VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

August 25, 2003

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

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

VIRGINIA ELECTRIC AND POWER COMPANY
SURRY POWER STATION UNIT 2
FOURTH INTERVAL INSERVICE INSPECTION PROGRAM

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 2 for Class 1, 2, and 3 components and component supports. The Inservice Inspection Plan, included as an attachment to this letter, describes the programmatic aspects of inservice examinations of components and component supports. The System Pressure Test Plan and Risk Informed ISI Program are not included in this submittal. They 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 May 10, 2004, and Surry Unit 2 will begin implementation of the fourth interval inservice inspection program on that date. As allowed by 10 CFR 50.55a(g)(4)(iv), Surry Unit 2 will implement the inservice examination requirements of components and component supports set forth in the 98-2000 Code. Surry Unit 2 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.

The implementation of the 98-2000 Code will be limited for Surry Unit 2 because the plant 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 the attachment. Previously approved relief requests are identified along with the appropriate reference, where applicable.

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The Surry Unit 2 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

Attachment

Commitments made in this letter: None

cc: U.S. Nuclear Regulatory Commission
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Attachment

Fourth Interval Inservice Inspection Program
Surry Power Station Unit 2

Virginia Electric and Power Company
(Dominion)

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION) SURRY POWER STATION, UNIT 2

INSERVICE INSPECTION PLAN FOR COMPONENTS AND COMPONENT SUPPORTS

THE FOURTH INSERVICE INSPECTION INTERVAL MAY 10, 2004 – MAY 09, 2014

REVISION 0
JULY 2003

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ABSTRACT
VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION) SURRY POWER STATION UNIT 2 INSERVICE INSPECTION PLAN FOR COMPONENTS AND COMPONENT SUPPORTS FOURTH INSERVICE INSPECTION INTERVAL MAY 10, 2004 THROUGH MAY 09, 2014

As required by Title 10, Code of Federal Regulations, Section 50.55a (10 CFR 50.55a), paragraph (g)(4), as revised October 28, 2002, the Surry Power Station, Unit 2 (SPS 2), "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 May 10, 2004 and is scheduled to be completed May 09, 2014. In cases where the requirements of the Code have been determined to be impractical, requests for relief have been prepared and are submitted as part of this document 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 also been prepared and submitted as part of this document to the NRC as allowed by 10 CFR 50.55a(a)(3)(i) or (ii).

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," the Risk Informed Inservice Inspection (RI-ISI) Program, 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 2 Containment Inservice Inspection Program is being implemented in accordance with the regulations. 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 2 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, 11548-CB-L&S-4.

The most significant proposed 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". Dominion's basis for use of the risk informed technology is provided in the RI-ISI program alternative, which is being submitted by separate correspondence.

The "Inservice Inspection Schedule for Components and Component Supports" (ISI Schedule) is prepared 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 specific document 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 RI-ISI Program, the ISI Schedule and the SPT Plan, jointly, meet the documentation requirements of the Code, IWA-2420, for the SPS 2, Fourth Inservice Inspection Interval.

SURRY POWER STATION UNIT 2, 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.
CMP-006 – Requests permission to eliminate the volumetric examination of the regenerative heat exchanger welds due to high dose.
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 – 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 on the small piping lines to detect evidence of leakage.
SPT-003 – 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 (none)
Partial Coverage Relief Requests (none)
Code Cases - Relief Requests (none)

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 2 (SPS 2) 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 required by Title 10, Code of Federal Regulations, Part 50, Section 50.55a (10 CFR 50.55a), paragraph (g)(4), the SPS 2 fourth inservice inspection interval ISI Plan was prepared to meet the requirements of 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 2 update to the 98-2000 Code also complies with the supplemental requirements contained within the October 28, 2002 update of 10 CFR 50.55a regarding the implementation of the 98-2000 Section XI Code. The fourth inservice inspection interval begins on May 10, 2004, and will extend through May 09, 2014. The fourth inservice inspection interval is scheduled for a period of 10 years. The dates for the inspection periods are scheduled as follows:

Period 1: May 10, 2004 through May 09, 2007
Period 2: May 10, 2007 through May 09, 2011
Period 3: May 10, 2011 through May 09, 2014

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 2 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 most significant proposed 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". The basis for use of the risk informed technology is provided in the RI-ISI program alternative, which is prepared as a separate document and submitted separately.

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. Components selected for examination by the RI-ISI Plan are also identified, scheduled for examination, and performance tracked as part of the ISI Schedule.

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, RI-ISI Program and the SPT Plan together, provide the required information.

1.3.3 Inspection Program Employed

The ISI Plan for SPS 2 utilizes the interval format of Inspection Program B, as defined in IWA-2432.

1.3.4 Classification and Identification of Components

At the time the construction permit for SPS 2 was issued, 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. Additionally, 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 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 were 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 1.6.

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 11548-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 11548-CB-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 have been proposed, they have been 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
- an R series for risk-informed ISI requirements.

The partial examination requests for relief are necessary because 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.)

Risk-informed ISI relief requests are contained in the RI-ISI Program.

1.3.7 ASME Section XI Code Cases

As allowed by 10 CFR 50.55a and/or USNRC Regulatory Guide 1.147, Revision 13, certain Code Cases have been incorporated into this program. Additionally, certain Code Cases that have been approved by the ASME Code Committee, 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. Implementation of SPS 2 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 has been prepared and submitted in accordance with the requirements of 10 CFR 50.55a. However, this action was taken separately from the ISI Plan.

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 Surry site ISI staff or the corporate 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 commitments are outside of Section XI Code requirements, the Authorized Nuclear Inservice Inspector is not involved with the augmented examinations.

1.5 Repair/Replacement Program

Repair/Replacement activity will be in accordance with Surry Power Station Unit 2 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.

1.6 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, 11548-CB-L&S-4.

Two additional sets of drawings are used to implement the ISI Plan - the 11548-WMKS series and the 11548-SPB/SPM series. The 11548-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 11548-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 2, INTERVAL 4
ISI CLASSIFICATION BOUNDARY DRAWINGS

11548-CB-L&S-4, SH-001	LEGEND AND SYMBOLS
11548-CBB-006A-4, SH-001	AIR COOLING AND PURGING SYSTEM
11548-CBB-047B-4, SH-001	FIRE PROTECTION SYSTEM
11548-CBM-064A-4, SH-001	MAIN STEAM SYSTEM
11548-CBM-064A-4, SH-002	MAIN STEAM SYSTEM
11548-CBM-064A-4, SH-003	MAIN STEAM SYSTEM
11548-CBM-064A-4, SH-004	MAIN STEAM SYSTEM
11548-CBM-064B-4, SH-001	STEAM GEN NITROGEN CONN SYSTEM
11548-CBM-066A-4, SH-002	AUX STEAM AND AIR REMOVAL SYSTEM
11548-CBM-067A-4, SH-002	CONDENSATE SYSTEM
11548-CBM-068A-4, SH-001	FEEDWATER SYSTEM
11548-CBM-068A-4, SH-003	FEEDWATER SYSTEM
11548-CBM-068A-4, SH-004	FEEDWATER SYSTEM
11548-CBM-071A-4, SH-002	CIRCULATING AND SERVICE WATER SYSTEM
11548-CBM-071A-4, SH-003	CIRCULATING AND SERVICE WATER SYSTEM
11548-CBM-071B-4, SH-001	CIRCULATING AND SERVICE WATER SYSTEM
11548-CBM-071B-4, SH-002	CIRCULATING AND SERVICE WATER SYSTEM
11548-CBM-072A-4, SH-001	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-002	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-003	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-004	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-005	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-006	COMPONENT COOLING WATER SYSTEM
11548-CBM-072A-4, SH-007	COMPONENT COOLING WATER SYSTEM
11548-CBM-072B-4, SH-001	COMPONENT COOLING WATER SYSTEM
11548-CBM-072B-4, SH-002	COMPONENT COOLING WATER SYSTEM
11548-CBM-072B-4, SH-003	COMPONENT COOLING WATER SYSTEM
11548-CBM-072C-4, SH-001	COMPONENT COOLING WATER SYSTEM
11548-CBM-072C-4, SH-002	COMPONENT COOLING WATER SYSTEM
11548-CBM-075B-4, SH-002	COMPRESSED AIR SYSTEM
11548-CBM-075C-4, SH-001	COMPRESSED AIR SYSTEM
11548-CBM-075E-4, SH-001	COMPRESSED AIR SYSTEM
11548-CBM-075J-4, SH-001	CONTAINMENT INSTRUMENT AIR SYSTEM
11548-CBM-082A-4, SH-002	SAMPLING SYSTEM
11548-CBM-082A-4, SH-003	SAMPLING SYSTEM
11548-CBM-083A-4, SH-001	VENTS AND DRAINS SYSTEM
11548-CBM-083A-4, SH-002	VENTS AND DRAINS SYSTEM
11548-CBM-083B-4, SH-001	VENTS AND DRAINS SYSTEM
11548-CBM-083B-4, SH-003	VENTS AND DRAINS SYSTEM
11548-CBM-084A-4, SH-001	CONTAINMENT SPRAY SYSTEM
11548-CBM-084A-4, SH-002	CONTAINMENT SPRAY SYSTEM
11548-CBM-084A-4, SH-003	CHEMICAL AND VOLUME CONTROL SYSTEM

11548-CBM-084B-4, SH-001	RECIRCULATION SPRAY SYSTEM
11548-CBM-084B-4, SH-002	RECIRCULATION SPRAY SYSTEM
11548-CBM-085A-4, SH-001	CONT VAC LEAKAGE MONITOR SYSTEM
11548-CBM-085A-4, SH-002	CONT VAC & LEAKAGE MONITOR SYSTEM
11548-CBM-086A-4, SH-001	REACTOR COOLANT SYSTEM
11548-CBM-086A-4, SH-002	REACTOR COOLANT SYSTEM
11548-CBM-086A-4, SH-003	REACTOR COOLANT SYSTEM
11548-CBM-086B-4, SH-001	REACTOR COOLANT SYSTEM
11548-CBM-086B-4, SH-002	REACTOR COOLANT SYSTEM
11548-CBM-086B-4, SH-003	REACTOR COOLANT SYSTEM
11548-CBM-086C-4, SH-001	REACTOR VESSEL LEVEL INSTM SYSTEM
11548-CBM-086C-4, SH-002	REACTOR VESSEL LEVEL INSTM SYSTEM
11548-CBM-087A-4, SH-001	RESIDUAL HEAT REMOVAL SYSTEM
11548-CBM-087A-4, SH-002	RESIDUAL HEAT REMOVAL SYSTEM
11548-CBM-088A-4, SH-001	CHEMICAL AND VOLUME CONTROL SYSTEM
11548-CBM-088A-4, SH-002	CHEMICAL AND VOLUME CONTROL SYSTEM
11548-CBM-088B-4, SH-001	CHEMICAL & VOLUME CONTROL SYSTEM
11548-CBM-088B-4, SH-002	CHEMICAL AND VOLUME CONTROL SYSTEM
11548-CBM-088C-4, SH-001	CHEMICAL AND VOLUME CONTROL SYSTEM
11548-CBM-088C-4, SH-002	CHEMICAL AND VOLUME CONTROL SYSTEM
11548-CBM-089A-4, SH-001	SAFETY INJECTION SYSTEM
11548-CBM-089A-4, SH-002	SAFETY INJECTION SYSTEM
11548-CBM-089A-4, SH-003	SAFETY INJECTION SYSTEM
11548-CBM-089B-4, SH-001	SAFETY INJECTION SYSTEM
11548-CBM-089B-4, SH-002	SAFETY INJECTION SYSTEM
11548-CBM-089B-4, SH-003	SAFETY INJECTION SYSTEM
11548-CBM-089B-4, SH-004	SAFETY INJETION SYSTEM
11548-CBM-118A-4, SH-001	REACTOR CAVITY PURIFICATION SYSTEM
11548-CBM-124A-4, SH-001	STEAM GEN BLOWDOWN, RECIRC, & XFER SYSTEM
11548-CBM-124A-4, SH-002	STEAM GEN BLOWDOWN, RECIRC, & XFER SYSTEM
11548-CBM-124A-4, SH-003	STEAM GEN BLOWDOWN, RECIRC, & XFER SYSTEM
11548-CBM-130A-4, SH-001	RAD MONITOR SYS, CIRC & SERVICE WATER SYSTEM
11548-CBM-130B-4, SH-001	RAD MONITOR CONT PARTICULATE SYSTEM

SECTION 2 REQUESTS FOR RELIEF - COMPONENTS

2.1 Nondestructive Examination

RELIEF REQUEST CMP-001

I. IDENTIFICATION OF COMPONENTS

Nozzle Inner Radius Section (Pressurizer Surge Nozzle)

Weld #	Component #	Drawing #	Class
15NIR	2-RC-E-2	11548-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 2 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.

RELIEF REQUEST CMP-001 (CON'T)

Access to the SPS 2 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 2 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

RELIEF REQUEST CMP-001 (CON'T)

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.

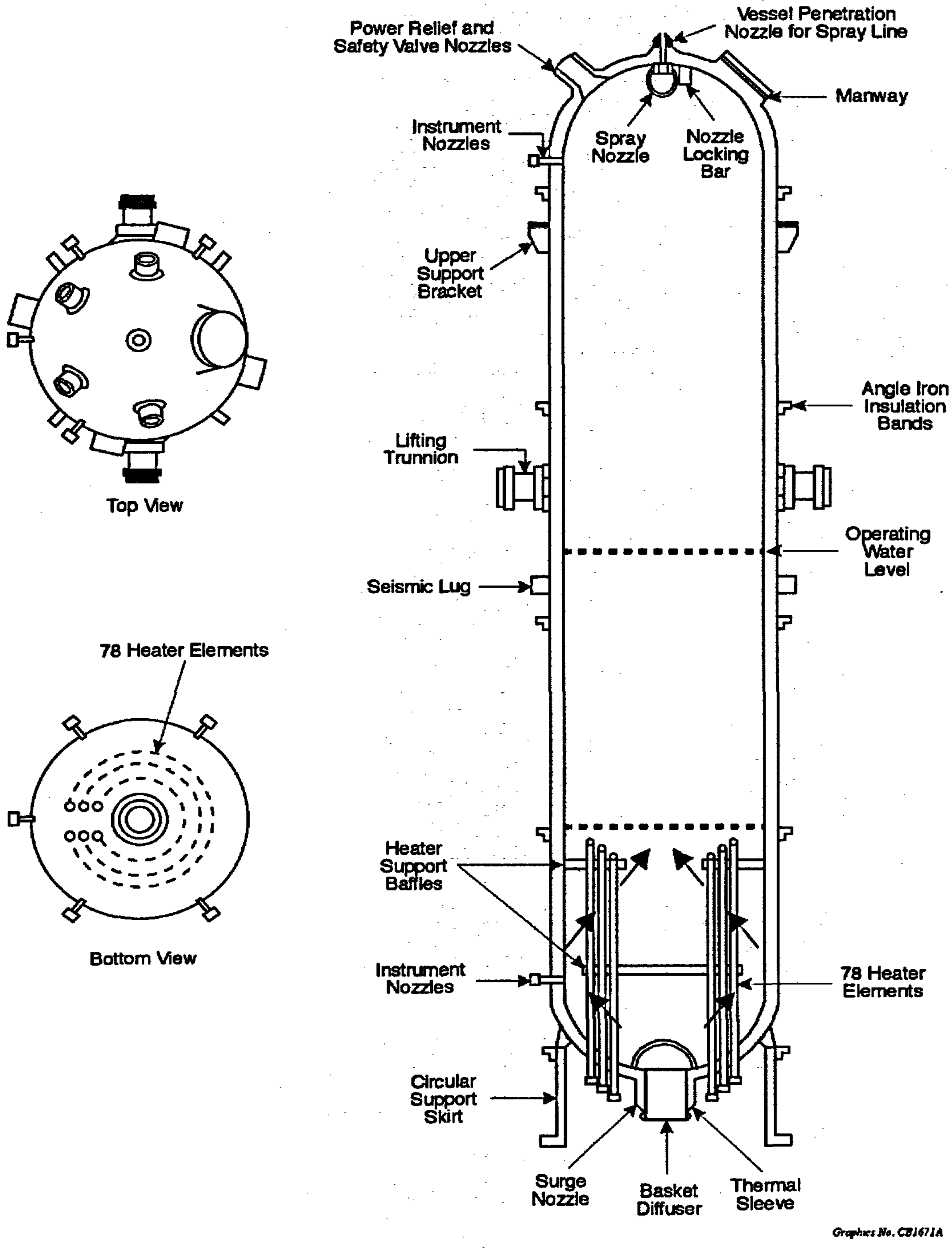
Any ultrasonic examination on this nozzle could only be described as "best effort." The benefit gain would not be commensurate with the difficulty and anticipated exposure estimate of 55.773 man-rem to perform this examination. An alternative examination employing a remote visual technology has very 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 VT-2 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
2-RS-P-2A	2-01	11548-WMKS-RS-P-2A	2
2-RS-P-2A	2-02	11548-WMKS-RS-P-2A	2
2-RS-P-2A	2-03	11548-WMKS-RS-P-2A	2
2-RS-P-2A	2-04	11548-WMKS-RS-P-2A	2
2-RS-P-2A	0-12*	11548-WMKS-RS-P-2A	2
2-RS-P-2B	2-01	11548-WMKS-RS-P-2B	2
2-RS-P-2B	2-02	11548-WMKS-RS-P-2B	2
2-RS-P-2B	2-03	11548-WMKS-RS-P-2B	2
2-RS-P-2B	2-04	11548-WMKS-RS-P-2B	2
2-RS-P-2B	0-12*	11548-WMKS-RS-P-2B	2
2-SI-P-1A	2-01	11548-WMKS-SI-P-1A	2
2-SI-P-1A	2-02	11548-WMKS-SI-P-1A	2
2-SI-P-1A	2-03	11548-WMKS-SI-P-1A	2
2-SI-P-1A	2-04	11548-WMKS-SI-P-1A	2
2-SI-P-1A	0-13*	11548-WMKS-SI-P-1A	2
2-SI-P-1B	2-01	11548-WMKS-SI-P-1B	2
2-SI-P-1B	2-02	11548-WMKS-SI-P-1B	2
2-SI-P-1B	2-03	11548-WMKS-SI-P-1B	2
2-SI-P-1B	2-04	11548-WMKS-SI-P-1B	2
2-SI-P-1B	0-13*	11548-WMKS-SI-P-1B	2

(* Welds 0-12 on 2-RS-P-2A, 2B and 0-13 on 2-SI-P-1A have not been verified to exist. Weld 0-13 on 2-SI-P-1B has been verified to exist above the concrete. However, in the case of four pumps, 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 casing 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 (CON'T)

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 of 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 (CON'T)

(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 constructed prior to the issue and adoption of the current requirements of ASME Section XI. Therefore, the original ultrasonic calibration blocks used for SPS 2 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 2, 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

RELIEF REQUEST CMP-003 (CON'T.)

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. ALTERNATE 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 the 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 08/30/95, TAC No. M89085). 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 2 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

RELIEF REQUEST CMP-004 (CON'T.)

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 welds examined in the third interval 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.

(Note: A similar relief request was approved for North Anna Unit 1 for the 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).

RELIEF REQUEST CMP-005 (CONT.)

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.

(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, 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.)

RELIEF REQUEST CMP-006

I. IDENTIFICATION OF COMPONENTS

Various welds forming the Regenerative Heat Exchanger (2-CH-E-3). The welds are:

Welds	Description	Code Item Number	Class
1-04, 1-17, & 1-19	circumferential head welds	B2.51	1
1-03, 1-18, & 1-22	tubesheet-to-shell welds	B2.80	1
1-06, 1-08, 1-09, 1-11, 1-13, & 1-15	nozzle-to-vessel welds	B3.150	1
NIR-06, NIR-08, NIR-09, NIR-11, NIR-13, & NIR-15	nozzle inside radius areas	B3.160	1
1-01, 1-21, & 1-24	circumferential head welds	C1.20	2
1-02, 1-20, & 1-23	tubesheet-to-shell welds	C1.30	2

II. CODE REQUIREMENTS

Examination Categories B-B, B-D, and C-A require that volumetric examinations be performed of the welds and nozzle inside radius areas listed above.

III. BASIS FOR RELIEF

Background

The regenerative heat exchanger (2-CH-E-3) provides preheat for the normal charging water flowing into the reactor coolant system (RCS). The preheat is derived from normal letdown water coming from the RCS. Charging and letdown constitute the normal chemical and volume control within the RCS. The heat exchanger itself is actually three heat exchangers or sub-vessels in series interconnected with piping. Therefore, examinations are limited to one of the

RELIEF REQUEST CMP-006 (CON'T)

heat exchangers as allowed by the Code for multiple vessels of similar design and function. (Table IWB-2500-1, Category B-B, Note (1) and Table IWC-2500-1, Category C-A, Note (3), Reference: Figure CMP-006.) The lower heat exchanger has historically been chosen for examination to preclude the need for scaffolding and thus minimize personnel dose.

The heat exchanger has an outside shell diameter of 9.25 inches. The shells were manufactured with ASTM A213 TP 304 stainless steel material. The nozzles are 3-inch schedule 160 of similar material. The charging or tube side of the heat exchanger is classified as ASME Class 1. The classification of the letdown (shell) side of the heat exchanger is ASME Class 2. All Class 1 nozzles are required to be examined, and the examinations are not limited to one heat exchanger.

The purpose of this request for relief is to eliminate the Category B-B, B-D, and C-A examination requirements of the regenerative heat exchanger.

Geometric Restrictions

The nozzle-to-vessel welds and nozzle inside radius sections for this vessel were not designed for ultrasonic examination from the outside diameter of the vessel. The small diameter of the vessel and nozzles prevents a meaningful ultrasonic examination of these components. The joint design of the nozzle weld specifies a 3-inch schedule 160 weldolet joined to a 9.25-inch O.D. x 0.875-inch thick vessel. The configuration of the weldolet precluded axial ultrasonic examination from the nozzle side and circumferential examination in either direction. This limits volumetric examination to a single axial scan from the vessel side of the nozzle. It is our position that a meaningful ultrasonic examination cannot be performed on the weld or inner radius with a single axial scan, due to the small diameter of the vessel and weldolet. Further, the change in dihedral around the joint results in a corresponding change in the ultrasonic beam angle, which makes position measurements unreliable. It would also be necessary to extend the beam path to at least two full Vee paths, which would further complicate this examination. These limitations would substantially diminish the ability to discriminate flaw indications from the geometry existing around the joint. The configuration precludes placement of film on the outside diameter for radiography, and the inside surfaces are inaccessible.

Dose Considerations

As part of a third ISI interval request for relief (submitted in 2001) addressing the same examination requirements of the regenerative heat exchanger, an estimate of the potential radiation exposure associated with the examinations was provided. This estimate is provided in Table CMP-006-1 and is presented in

RELIEF REQUEST CMP-006 (CON'T)

support of this request. A personnel dose of 11.968 man-rem is estimated to complete these examinations over the interval. This estimate utilizes dose savings by limiting the circumferential head and tubesheet-to-shell weld examinations to the lower heat exchanger as allowed by the Code. Optimum inspection and preparation times were assumed. However, if difficulties are encountered, a corresponding increase in dose would be expected. Shielding is not considered practical since the source of radiation is the component receiving the examinations.

Conclusion

If the Code required examinations were performed, the geometric restrictions would severely limit the amount of meaningful information that could be obtained concerning the condition of the heat exchanger. Therefore, the significant personnel dose involved with performing the examinations would result in a hardship without a compensating increase in the level of quality and safety. Considering the alternative requirements discussed in Section IV, relief from the Code required examinations on the regenerative heat exchanger is requested pursuant to the provisions of 10 CFR 50.55a(a)(3)(ii).

IV. ALTERNATE REQUIREMENTS

Technical Specifications require that the RCS leak rate be limited to 1 gallon per minute unidentified leakage. This value is calculated periodically in accordance with Technical Specification requirements. Additionally, the containment atmosphere particulate radioactivity is monitored periodically per Technical Specification requirements. As a result, new leakage is rapidly identified and located during operation. Leakage identified from these components can be easily isolated by upstream valves with manual operation from within the control room. The letdown valves also receive an automatic control signal to close on inventory loss based on pressurizer level.

Furthermore, the Class 1 side of the regenerative heat exchanger receives a system leakage test prior to startup up after each refueling outage. During this system leakage test the components receive a visual (VT-2) examination. The Class 2 side of the heat exchanger will continue to receive a periodic pressure test in accordance with IWC-2500, Category C-H and IWC-5000. The supports of the heat exchanger will continue to receive a VT-3 visual examination in accordance with the Code.

RELIEF REQUEST CMP-006 (CON'T)

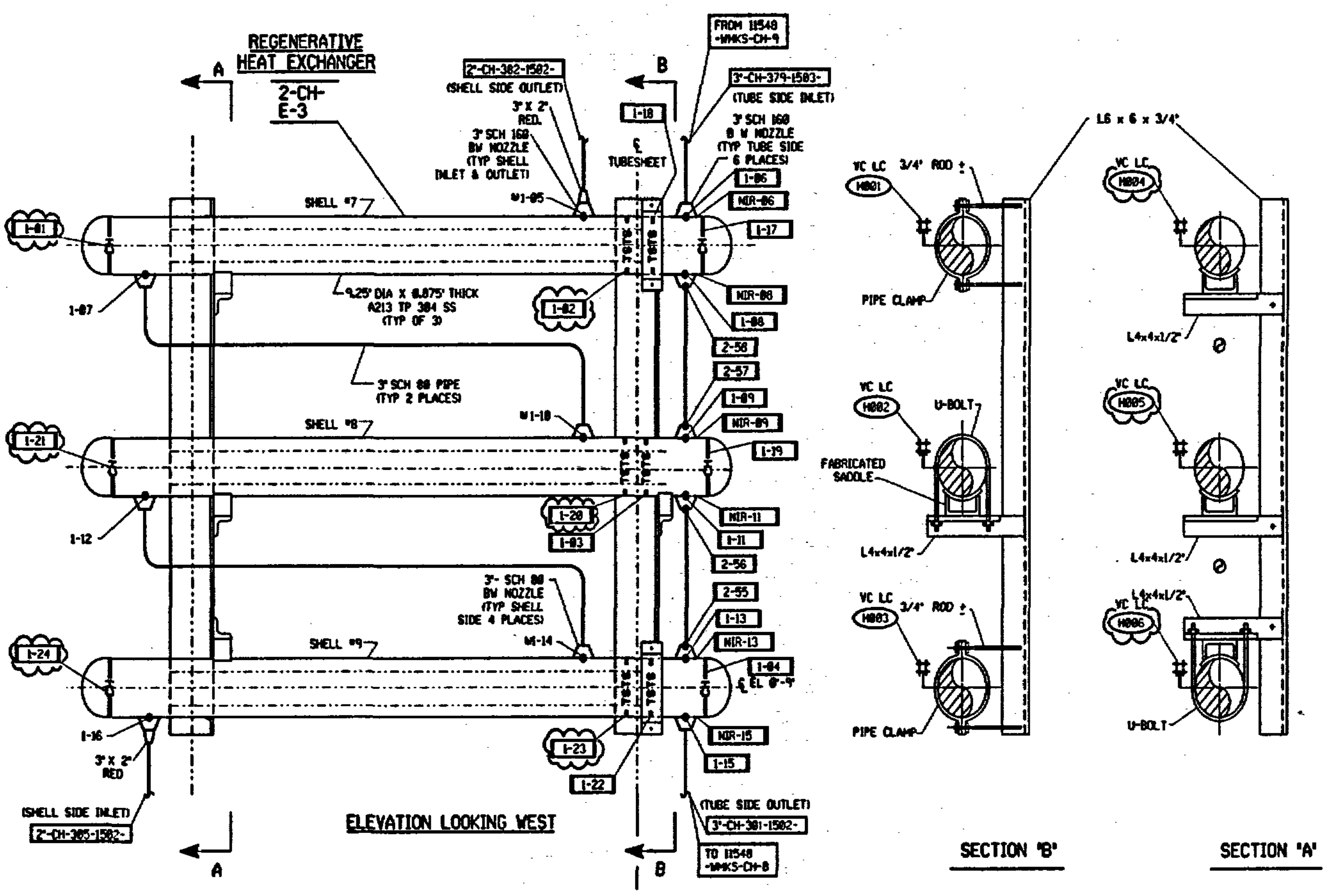
TABLE-CMP-006-1
SURRY UNIT 2 REGENERATIVE HEAT EXCHANGER (2-CH-E-3)
MAN-REM ESTIMATE FOR THE REQUIRED EXAMINATIONS

Work Task	Job Site Man-Hours	Dose Rates Rem/Hr	Estimated Man-Rem All Welds	Est. Man-Rem/Interval 16/24 Welds Required
Remove/install insulation	5.3	0.800	4.240	4.240
Install/remove shielding	0.25	0.800	0.200	0.200
Install/remove scaffolding	2	0.800	1.600	1.600
Remove/install clamp	2	0.500	1.000	1.000
Weld prep	1.25	0.500	0.625	0.446
HP coverage	6.25	0.015	0.094	0.067
Nozzle-to-vessel welds 1-06, 1-08, 1-09, 1-11, 1-13, 1-15	3	0.800	2.400	2.400
Nozzle-to inside radius NIR-06, NIR-08, NIR-09, NIR-11, NIR-13, NIR-15	2.25	0.800	1.800	1.800
Circumferential head welds Class1: 1-04, 1-17, 1-19 Class 2: 1-01, 1-21, 1-24	0.75	0.500	0.375	0.125
Tube-to-tubesheet welds Class1: 1-03, 1-18, 1-22 Class 2: 1-02, 1-20, 1-23	0.54	0.500	0.270	0.090

Total for Required Examinations - 11.968 Man-Rem/Interval

RELIEF REQUEST CMP-006 (CON'T)

Figure CMP-006-1
SURRY UNIT 2 REGENERATIVE HEAT EXCHANGER



(Note: Similar requests for relief were submitted and approved for Joseph M. Farley, under TAC No. MA3449; North Anna Power Station Unit 2, under TAC No. MB07050; Surry Power Station Unit 1, the third inservice inspection interval, under TAC No. MB1998 and Surry Power Station Unit 2, the third inservice inspection interval, under TAC No. MB1999.)

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 startup process. The SPS 2 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.

RELIEF REQUEST SPT-001 (CONT.)

It would be necessary to erect scaffolding to access, within six feet, all surfaces that require examination within the maximum distance requirements of IWA-2212(b). 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."

RELIEF REQUEST SPT-001 (CONT.)

IV. ALTERNATE PROVISIONS

Dominion requests approval in accordance with 10 CFR 50.55a(a)(3)(ii) for SPS 2 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

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.

RELIEF REQUEST SPT-002 (CONT.)

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 near 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 toward 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 2 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 2 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 on 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.

RELIEF REQUEST SPT-002 (CONT.)

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-003

I. IDENTIFICATION OF COMPONENTS

System: Reactor Coolant (RC)

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

II. CODE REQUIREMENTS

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

III. BASIS FOR RELIEF

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

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

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

RELIEF REQUEST SPT-003 (CONT.)

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 subatmospheric conditions of the containment, the Code required examinations at the bottom of the reactor vessel during system leakage tests, results in a hardship without a compensating increase in quality and safety over the proposed alternative. Therefore, approval of this request for relief is requested in accordance with 10 CFR 50.55a(a)(3)(ii).

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

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 relief request 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 2 (SPS 2) 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 2 Technical Specification 4.17. 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

RELIEF REQUEST CS-001 (CONT.)

edition have formed the basis of the Surry Unit 2 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 2 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. Dominion determined that the proposed alternative approach for SPS 2 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; 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 heatup and cooldown of the plant. It would be a hardship to try and impose these requirements on an operating plant such as SPS 2, which is constructed with a subatmospheric containment. As an alternative to the requirements of paragraph 4.3, SPS 2 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

RELIEF REQUEST CS-001 (CONT.)

Section XI examination requirement has been accepted by regulation as providing acceptable quality and safety for supports, and 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. ALTERNATE PROVISIONS

SPS 2 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."

4. As an alternative to paragraph 4.3 of ISTD, for systems that operate at a temperature greater than 200 degrees F, SPS 2 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.

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

6.1 Code Cases Utilized in the Fourth Inservice Inspection Interval

NRC APPROVED CODE CASES UTILIZED IN THE FOURTH INSERVICE INSPECTION INTERVAL

Case N-416-2	"Alternative Pressure Test Requirement for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3, Section XI, Division 1." (Approved by the NRC as stated in Regulatory Guide 1.147, Rev. 13, June 2003, with the following condition. The provisions of IWA-5213, "Test Condition Holding Times," 1989 Edition, are to be used.)
Case N-460	"Alternative Examination Coverage for Class 1 and Class 2 Welds."
Case N-498-4	"Alternative Requirements for 10-Year System Hydrostatic Testing for Class 1, 2, and 3 Systems, Section XI, Division 1." (Approved by the NRC as stated in Regulatory Guide 1.147, Rev. 13, June 2003, with the following condition. The provisions of IWA-5213, "Test Condition Holding Times," 1989 Edition, are to be used.)
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-523-2	"Mechanical Clamping Devices for Class 2 and 3 Piping."
Case N-532-1	"Alternative Requirements to Repair Replacement Documentation Requirements and Inservice Summary Report Preparation and by IWA-4000 and IWA-6000." (Approved by the NRC as stated in Regulatory Guide 1.147, Rev. 13, June 2003, with the following condition. The OAR-1 forms must be submitted to the NRC within 90 days of the completion of the refueling outage.)

Case N-533-1 "Alternative Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure-Retaining Bolted Connections." (Approved by the NRC as stated in Regulatory Guide 1.147, Rev. 13, June 2003, with the following condition. The provisions of IWA-5213, "Test Condition Holding Times", 1989 Edition, are to be followed.)

Case N-566-1 "Corrective Action for Leakage Identified at Bolted Connections."

Case N-597-1 (For FAC Program Application) "Requirements for Evaluation of Pipe Wall Thinning." (Approved by the NRC as stated in Regulatory Guide 1.147 Rev. 13, June 2003, with the following conditions:

1. Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, April 1999, "Recommendations for an Effective Flow Accelerated Corrosion Program," for developing the inspection requirements, the method of predicting the rate of wall thickness. As used in NSAC-202L-R2, the terms "should" and "shall" have the same expectation of being completed.
2. Components affected by the flow-accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction code of record and Owner's requirements or a later NRC approved edition of Section III of the ASME Code prior to the value of t_p reaching the allowable minimum wall thickness, t_{min}, as specified in –3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to NRC review and approval.
3. For Class 1 piping not meeting the criteria of –3221, the use of evaluation methods and criteria is subject to NRC review and approval.
4. For those components that do not require immediate repair or replacement, the rate of wall thickness loss is to be used to determine a suitable inspection frequency so that repair or replacement occurs prior to reaching allowable minimum wall thickness, t_{min}.
5. For corrosion phenomenon other than flow accelerated corrosion, use of the Code Case is subject to NRC review and approval. Inspection plans and wall thinning rates may be difficult to justify for certain degradation mechanisms such as MIC and pitting.)

Case N-623 "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel, Section XI, Division 1."

SECTION 6 CODE CASES

6.2 Requests for Relief

(Reserved for use in later revision.)