May 24, 2002

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

SUBJECT: SURRY POWER STATION UNITS 1 AND 2 RE: ASME SECTION XI, INSERVICE INSPECTION (ISI) PROGRAM RELIEF REQUESTS SR-023 (UNIT 1) AND SR-029 (UNIT 2) (TAC NOS. MB1998 AND MB1999)

Dear Mr. Christian:

This letter grants the relief you requested from the requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section XI for Surry Power Station (Surry), Units 1 and 2. The relief relates to volumetric examination of welds associated with the regenerative heat exchanger.

By letters dated May 16, 2001, and March 25, 2002, you proposed relief from the ASME B&PV Code Section XI. The proposed relief requested approval to use an alternative to the Code-required volumetric examination of welds associated with the Surry Unit 1 and Unit 2 regenerative heat exchangers. The relief proposals were identified as SR-023 (Unit 1) and SR-029 (Unit 2).

Our evaluation and conclusion are contained in the enclosed Safety Evaluation. We concluded that imposition of the Code requirements at Surry for volumetric examination of regenerative heat exchanger welds would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that your proposed alternative, reliance on reactor coolant leak detection systems and the associated Technical Specification allowable leakage limits to assure system integrity, provides reasonable assurance of the structural integrity of the regenerative heat exchangers. Therefore, your proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Surry Power Station, Units 1 and 2, for the third 10-year interval.

The staff has completed its evaluation of this request; therefore, we are closing TAC Nos. MB1998 and MB1999.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosure: As stated

cc w/encl: See next page

The staff has completed its evaluation of this request; therefore, we are closing TAC Nos. MB1998 and MB1999.

Sincerely,

/RA/

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-280 and 50-281

Enclosure: As stated

cc w/encl: See next page

Distribution
PUBLIC
RidsNrrDlpmLpdii
RidsOgcRp
RidsRgn2MailCenter
PDII-1 R/F
G. Edison (paper copy)
G. Hill (4)
RidsAcrsAcnwMailCenter
RidsNrrLAEDunnington (paper copy)
SRosenberg (e-mail)

ADAMS ACCESSION NUMBER: ML021490538

OFFICE	PM:PDII/S1	LA:PDII/S2	OGC	SC/PDII/S1
NAME	GEdison	EDunnington	RHoefling	JNakoski
DATE	05/01/02	04/30/02	05/08/02	05/09/02
COPY	**Yes**/No	**Yes**/No	Yes/No	Yes/**No**

OFFICIAL RECORD COPY

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF SR-023 (UNIT 1) AND SR-029 (UNIT 2)

SURRY POWER STATION, UNITS 1 AND 2

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NUMBERS 50-280 AND 50-281

1.0 INTRODUCTION

Inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel (B&PV) Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection (ISI) of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The Code of record for the Surry Power Station, Units 1 and 2, third 10-year ISI interval is the 1989 Edition of the ASME B&PV Code.

The NRC staff has reviewed the information concerning ISI program Requests for Relief SR-023 (Unit 1) and SR-029 (Unit 2), third 10-year interval for Surry Power Station, Units 1 and 2, provided in a Virginia Electric and Power Company (the licensee) letter dated May 16, 2001, and additional information provided in its letter dated March 25, 2002.

The information provided by the licensee in support of the request for relief from the Code requirements has been evaluated and the basis for disposition is documented below.

Enclosure

2.0 EVALUATION

2.1 Request for Relief SR-023 (Unit 1)

Code Requirement: 1989 ASME Code, Section XI, Table IWC-2500-1, Examination Category C-A requires that volumetric examinations be performed.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the Code-required volumetric examinations for the welds associated with the Unit 1 regenerative heat exchanger (1-CH-E-3) identified below:

Identification of Components:
System: Chemical and Volume Control (CH)
Component: Regenerative Heat Exchanger (1-CH-E-3)
Drawing: 11448-WMKS-CH-E-3[1]

Welds	Description	Code Item#	Class
1-01	circumferential head weld	C1.20	2
1-04	circumferential head weld	C1.20	2
1-06	circumferential head weld	C1.20	2
1-07	circumferential head weld	C1.20	2
1-09	circumferential head weld	C1.20	2
1-10	circumferential head weld	C1.20	2
1-02	tubesheet to shell weld	C1.30	2
1-03	tubesheet to shell weld	C1.30	2
1-05	tubesheet to shell weld	C1.30	2
1-08	tubesheet to shell weld	C1.30	2
1-11	tubesheet to shell weld	C1.30	2

Licensee's Basis for Requesting Relief:

> The regenerative heat exchanger (1-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 heat exchangers as allowed by the Code for multiple vessels of similar design and function. (Table IWC-2500-1, Category C-A, Note (3), Reference: Figure SR-023-1[2].) 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 [type] material. The heat exchanger

---

1. Drawing 11448-WMKS-CH-E-3 is not included in this Safety Evaluation. It is in the licensee's submittal dated March 25, 2002.

2. Figure SR-023-1 is not included in this Safety Evaluation. It is in the licensee's submittal dated March 25, 2002.

is ASME Class 2. The nozzles are 3 inch schedule 160 of similar material and are exempt from examination by IWC-1222(a) for components 4 inch nominal pipe size (NPS) and smaller.

The purpose of this relief request is to eliminate Category C-A weld examinations on the regenerative heat exchanger.

A dose evaluation has been conducted on each activity associated with the examinations of the lower regenerative heat exchanger vessel. The lower vessel was chosen to minimize dose in that scaffolding is not required. Table SR-023-1[3] gives the personnel dose expected from these activities. A personnel dose of 3.072 man-rem is estimated to complete the required examinations over the interval. This estimate assumes optimum inspection and preparation times and should be considered conservatively low. 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.

As previously noted, the inlet and outlet piping for this Class 2 Regenerative Heat Exchanger is 3 inch NPS. Therefore, a crack or defect in the heat exchanger shell could not produce a leak greater than that allowed by the corresponding 3 inch inlet or outlet piping. To produce a leak greater than that produced by a 4 inch line would require multiple shell and/or tube failures in the group of three heat exchangers, which is not considered a credible inservice failure. Therefore, the intent of the ASME Section XI Code 4 inch exemption is maintained.

The radiation exposure expended to perform the discussed examinations would result in a hardship without a compensating increase in the level of quality and safety. We believe the intent of the ASME Section XI Code would be maintained in the Surry ISI program considering the NPS 4 inch and less exemption for Class 2 vessels, piping, pumps and valves. Considering the alternative requirements discussed in Section IV[4], relief from the Code required examinations on the regenerative heat exchanger is requested per the provisions of 10 CFR 50.55a(a)(3)(ii).

Licensee's Proposed Alternative Examination:

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 Specifications 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 that can be operated from the control room. The letdown valves also receive an automatic control signal to close on inventory loss based on pressurizer level.

---

3. Table SR-023-1 is contained in the licensee's submittal dated March 25, 2002, and is not included in this Safety Evaluation.

4. Section IV "Alternative Requirements" is contained in the licensee's submittal dated March 25, 2002, and is stated in this Safety Evaluation under "Licensee's Proposed Alternative Examination."

Furthermore, the heat exchanger will continue to receive a periodic pressure test in accordance with IWC-2500, category C-H, and IWC-5000. The heat exchanger supports will continue to receive VT-3 examinations in accordance with Code Case N-491, Table 2500-1.

Staff Evaluation:

The Code requires 100-percent volumetric and/or surface examination of the subject Class 2 Regenerative Heat Exchanger welds noted in the table above. However, examination of these items is restricted due to high radiological conditions and component geometric configuration. The licensee proposed to eliminate the examinations on the entire regenerative heat exchanger, including the terminal ends, and instead, proposed to rely on reactor coolant system leakage detection systems and the associated Technical Specification allowable leakage limits to assure system integrity.

The configuration of the heat exchanger and the materials from which it is fabricated restrict ultrasonic examination. The inlet and outlet piping to this heat exchanger are exempt from Code volumetric and surface examination requirements, based on size (3-inch NPS). In addition, radiation doses are estimated to be 3.072 man-rem in order to complete the Code-required examinations of the subject components. Therefore, considering the ALARA concerns surrounding the performance of these examinations and the limited access to the subject welds, imposition of the Code requirements would result in hardship on the licensee without a compensating increase in the level of quality and safety. The VT-2 visual examinations for evidence of leakage to be performed during the system leakage test prior to start up after each refueling outage, and the VT-3 visual examination that the integral attachments receive, provide reasonable assurance of structural integrity of the regenerative heat exchanger.

2.2 Request for Relief SR-029

Code Requirement: 1989 ASME Code, Section XI, Table IWB-2500-1 Examination Categories B-B and B-D require that volumetric examinations be performed and Table IWC-2500-1, Examination Category C-A requires that volumetric examinations be performed.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee proposed an alternative to the Code-required volumetric examinations for the welds associated with the Unit 2 regenerative heat exchanger (2-CH-E-3), identified below:

Identification of Components:
System: Chemical and Volume Control (CH)
Component: Regenerative Heat Exchanger (2-CH-E-3)
Drawing: 11548-WMKS-CH-E-3[5]

5. Drawing 11548-WMKS-CH-E-3 is not included in this Safety Evaluation. It is in the licensee's submittal dated March 25, 2002.

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

Licensee's Basis for Requesting Relief:

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 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 SR-029-1[6].) 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 [type] material. The nozzles are 3 inch schedule 160 of similar material. The charging or tube side of the heat exchanger is

6. Figure SR-029-1 is not included in this Safety Evaluation. It is in the licensee's submittal dated March 25, 2002.

classified 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 relief request is to eliminate Category B-B, B-D and C-A examinations on the regenerative heat exchanger.

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 precludes 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 opinion 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 our ability to discriminate flaw indications from the geometry existing around the joint. The configuration also precludes placement of film on the outside diameter for radiography, and the inside surfaces are inaccessible.

A dose evaluation has been conducted on each activity associated with the examinations for the entire regenerative heat exchanger. Table SR-029-1[7] provides the personnel dose expected from these activities. 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 welds 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.

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[8], 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).

---

7. Table SR-029-1 is contained in the licensee's submittal dated March 25, 2002, and is not included in this Safety Evaluation.

8. Section IV "Alternative Requirements" is contained in the licensee's submittal dated March 25, 2002, and is stated in this Safety Evaluation under "Licensee's Proposed Alternative Examination."

Licensee's Proposed Alternative Examination:

> 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 start 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 heat exchanger supports will continue to receive VT-3 examinations in accordance with Code Case N-491, Table 2500-1.

Staff Evaluation:

The Code requires 100-percent volumetric and/or surface examination of the subject Class 1 and 2 Regenerative Heat Exchanger welds noted in the table above. However, examination of these items is restricted due to high radiological conditions and component geometric configuration. The licensee proposed to eliminate the examinations on the entire regenerative heat exchanger, including the terminal ends, and instead, proposed to rely on reactor coolant system leakage detection systems and the associated Technical Specification allowable leakage limits to assure system integrity.

The configuration of the heat exchanger and the materials from which it is fabricated restrict ultrasonic examination. The inlet and outlet piping to this heat exchanger are exempt from Code volumetric and surface examination requirements, based on size (3-inch NPS). In addition, radiation doses are estimated to be 11.968 man-rem in order to complete the Code-required examinations of the subject components. Therefore, considering the ALARA concerns surrounding the performance of these examinations and the limited access to the subject welds, imposition of the Code requirements would result in hardship on the licensee without a compensating increase in the level of quality and safety. The VT-2 visual examinations for evidence of leakage to be performed during the system leakage test prior to start up after each refueling outage, and the VT-3 visual examination that the integral attachments receive, provide reasonable assurance of structural integrity of the regenerative heat exchanger.

3.0 CONCLUSION

The staff concludes that for Requests for Relief SR-023 (Unit 1) and SR-029 (Unit 2), imposition of the Code requirements on the licensee would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternative provides reasonable assurance of structural integrity of the regenerative heat exchanger. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Surry Power Station, Units 1 and 2, for the third 10-year interval.

Principal Contributor: T. McLellan

Date: May 24, 2002

cc:

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

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

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

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

Dr. W. T. Lough
Virginia State Corporation
Commission
Division of Energy Regulation
P. O. Box 1197
Richmond, Virginia 23209

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

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

Mr. Stephen P. Sarver, Director
Nuclear Licensing & Operations Support
Innsbrook Technical Center
Virginia Electric and Power Company
5000 Dominion Blvd.
Glen Allen, Virginia 23060-6711

Mr. David A. Heacock
Site Vice President
North Anna Power Station
Virginia Electric and Power Company
P. O. Box 402
Mineral, Virginia 23117-0402

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