has been issued and provides the substitution in note 12 of Table 1 of the Code Case. Incorporation of the branch connection, NPS 2 and smaller, into the Code Case is now under consideration by the committee for similar size and joint configuration limitation reasons.

Performing a volumetric examination on socket weld connections or their branch connections, NPS 2 and smaller, would result in unusual difficulty without providing any meaningful results, and thus no compensating increase in the level of quality and safety. As such, relief is requested per 10 CFR 50.55a(a)(3)(ii).

Substituting a VT-2 examination as an alternative on a refueling outage frequency for these locations ensures reasonable assurance of component integrity.

IV. ALTERNATIVE PROVISIONS

A VT-2 exam will be performed on the subject socket weld connections and their branch connections, NPS 2 and smaller, on a refueling outage frequency while the component is pressurized.

(Note: Similar relief requests, including the branch connection relief, have been previously approved by the NRC for Sequoyah Units 1 and 2 and Watts Bar Unit 1. The Sequoyah Units 1 and 2 approval letters were dated October 19, 2001 and October 23, 2001, TAC Nos. MB1566 and MB1567, respectively; and the Watts Bar Unit 1 approval letter was dated January 24, 2002, TAC No. MB2082.)

Attachment 2

Inservice Inspection Schedule Summary
Surry Power Station Unit 1

Virginia Electric and Power Co.
(Dominion)

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
B-A		Reactor Vessel							
	B1.11	Circumferential shell welds	3	3-UT	0	0	3	3	
	B1.12	Longitudinal	4	4-UT	0	0	4	4	
	B1.21	Circumferential head welds	1	1-UT	0	0	1	1	
	B1.22	Meridional head welds	0	0	0	0	0	0	No Item B1.22 examinations are scheduled.
	B1.30	Shell-to-flange welds	1	1-UT	0	0	1	1	RR CMP-008.
	B1.40	Head-to-flange welds	1	1-UT/MT	1/3 of weld	1/3 of weld	1/3 of weld	1	
	B1.50	Repair Welds	0	0	0	0	0	0	No Item B1.50 examinations are scheduled.
		Total	10	9-UT 1-UT/MT	0 1/3 of weld	0 1/3 of weld	9 1/3 of weld	9 1	
		Percentages for Category			3.3	3.3	93.3	100	See IWB-2412(a).
B-B		Vessels Other Than Reactor Vessel							
	B2.11	Pressurizer - Shell to head circumferential welds	2	2-UT	1/2 of weld	1/2 of weld	1	2	
	B2.12	Pressurizer - Longitudinal welds (1' of one weld per head)	2	2-UT	1/2 of weld	1/2 of weld	1	2	
	B2.20	Pressurizer head welds	0	0	0	0	0	0	No Item B2.20 examinations are scheduled.
	B2.30	SG (primary side) head welds	0	0	0	0	0	0	No Item B.30 examinations are scheduled.
	B2.40	SG(primary side) tubesheet to head	3	1-UT	1/3 of weld	1/3 of weld	1/3 of weld	1	
	B2.50	Heat Exchangers (primary side) – Head welds	0	0	0	0	0	0	No Item B2.50 examinations are scheduled.
	B2.60	Heat Exchangers (primary side) - Shell tubesheet to head	0	0	0	0	0	0	No Item B2 60 examinations are scheduled.
	B2.70	Heat Exchangers (primary side) - Shell longitudinal Welds	0	0	0	0	0	0	No Item B2.70 examinations are scheduled.
	B2.80	Heat Exchangers (primary side) Shell tubesheet-to-shell	0	0	0	0	0	0	No Item B2.80 examinations are scheduled.
		Total	7	5-UT	1-1/3	1-1/3	2-1/3	5	
		Percentages for Category			**26.7**	**26.7**	**46.6**	**100**	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
B-D		**Full Penetration Welds of Nozzles in Vessels**							
	B3.90	Reactor Vessel – Nozzle-to-vessel welds	6	6-UT	0	0	6	6	
	B3.100	Reactor Vessel – Nozzle inside radius section	6	6-UT	0	0	6	6	
	B3.110	Pressurizer – Nozzle-to-vessel welds	0	0	0	0	0	0	No Item B3.110 examinations are scheduled.
	B3.120	Pressurizer – Nozzle inside radius section	6	5-UT 7-VT-2	2-UT 2-VT-2	2-UT 3-VT-2	1-UT 2-VT-2	5-UT 7-VT-2	See RR CMP-001. Examination requirements are to the 1998 Edition per 10CFR50.55a.
	B3.130	SG (primary side) – Nozzle-to-vessel welds	0	0	0	0	0	0	No Item B3.130 examinations are scheduled.
	B3.140	SG (primary side) – Nozzle inside radius	6	6-Enhanced Visual	2-Enhanced Visual	2-Enhanced Visual	2-Enhanced Visual	6-Enhanced Visual	The rule continues to require this examination even though the Code removed the exam in the 1999 Addenda. Per 10CFR50.55a(b)(2)(xxi)(A) in lieu of the UT required by the Code, a visual examination with enhanced magnification that has a resolution sensitivity to detect a 1-mil width wire or crack, utilizing the allowable flaw length criteria of Table IWB-3512 may be performed.
	B3.150	Heat Exchangers (primary side) - Nozzle-to-vessel welds	0	0	0	0	0	0	No Item B3.150 examinations are scheduled.
	B3.160	Heat Exchangers (primary side) -.Nozzle inside radius section	0	0	0	0	0	0	No Item B3.160 examinations are scheduled.
		Total	24	17-UT 7-VT-2 6-Enhanced Visual	6	7	17	30	
		Percentages for Category			**20**	**23.3**	**56.7**	**100**	Percentage of examinations planned is acceptable per IWB-2412(a).

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
B-F		**Pressure Retaining Dissimilar Metal Welds**	NA	NA	NA	NA	NA	NA	No Item B-F examinations are scheduled. B-F welds are included in the RI-ISI alternative selection process. See Category "R-A" at the end of this table and Section 7 of the ISI Plan.
B-G-1		Pressure Retaining Bolting, > 2 inch							
	B6.10	Reactor vessel – Closure head nuts	58	58-VT-1	20	18	20	58	
	B6.20	Reactor vessel- Closure studs, in place	0	0	0	0	0	0	No Item B6.20 examinations are scheduled.
	B6.30	Reactor vessel – Closure head studs when removed	58	58-MT or UT	20	18	20	58	
	B6.40	Reactor vessel – Threads in flange	58	58-UT	0	0	58	58	
	B6.50	Reactor vessel - Closure washers	116	116-VT-1	40	36	40	116	
	B6.60	Pressurizer – Bolts and Studs	0		0	0	0	0	No Item B6.60 examinations are scheduled.
	B6.70	Pressurizer - Flange Surface	0	0	0	0	0	0	No Item B6.70 examinations are scheduled.
	B6.80	Pressurizer - Nuts, bushings, and washers	0	0	0	0	0	0	No Item B6.80 examinations are scheduled.
	B6.90	Steam Generator - Bolts and studs	0	0	0	0	0	0	No Item B6.90 examinations are scheduled.
	B6 100	Steam Generator - Flange surface	0	0	0	0	0	0	No Item B6.100 examinations are scheduled.
	B6.110	Steam Generator - Nuts, bushings, and washers	0	0	0	0	0	0	No Item B6.110 examinations are scheduled.
	B6.120	HX's - Bolts and studs	0	0	0	0	0	0	No Item B6.120 examinations are scheduled.
	B6.130	HX's - Flange surface	0	0	0	0	0	0	No Item B6.130 examinations are scheduled.
	B6.140	HX's - Nuts, bushings and washers	0	0	0	0	0	0	No Item B6.140 examinations are scheduled.
	B61.50	Piping - Bolts and studs	0	0	0	0	0	0	No Item B6.150

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
									examinations are scheduled.
	B6.160	Piping - Flange surfaces	0	0	0	0	0	0	No Item B6.160 examinations are scheduled.
	B6.170	Piping	0	0	0	0	0	0	No Item B6.170 examinations are scheduled.
	B6.180	Pumps – Bolts and studs	72	24-UT	8	8	8	24	
	B6.190	Pumps – Flange surface, when connection disassembled	3	3-VT-1	3	3	3	3	Examine only if disassembled. Each pump flange scheduled each period due to unknown work schedule.
	B6.200	Pumps - Nuts, bushings and washers	0	0	0	0	0	0	No Item B6 200 examinations are scheduled.
	B6 210	Valves – Bolts and studs	156	26-UT	0	13	13	26	
	B6.220	Valves – Flange surface when connection is disassembled	6	6-VT-1	6	6	6	6	Examine only if disassembled. Each valve flange scheduled each period due to unknown work schedule.
	B6 230	Valves – Nuts, bushings and washers	156	26-VT-1	0	13	13	26	
		Total	683	209-VT-1 58-MT or UT 108-UT	97	115	181	375	
		Percentages for Category			**24.7**	**29.3**	**46.0**	**100**	
B-G-2		**Pressure Retaining Bolting, ≤ 2inch**							
	B7.10	Reactor Vessel - Bolts, studs, and nuts	0	0	0	0	0	0	No Item B7.10 examinations are scheduled.
	B7.20	Pressurizer – Bolts, studs, and nuts	16	16-VT-1	5	5	6	16	
	B7.30	Steam generators – Bolts, studs, and nuts	96	96-VT-1	32	32	32	96	
	B7.40	Heat Exchangers - Bolts, studs and nuts	0	0	0	0	0	0	No Item B7.40 examinations are scheduled.
	B7.50	Piping – Bolts, studs, and nuts	60	60-VT-1	12	16	32	60	
	B7.60	Pumps – Bolts, studs, and nuts	36	12-VT-1	8	4	0	12	Examination is required only if the associated pump is selected for examination.
	B7.70	Valves – Bolts, studs, and nuts	328	68-VT-1	16	20	32	68	Examination is required only if the associated valve

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
									Is selected for examination.
	B7.80	CRDM Housing - Bolts, studs, and nuts	12	12	4	4	4	12	Examination is required whenever a CRDM housing is disassembled and the bolting is reused. Examination requirements are to the 1995 Addenda per the requirements of 10CFR50 55a.
		Total	548	264-VT-1	77	81	106	264	
		Percentages for Category			**29.2**	**30.7**	**40.1**	**100.0**	
B-J		Pressure Retaining Welds in Piping	NA	NA	NA	NA	NA	NA	No Item B-J examinations are scheduled. B-J welds are included in the RI-ISI Alternative selection process. See Category "R-A" at the end of this table and Section 7 of the ISI Plan.
B-K		**Integral Attachments for Piping, Pumps, and Valves**							
	B10.10	Pressure Vessels	5	1-UT 4 - MT	1/3 of weld 1	1/3 of weld 1	1/3 of weld 2	5	The ultrasonic examination of the pressurizer support skirt weld is to the 1995 Edition per the requirements of 10CFR 50.55a.
	B10.20	Piping	144	15-PT	5	10	8	23	
	B10.30	Pumps							No Item B10.30 examinations are scheduled.
	B10.40	Valves							No Item B10.40 examinations are scheduled.
		Total	149	1-UT 4-MT 15-PT	6-1/3	11-1/3	10-1/3	28	
		Percentages for Category			**22.6**	**40.5**	**36.9**	**100**	
B-L-1	B12.10	**Pressure Retaining Welds in Pump Casings**	3	1-VT-1	0	0	1	1	
		Total	3	1-VT-1	0	0	1	1	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Percentages for Category			0	0	100	100	See IWB-2412(a).
B-L-2	B12.20	Pump Casings	3	1-VT-3	0	0	1	1	Examine only if disassembled for maintenance or repair. Only one pump of the three pumps needs to be examined.
		Total	3	1-VT-3	0	0	1	1	
		Percentages for Category			0	0	100	100	See IWB-2412(a)(4).
B-M-1	B12.30 and B12.40	Valves, Less than NPS 4 with welds Valves NPS 4 and greater with welds	0 0	0 0	0 0	0 0	0 0	0 0	No Item B12.30 or B12.40 examinations are scheduled.
B-M-2	B12.50	Valve Bodies	34	6-VT-3	2	2	2	6	Examination of only one valve in each valve group is required, and only if disassembled for maintenance or repair.
		Total	34	6-VT-3	2	2	2	6	
		Percentages for Category			33.3	33.3	33.4	100	
B-N-1	B13.10	Interior of Reactor Vessel	1	3-VT-3	1	1	1	3	
		Total	1	3-VT-3	1	1	1	3	
		Percentages for Category			33.3	33.3	33.4	100	
B-N-2		Integrally Welded Core Support Structures and Interior Attachments to Reactor Vessels							
	B13 50	Interior Attachments - Within beltline region	2	2-VT-1	0	0	2	2	
	B13.60	Interior Attachments - Beyond belt line	2	2-VT-3	0	0	2	2	
		Total	4	2-VT-1 2-VT-3	0	0	4	4	
		Percentages for Category			0	0	100	100	See IWB-2412(a).
B-N-3	B13.70	Removable Core Support Structures	1	1-VT-3	0	0	1	1	
		Total	1	1-VT-3	0	0	1	1	
		Percentages			0	0	100	100	See IWB-2412(a).

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		for Category							
B-O	B14.10	Reactor Vessel – Welds in CRD Housings	24	3-PT or UT	0	0	3	3	
		Total	24	3-PT or UT	0	0	3	3	
		Percentages for Category			**0**	**0**	**100**	**100**	See IWB-2412(a).
B-P		All Pressure Retaining Components	NA	NA	NA	NA	NA	NA	See "System Pressure Test Schedule".
B-Q	B16.10	SG Tubing is Straight Tube Design	NA	NA	NA	NA	NA	NA	No Item B16.10 examinations are scheduled.
	B16.20	SG Tubing in U-Tube Design	3	7-ET	2	3	2	7	Technical Specifications and commitments govern these examinations.
		Total	3	7-ET	2	3	2	6	
		Percentages for Category			**28.6**	**42.8**	**28.6**	**100**	
C-A		**Pressure Retaining Welds in Pressure Vessels**							
	C1.10	Shell circumferential welds	11	4-UT	1/3 each of 4 welds	1/3 each of 4 welds	1/3 each of 4 welds	4	
	C1.20	Head circumferential welds	5	2-UT	1/3 each of two welds	1/3 each of two welds	1/3 each of two welds	2	
	C1.30	Tubesheet-to-shell weld	3	1-UT	1/3 of one weld	1/3 of one weld	1/3 of one weld	1	
		Total	19	7-UT	1/3 each of 7 welds	1/3 each of 7 welds	1/3 each of 7 welds	7	
		Percentages for Category			**33.3**	**33.3**	**33.4**	**100**	
C-B		**Pressure Retaining Nozzle Welds in Vessels**							
	C2.11	Nozzles in vessels ≤ 1/2 inch nominal	0	0	0	0	0	0	No Item C2.11 examinations are scheduled.
	C2.21	Nozzle-to-Shell (Nozzle-to-Head or Nozzle-to-Nozzle) weld	9	3-UT/MT	1/3 each of three welds	1/3 each of three welds	1/3 each of three welds	3	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
	C2.22	Nozzle inside radius section	6	2-UT	1/3 each of two nozzle inside radius areas.	1/3 each of two nozzle inside radius areas	1/3 each of two nozzle inside radius areas	2	
	C2.31	Reinforcing plate welds to nozzle and vessel	8	4-PT	1	1	2	4	
	C2.32	Nozzles with reinforcing plate - inside accessible for examination	0	0	0	0	0	0	No Item C2.32 examinations are scheduled.
	C2.33	Nozzle-to-shell (or head) - inside of vessel is inaccessible for examination	4	6 -VT-2	2	2	2	6	
		Total	27	3-UT/MT 2-UT 4-PT 6-VT-2	3 and 1/3 each of five welds	3 and 1/3 each of five welds	4 and 1/3 each of five welds	15	
		Percentages for Category			**31.1**	**31.1**	**37.8**	**100**	
C-C		**Integral Attachments for Class 2 Vessels, Piping, Pumps, and Valves**							
	C3.10	Pressure vessels	4	1-PT or MT	0	1	0	1	
	C3.20	Piping	180	9-PT or MT	3	5	5	13	A sample of 10% of the welded attachments associated with the supports selected for examination under IWF 2510. 90 supports are selected for examination under IWF, Item F1.20.
	C3.30	Pumps	4	1-PT or MT	0	0	1	1	
	C3.40	Valves	0	0	0	0	0	0	No Item C3.40 examinations are scheduled.
		Total	188	11	3	6	6	15	
		Percentages **for Category**			**20**	**40**	**40**	**100**	
C-D		Pressure Retaining Bolting > 2 inch							
	C4.10	Pressure vessels	0	0	0	0	0	0	No Item C4.10 examinations are scheduled.
	C4.20	Piping	0	0	0	0	0	0	No Item C4.20

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
									examinations are scheduled.
	C4.30	Pumps	0	0	0	0	0	0	No Item C4.30 examinations are scheduled.
	C4.40	Valves	0	0	0	0	0	0	No Item C4.40 examinations are scheduled.
		Total	0	0	0	0	0	0	
		Percentages for Category			**0**	**0**	**0**	**0**	
C-F-1		**Pressure Retaining Welds in Austenitic Stainless Steel or High Alloy Piping**	NA	NA	NA	NA	NA	NA	No Item C-F-1 examinations are scheduled. C-F-1 welds are included in the RI-ISI Alternative selection process. See Category "R-A" at the end of this table and Section 7 of the ISI Plan.
		Total	NA	NA	NA	NA	NA	NA	
		Percentages for Category			**`NA**	**NA**	**NA**	**NA**	
C-F-2		**Pressure Retaining Welds in Carbon Or Low Alloy Steel Piping**	NA	NA	NA	NA	NA	NA	No Item C-F-2 examinations are scheduled. C-F-2 welds are included in the RI-ISI Alternative selection process See Category "R-A" at the end of this table and Section 7 of the ISI Plan.
		Total	NA	NA	NA	NA	NA	NA	
		Percentages for Category			**NA**	**NA**	**NA**	**NA**	
C-G		**Pressure Retaining Welds in Pumps and Valves**							
	C6.10	Pump casing welds	36	10-PT/ 20-VT-1 (only if disassembled)	2 & 1/3 welds	3 & 1/3 welds	4 & 1/3 welds	10	Relief Request CMP-002.

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
	C6 20	Valve body welds	0	0	0	0	0	0	No Item C6.20 examinations are scheduled.
		Total	36	10-PT/ 20-VT-1 (only if disassembled)	2 & 1/3 welds	3 & 1/3 welds	4 & 1/3 welds	10	
		Percentages for Category			**23.3**	**33.3**	**43.4**	**100**	
C-H	C7.10	All Pressure Retaining Components	NA	NA	NA	NA	NA	NA	See "System Pressure Test Schedule".
D-A		**Welded Attachments For Vessels, Piping, and Valves.**							
	D1.10	Pressure Vessels	30	10-VT-1	2	5	3	10	
	D1.20	Piping	269	27-VT-1	10	13	18	41	
	D1.30	Pumps	0	VT-1	0	0	0	0	No Item D1.30 examinations are scheduled.
	D1.40	Valves	0	VT-1	0	0	0	0	No Item D1.40 examinations are scheduled.
		Total	299	37-VT-1	12	18	21	51	
		Percentages for Category			**23.5**	**35.3**	**41.2**	**100**	
F-A		**Plate and Shell Type Supports Liner Type Supports Component Standard Supports**							
	F1.10	Class 1 - Pipe supports	207	52-VT-3	18	29	20	67	
		System - SI	41						
		Type - A	6						
		Design - HARES	6		0	6	0	6	
		Type - B	3						
		Design - HAANC	3		1	0	0	1	
		Type - C	32						
		Design - HAHSB	11		1	2	1	4	
		HAVAR	21		2	4	1	7	
		System - CH	51						
		Type - A	35						

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Design - HARES	30		2	4	1	7	
		HASUPC	1		0	0	1	1	
		HASUPT	4		0	0	1	1	
		Type - B	13						
		Design - HAANC	2		0	0	1	1	
		HAGUD2	11		1	0	2	3	
		Type - C	3						
		Design - HAVAR	3		1	0	1	2	
		System - RC	89						
		Type - A	22						
		Design - HARES	16		0	1	3	4	
		HASUPT	6		0	0	2	2	
		Type - B	36						
		Design - HAANC	6		0	0	1	1	
		HAGUD2	30		3	7	1	11	
		Type - C	31						
		Design - HAHSB	8		1	1	0	2	
		HAVAR	23		3	2	2	7	
		System - RH	26						
		Type - A	11		0	0	0	0	
		Design - HARES	11		1	1	1	3	
		Type - B	1						
		Design - HAGUD3	1		1	0	0	1	
		Type - C	14						
		Design - HAHSB	8		1	1	0	2	
		HAVAR	6		0	0	1	1	
		Total for Item F1.10	207	52-VT-3	18	29	20	67	32.3% of Item F1.10 is scheduled for examination.
	F1.20	Class 2 - Pipe supports	501	76-VT-3	20	29	40	89	
		System - SI	274						
		Type - A	101						
		Design - HARES	72		3	4	3	10	
		HASUPC	29		1	1	2	4	
		Type - B	137						
		Design - HAANC	24		3	0	1	4	
		HAGUD2	113		3	6	8	17	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Type - C	36						
		Design - HAHSB	4		0	0	1	1	
		HAVAR	31		2	2	4	8	
		HACON	1		0	0	0	0	
		System - CH	48						
		Type - A	11						
		Design - HARES	11		0	1	1	2	
		Type - B	28						
		Design - HAANC	9		0	1	1	2	
		HAGUD2	19		2	0	1	3	
		Type - C	9						
		Design - HAVAR	7		0	0	1	1	
		HAHSB	2		0	0	1	1	
		System -SHP	42						
		Type - A	3						
		Design - HASUPC	3		0	0	0	0	
		Type - B	3						
		Design - HAANC	3		1	0	0	1	
		Type - C	36						
		Design - HAHSB	28		0	1	3	4	
		HAVAR	8		0	1	1	2	
		System - RH	47						
		Type - A	5						
		Design - HARES	2		0	0	1	1	
		HASUPT	3		0	0	1	1	
		Type - B	2						
		Design - HAANC	2		0	0	0	0	
		Type - C	40						
		Design - HAHSB	14		2	1	0	3	
		HAVAR	26		1	2	1	4	
		System - WFPD	40						
		Type - A	3						
		Design - HASUPC	3		0	0	0	0	
		Type - B	4						
		Design - HAANC	3		1	0	0	1	
		HAGUD2	1		0	0	0	0	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Type - C	33						
		Design - HAHSB	17		0	5	0	5	
		HAVAR	16		0	1	2	3	
		System - CS	27						
		Type - A	15						
		Design - HARES	8		0	0	2	2	
		HASUPC	6		1	0	1	2	
		HASUPT	1		0	0	0	0	
		Type - B	4						
		Design - HAGUD2	4		0	1	0	1	
		Type - C	8						
		Design - HAVAR	8		0	0	1	1	
		System - WAPD	16						
		Type - A	1						
		Design - HARES	1		0	0	0	0	
		Type - B	6						
		Design - HAANC	1		0	0	0	0	
		HAGUD2	5		0	0	1	1	
		Type C	9						
		Design - HAHSB	3		0	0	1	1	
		HAVAR	6		0	2	0	2	
		System - RS	7						
		Type - A	0						
		Type - B	0						
		Type - C	7						
		Design - HAHSB	1		0	0	0	0	
		HAVAR	6		0	0	1	1	
		Total for Item F1.20	501	76-VT-3	20	29	40	89	17.8% of Item F1.20 is scheduled for examination.
	F1.30	Class 3 - Pipe supports	659	66-VT-3	20	21	36	77	
		System - WS	57						
		Type - A	28						
		Design - HARES	20		0	0	2	2	
		HASUPC	1		0	0	0	0	
		HASUPT	7		0	1	0	1	

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Type - B	23						
		Design - HAANC	5		0	0	2	2	
		HAGUD2	18		1	0	1	2	
		Type - C	6						
		Design - HAVAR	6		1	0	0	1	
		System - CC	528						
		Type - A	304						
		Design - HARES	257		7	8	11	26	
		HASUPT	23		0	0	2	2	
		HASUPC	24		0	1	1	2	
		Type - B	141						
		Design - HAANC	41		3	1	0	4	
		HAGUD2	92		2	6	1	9	
		HAGUD3	8		0	0	1	1	
		Type - C	83						
		Design - HAHSB	7		0	1	0	1	
		HAVAR	76		3	0	8	11	
		System - WAPD	47						
		Type - A	17						
		Design - HARES	8		1	0	3	4	
		HASUPC	1		0	0	1	1	
		HASUPT	8		1	0	0	1	
		Type - B	29						
		Design - HAANC	3		0	0	0	0	
		HAGUD2	24		0	2	1	3	
		HAGUD3	2		0	0	0	0	
		Type - C	1						
		Design - HAVAR	1		0	0	0	0	
		System - WCMU	27						
		Type - A	12						
		Design - HARES	4		0	0	1	1	
		HASUPT	8		1	0	0	1	
		Type - B	12						
		Design - HAANC	1		0	0	0	0	
		HAGUD2	11		0	1	1	2	
		Type - C	3						

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Design - HAASB	3		0	0	0	0	
		Total for Item F1.30	659	66-VT-3	20	21	36	77	11.7% of item F1.30 is scheduled for examination.
	F1.40	Class 1, 2 ,3 and MC Equipment supports other than piping	84	41-VT-3	11	9	21	41	Multiple streaming is allowed by Table IWF-2500-1, Category F-A, Note 3.
		Total	1454	235-VT-3	69	88	118	275	
		Percentages for Category			**25.1**	**32.0**	**42.9**	**100**	
R-A		**Risk Informed Piping Examinations**							See Code Case N-577-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method A. Dated March 28, 2000.
R-A	R1.11	Elements Subject to Thermal Fatigue	88	80-UT 45-PT 24-MT 56-VT-2	20 3 16 16	33 30 1 24	27 12 7 16	80 45 24 56	Per relief request R-001, socket welds less than/equal 2-inch NPD will be tested by leakage testing each refueling outage instead of UT.
R-A	R1.12	Elements Subject To High Cycle Mechanical Fatigue	39	7-UT 31-PT 5-MT	3 13 4	2 10 0	2 8 1	7 31 5	Each R1.12 segment receives a VT-2 visual examination each refueling outage. These VT-2 examinations are tracked as part of the system pressure test PT's and are not reported as part of the ISI Schedule.
R-A	R1.13	Elements Subject to Corrosive, Erosive, or Cavitation Wastage	6	6-UT-TK	1	3	2	6	.
R-A	R1.14	Elements Subject to Crevice Corrosion Cracking	0	0	0	0	0	0	There are no item number R1.14 items scheduled.
R-A	R1.15	Elements Subject to Primary Water Stress Corrosion Cracking	0	0	0	0	0	0	There are no item number R1.15 items scheduled.
R-A	R1.16	Elements Subject to Intergranular Stress Corrosion Cracking	13	11-UT 1-UT-TK 5-PT 7-VT-2	2 0 0 2	4 0 3 3	5 1 2 2	11 1 5 7	Per relief request R-001, socket welds less than/equal 2-inch NPD will be tested by leakage testing each refueling outage instead of UT.
R-A	R1.17	Elements Subject to Microbiologically	0	0	0	0	0	0	There are no item number

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
		Influenced Corrosion							R1.17 scheduled.
R-A	R1.18	Elements Subject to Flow Accelerated Corrosion (FAC)	22	NA	NA	NA	NA	NA	The examination method and the frequency of examination is determined by the FAC program.
		Total	168	UT-98 UT-TK-7 PT-81 MT-29 VT-2-63	25 1 16 20 18	39 3 43 1 27	34 3 22 8 18	98 7 81 29 63	
		Percentages for Category			**28.8**	**40.7**	**30.6**	**100**	
		Augmented Program							
AUG	22	Flywheel Inspection	3	3-UT or MT	0	3	0	0	
AUG	25	Sensitized Stainless Steel Piping Welds TS 2.1.1	23	18-UT/PT	6	8	4	18	75% of the population is to be examined each interval. At least 2 welds will be examined each refueling outage.
AUG	26	Sensitized Stainless Steel, TS 2.1.2	77	18-UT/PT or PT	7	8	6	20	22.5% of the population is to be examined each interval. At least 2 welds will be examined each refueling outage.
AUG	27	Sensitized Stainless Steel, TS 2.2.1	71	71-VT-1 36-PT	21 12	25 12	25 12	71 36	
AUG	28	High Energy Lines - Main Steam	26	21-UT/MT	6	8	7	21	One third to be examined each period.
AUG	29	High Energy Lines - Feedwater	11	11 -UT/MT	4	3	4	11	
AUG	30	Loop Stop Valve Stems	6	30-UT CGWT	9	12	9	30	A hot leg valve is examined each refueling outage. A Cold leg valve is examined every other refueling outage.
AUG	31	Reactor Vessel Incore Detector Thimble Tubes	1 set	7-ET	2	3	2	7	The set of thimble tubes is examined each outage.
AUG	32	Special Lifting Devices	13	13-Various	1	4	8	13	
AUG	33	Component Supports	42	26-VT-3	9	9	9	27	Voluntary exams resulting from a QA recommendation.

SURRY POWER STATION UNIT 1, INTERVAL 4, ISI SCHEDULE SUMMARY

CAT	ITEM	DESCRIPTION	TOTAL NO OF ITEMS	REQ'D-EXAM-TYPE	PD 1 SCHED	PD 2 SCHED	PD 3 SCHED	TOTAL SCHED	COMMENTS
AUG	34	Steam Generator Feedwater Nozzles Welds	9	9-UT	18	27	18	63	Examinations for thermal fatigue is required each outage.
AUG	35	Pressurizer Instrument Connection	1	1-VT-2	2	3	2	7	The examination is required each refueling outage.
AUG	36	Reactor Vessel Head CRDM Visual Exam	7	7-VT-2	2	3	2	7	Visual examination of the set of CRDM penetrations is required each refueling outage.
AUG	37	Reactor Coolant Pump - Seal Injection Line Supports	3	21-VT-3	6	9	6	21	The support on each line is to be examine each outage.