June 12, 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: NORTH ANNA POWER STATION, UNIT 2 - ASME SECTION XI INSERVICE INSPECTION (ISI) PROGRAM THIRD 10-YEAR INTERVAL REQUESTS FOR RELIEF (TAC NO. MB2280)

Dear Mr. Christian:

This letter grants you Relief Requests NDE-001 through 003, NDE-005 through 007, NDE-010, NDE-013, SPT-002, and SPT-005 through 007 that you submitted for North Anna Power Station, Unit 2.

By letter dated June 13, 2001, as supplemented by letter dated January 31, 2002, Virginia Electric and Power Company (VEPCO) submitted requests for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements for the third 10-year ISI interval for North Anna, Unit 2. NDE-012 and NDE-015 were withdrawn as you had requested on your January 31, 2002, submittal. The staff has already approved and issued Relief Requests NDE-009, NDE-014, SPT-001, and CS-001.

Our evaluation of Relief Requests NDE-001 through 003, NDE-005 through 007, NDE-010, NDE-013, SPT-002, and SPT-005 through 007 is enclosed, including the regulatory basis for approval. The approval of NDE-013 is subject to the requirement that VEPCO must assure the implementation of Code Case N-598 will not result in the duration of time between the examination of a pipe, component, or support in the previous inspection interval and the third inspection interval exceeding 10 years. The staff has completed its evaluation of this matter; therefore, we are closing TAC No. MB2280. Relief Requests NDE-004, NDE-008, NDE-011, SPT-003, SPT-004, and SPT-008 are being dispositioned under TAC No. MB2223.

Sincerely,

/RA LOlshan for/

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

Docket No. 50-339

Enclosure: As stated

cc w/encl: See next page

June 12, 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: NORTH ANNA POWER STATION, UNIT 2 RE - ASME SECTION XI INSERVICE INSPECTION (ISI) PROGRAM THIRD 10-YEAR INTERVAL REQUESTS FOR RELIEF (TAC NO. MB2280)

Dear Mr. Christian:

This letter grants you Relief Requests NDE-001 through 003, NDE-005 through 007, NDE-010, NDE-013, SPT-002, and SPT-005 through 007 that you submitted for North Anna Power Station, Unit 2.

By letter dated June 13, 2001, as supplemented by letter dated January 31, 2002, Virginia Electric and Power Company (VEPCO) submitted requests for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI requirements for the third 10-year ISI interval for North Anna, Unit 2. NDE-012 and NDE-015 were withdrawn as you had requested on your January 31, 2002, submittal. The staff has already approved and issued Relief Requests NDE-009, NDE-014, SPT-001, and CS-001.

Our evaluation of Relief Requests NDE-001 through 003, NDE-005 through 007, NDE-010, NDE-013, SPT-002, and SPT-005 through 007 is enclosed, including the regulatory basis for approval. The approval of NDE-013 is subject to the requirement that VEPCO must assure the implementation of Code Case N-598 will not result in the duration of time between the examination of a pipe, component, or support in the previous inspection interval and the third inspection interval exceeding 10 years. The staff has completed its evaluation of this matter; therefore, we are closing TAC No. MB2280. Relief Requests NDE-004, NDE-008, NDE-011, SPT-003, SPT-004, and SPT-008 are being dispositioned under TAC No. MB2223.

Sincerely,
/RA LOlshan for/
John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-339

Enclosure: As stated
cc w/encl: See next page

Distribution

TChan	RidsNrrPMSMonarque	PUBLIC
RidsNrrDlpmLpdii	RidsNrrLAEDunnington	PDII-1 R/F
RidsOgcRp	RidsAcrsAcnwMailCenter	G. Hill (2)
RidsRgn2MailCenter	TMcLellan	

Accession Number: ML021630245

OFFICE	PDII-1/PM	PDII-2/LA	OGC	PDII-1/SC
NAME	SMonarque	EDunnington	SBrock (NLO)	LOlshan for JNakoski
DATE	06/11/02	06/11/02	06/11/02	06/11/02

OFFICIAL RECORD COPY

Mr. David A. Christian
Virginia Electric and Power Company

North Anna Power Station
Units 1 and 2

cc:
Mr. C. Lee Lintecum
County Administrator
Louisa County
P.O. Box 160
Louisa, Virginia 23093

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

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

Old Dominion Electric Cooperative
4201 Dominion Blvd.
Glen Allen, Virginia 23060

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

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

Senior Resident Inspector
North Anna Power Station
U.S. Nuclear Regulatory Commission
1024 Haley Drive
Mineral, Virginia 23117

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

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

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

NDE-001, NDE-002, NDE-003, NDE-005, NDE-006, NDE-007, NDE-010, NDE-013

SPT-002, SPT-005, SPT-006, AND SPT-007

THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

NORTH ANNA POWER STATION, UNIT 2

VIRGINIA ELECTRIC AND POWER COMPANY

DOCKET NO. 50-339

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 10 CFR 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 Director of the Office of Nuclear Reactor Regulation, 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 North Anna Power Station, Unit 2 third 10-year ISI interval is the 1995 Edition through the 1996 Addenda of the ASME Code, Section XI.

2.0 EVALUATION

The staff, with technical assistance from Brookhaven National Laboratory (BNL), has reviewed the information concerning the ISI program Request for Relief Nos. NDE-001, NDE-002, NDE-003, NDE-005, NDE-006, NDE-007, NDE-010, NDE-013, SPT-002, SPT-005, SPT-006,

and SPT-007 for the third 10-year interval for North Anna Power Station, Unit 2, provided in Virginia Electric and Power Company (the licensee) letter dated June 13, 2001.

The licensee provided additional information in its letter dated January 31, 2002. In its letter dated January 31, 2002, the licensee also withdrew Request for Relief Nos. NDE-012 and NDE-015.

Attachment 1 lists each relief request and the status of approval. The staff adopts the evaluations and recommendations for authorizing alternatives and granting relief contained in the Technical Letter Report (TLR), included as Attachment 2, prepared by BNL.

For Request for Relief Nos. NDE-001, NDE-002, NDE-003, and NDE-007, the staff determined that the inaccessibility of the subject welds makes the Code-required examinations impractical to perform. For complete examination coverage of the welds, redesign and modification of the subject components would be an unnecessary burden on the licensee. Furthermore, the licensee's proposed alternatives provide reasonable assurance of structural integrity of the subject components in the licensee's requests for relief.

For Request for Relief No. NDE-005, the licensee's proposed alternative in lieu of the Code-required weld reference system provides an acceptable level of quality and safety.

For Request for Relief Nos. NDE-006, NDE-010, NDE–013, and SPT-006, the licensee's proposed alternatives to use Code Cases N-623, N-532, N-598, and N-533-1, respectively, provide an acceptable level of quality and safety.

For Request for Relief No. SPT-007, the licensee's proposed alternative to perform a VT-2 visual examination while the subject valves remain open provides an acceptable level of quality and safety.

For Request for Relief Nos. SPT-002 and SPT-005, compliance with the Code requirements would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Furthermore, the licensee's proposed alternatives contained in the subject reliefs provide reasonable assurance of structural integrity of the subject components in the licensee's requests for relief.

3.0 CONCLUSION

The North Anna Power Station, Unit 2, Request for Relief Nos. NDE-001, NDE-002, NDE-003, NDE-005, NDE-006, NDE-007, NDE-010, NDE-013, SPT-002, SPT-005, SPT-006, and SPT-007 to the Code requirements have been reviewed by the staff with the assistance of its contractor, BNL. The TLR provides BNL's evaluation of these requests for relief. The staff has reviewed the TLR and adopts the evaluations and recommendations for granting relief.

For Request for Relief Nos. NDE-001, NDE-002, NDE-003, and NDE-007, the staff concludes that the Code requirements are impractical and if imposed would result in burden on the licensee. Furthermore, the licensee's proposed alternatives provide reasonable assurance of structural integrity of the subject components contained in the licensee's requests for relief. Therefore, the licensee's requests for relief are granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval.

For Request for Relief Nos. NDE-005 and SPT-007, the staff concludes that the licensee's proposed alternatives provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year ISI interval.

The staff concludes that for Request for Relief Nos. NDE-006, NDE-010, NDE–013, and SPT-006, the licensee's proposed alternatives to use Code Cases N-623, N-532, N-598, and N-533-1, respectively, provide an acceptable level of quality and safety. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year ISI interval or until such time Code Cases N-623, N-532, N-598, and N-533-1 are referenced in a future revision of Regulatory Guide (RG) 1.147. At that time, if the licensee intends to continue to implement these Code cases, the licensee must follow all provisions in the subject Code cases with the limitations (if any) listed in RG 1.147. The approval of NDE-013 is subject to the requirement that VEPCO must assure the implementation of Code Case N-598 will not result in the duration of time between the examination of a pipe, component, or support in the previous inspection interval and the third inspection interval exceeding 10 years.

For the alternatives contained in Request for Relief Nos. SPT-002 and SPT-005, the staff concludes that the imposition of the Code requirements would result in hardship without a compensating increase in the level of quality and safety and the proposed alternatives provide reasonable assurance of structural integrity of the subject components in the licensee's requests for relief. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval.

Attachments: 1. Summary of Relief Requests
2. Technical Letter Report, Brookhaven National Laboratory

Principal Contributor: T. McLellan

Date: June 12, 2002

TECHNICAL LETTER REPORT

THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

REQUESTS FOR RELIEF

VIRGINIA ELECTRIC AND POWER COMPANY

NORTH ANNA POWER STATION, UNIT 2

DOCKET NUMBER: 50-339

1.0 SCOPE

By letter dated June 13, 2001, Virginia Electric and Power Company (the licensee), submitted multiple requests for relief from the requirements of the ASME Code, Section XI, for the North Anna Power Station (NAPS), Unit 2. The licensee provided additional information in a letter dated January 31, 2002. These relief requests are for the third 10-year inservice inspection (ISI) interval. Brookhaven National Laboratory (BNL) reviewed the information submitted by the licensee and the evaluation of the subject requests for relief are discussed in the following section.

2.0 EVALUATION

The information provided by the licensee in support of the fourteen requests for relief from ASME Code requirements and the licensee responses to the staff's request for additional information (RAIs) in a letter dated January 31, 2002 has been evaluated and the bases for disposition are documented below. The Code of Record for the North Anna Power Station, Unit 2 (NAPS 2), third 10-year ISI interval, which began on December 14, 2001, is the 1995 Edition with Addenda up to and including 1996 Addenda of Section XI of the ASME Boiler and Pressure Vessel Code.

2.1 Request for Relief No. NDE-001, for Pipe Branch Connection Welds of Main Steam Relief Headers, Examination Category C-F-2, Item Number C5.81.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category C-F-2, Item Number C5.81 requires a surface examination of the pipe branch connection circumferential welds in the main steam relief headers during each ISI interval.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing Code-required 100% surface examination of the following pipe branch connection circumferential welds in the main steam relief headers:

Drawing No.	Weld Nos.
12050-WMKS-101A-1	SW-77 to SW-81
12050-WMKS-101A-2	SW-83 to SW-87
12050-WMKS-101A-3	SW-7W to SW-11W

Licensee's Basis for Requesting Relief (as stated):
The design of the main steam relief header branch connection welds requires the use of a reinforcing pad. These pads are fillet welded to both sections of pipe forming the branch connections and completely encase the branch connection welds. Therefore, the branch connection welds cannot receive the required examinations. A surface examination of the reinforcement pad's fillet welds associated with each branch connection weld selected for examination is proposed by this request for relief.

A similar relief request was approved for North Anna Unit 1 for the unit's third interval inspection ISI program by letter dated April 25, 2000, under TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
The following two "Detail Sections" were taken from drawing 12050-WMKS-101A-1, which is referenced in the request for relief. They are typical of the three sets of branch connections included in NDE-001. (Sketch attached to the licensee's RAI response not included in this report)

Examination of the reinforcement pad welds will provide insights into the operational history of the branch connection. If examinations of the welds joining the reinforcement pad to the pipe and the header do not show evidence of deterioration or stress, then an indication is provided that the branch connection weld is still sound. Additionally, the reinforcement pad is part of the design of the joint. For the joint to meet the design, the attachment welds need to be free of unacceptable defects. Therefore, the overall structural integrity of the joint is dependent on both the attachment fillet welds and the branch connection weld. Please note, while Section XI does not require the examination of the reinforcement pad as part of a piping examination, it does require the examination of the fillet welds associated with a reinforcement pad used on a vessel (Table IWC-2500-1, Item C2.31). Additionally, if through-wall cracking does occur in the branch connection concealed by reinforcement pad, the leakage will be able to access the exterior of the piping system by way of the "weep hole" in the reinforcement pad. This leakage will be detected as part of the Code required system leakage test.

Licensee's Proposed Alternative Examination (as stated):
A surface examination will be performed of the pad-to-pipe fillet welds of the reinforcement pad associated with each branch connection weld selected for examination during the interval.

Evaluation:
In accordance with the ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, a surface examination of the pipe branch connection circumferential welds in the main steam relief headers is required during each 10-year ISI interval. As shown in the sketches submitted by the licensee, the subject branch connection welds are completely encased under reinforcement pads, making them inaccessible for the Code-required surface examination.

The licensee proposed to perform a surface examination of the reinforcement pad's fillet welds associated with the branch connection welds during the third ISI interval as an alternative. The overall structural integrity of the joint is dependent on both the attachment fillet welds and the branch connection weld. If through-wall cracking does occur in the branch connection concealed by reinforcement pad, the leakage will be detected as part of the Code-required system leakage test.

The Code-required surface examination of the subject welds is impractical as they are completely encased under reinforcement pads. Since the reinforcement pad fillet welds are supplementing the branch connection welds to safeguard the pressure boundary, continued structural integrity is assured, and the overall level of plant quality and safety will not be compromised. These examinations will provide reasonable assurance that inservice flaws, if developed in the inaccessible fillet welds, will be detected during the Code-required system leakage test. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.2 Request for Relief No. NDE-002, for Outside Recirculation Spray Pump Casing Welds, Examination Category C-G, Item Number C6.10.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category C-G, Item Number C6.10 requires 100% surface examination of the pump casing welds as defined by Figure IWC-2500-8. The examination can be limited to one pump in the case of multiple pumps of similar design, size, function, and service in a system. The examination may be performed from either the inside or outside surface of the component.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% surface examination coverage of the following Outside Recirculation Spray Pump casing welds:

Drawing No.	Weld Nos.
12050-WMKS-RS-P-2A	SW-1, SW-2, SW-3 LS-6, LS-7, LS-8 LS-9 (Partial Access) LS-10 (Partial Access)
12050-WMKS-RS-P-2B	SW-1, SW-2, SW-3 LS-6, LS-7, LS-8 LS-9 (Partial Access) LS-10 (Partial Access)

Licensee's Basis for Requesting Relief (as stated):
A surface examination of all of the pumps casing welds associated with the Outside Recirculation Spray Pumps from either the outside diameter (O.D.) or the inside diameter (I.D.) is not practical. Each of the two outside recirculation spray pump casings has a total of five circumferential welds and five longitudinal welds. Three of the circumferential welds (SW-1, SW-2, and SW-3) and three of the longitudinal welds (LS-6, LS-7 and LS-8) are completely encased in concrete and are not accessible for examination from the O.D. Of the remaining two longitudinal welds, one weld is partially encased in concrete (LS-9) and one weld is partially covered by a vibration plate (LS-10). Complete O.D. examinations cannot be performed on both of these longitudinal welds. The remaining two circumferential welds are completely accessible for examinations from the O.D. Surface examinations from the Inside Diameter (I.D.) are not a practical alternative. Access to the inside of a pump casing is limited by its physical size (24 inch outside diameter for most of its 49 feet 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 weights [sic] approximately 7000

pounds and with the pump driver added, the weight is approximately 10,500 pounds. The pump assembly 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 and is considered to be impractical as it allows access to only a small portion of the overall weld area inaccessible when the pump assembly is in place.

A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program by letter dated April 25, 2000, under TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Access for examination of the casing welds with a boroscope is not possible if the pump is assembled. The only possible route for a boroscope is through two plugged ½-inch openings located on the discharge of the pump assembly. However, because of their location on the pump, distance to the welds of interest, and the obstructions resulting from the installed pump, these openings do not provide an examination opportunity for a boroscope. Even if the pump was partially disassembled to obtain limited access for a boroscope, the examination would still remain ineffective, because of the obstructions created by the installed pump components.

Licensee's Proposed Alternative Examination (as stated):
A surface examination of the accessible portions of the circumferential and longitudinal welds will be performed to the extent and frequency described in IWC-2500. A remote visual examination (VT-1) of the I.D. of the pump casing welds will be performed only if the pump is disassembled for maintenance.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, requires that 100% surface examination of one pump casing welds from either the inside or outside surface is performed during each 10-year ISI interval. As stated by the licensee, 8 of the 10 welds are either completely encased in concrete or partially inaccessible. The inaccessibility of the welds, therefore, makes the Code-required surface examination impractical to perform. For complete examination coverage from the outside surface of the welds, redesign and modification of the plant layout for relocation of these two pumps would be necessary.

The licensee's proposed alternative is to perform the Code-required surface examination of the accessible portions of the outside surface of accessible pump casing welds to the extent and frequency allowed by the Code, and to perform a remote VT-1 visual examination of the interior surface of the pump casing welds when the pumps are disassembled for maintenance and their shafts are removed.

The Code-required 100% surface examination of the outside weld surfaces is impractical. Also, the disassembly and removal of pump internals only to perform an examination of the inside weld surfaces is impractical due to the difficulty of performing such extensive maintenance on a large number of pumps every interval. It is likely that such maintenance would negatively contribute to the overall performance of the pumps. Should the pump be disassembled for maintenance, the use of a remote VT-1 visual examination will provide an acceptable alternative to the Code-required examination, considering the accessibility and configuration of the internal weld surfaces. The licensee's proposed examination is acceptable, given the extent of accessibility for

examination and will provide reasonable assurance that inservice flaws, if developed in the pump casing welds, will be detected and removed or repaired prior to the return of the pumps to service. Also, these pump types have not experienced any history of degradation affecting the pressure boundary integrity. Therefore, it is recommended that request for relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval.

2.3 Request for Relief No. NDE-003, for Low Head Safety Injection Pump Casing Welds, Examination Category C-G, Item Number C6.10.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category C-G, Item Number C6.10 requires 100% surface examination of the pump casing welds as defined by Figure IWC-2500-8. In case of multiple pumps and valves of similar design, size, function, and service in a system, the examination of only one pump and one valve among each group of multiple pumps and valves is required. The examination may be performed from either the inside or outside surface of the component.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required 100% surface examination coverage of the following Low Head Safety Injection Pump casing welds:

Drawing No.	Weld Nos.
12050-WMKS-SI-P-1A	1, 2, 3 LS-1, LS-2, LS-3 LS-4 (Partial Access) LS-5 (Partial Access)
12050-WMKS-SI-P-1B	1, 2, 3 LS-1, LS-2, LS-3 LS-4 (Partial Access) LS-5 (Partial Access)

Licensee's Basis for Requesting Relief (as stated):
A surface examination of all of the pumps casing welds from either the inside or outside surface is not practical. Each of the two low head safety injection pump casings has a total of five circumferential welds and five longitudinal welds. Three of the circumferential welds (1, 2 and 3) and three of the longitudinal welds (LS-1, LS-2 and LS-3) are completely encased in concrete and are not accessible for examination. Of the remaining two longitudinal welds, one weld is partially encased in concrete (LS-4) and one weld is partially covered by a vibration plate (LS-5). Partial Outside Diameter (O.D.) examinations can be performed on both of these longitudinal welds. The remaining two circumferential welds are completely accessible for examinations from the O.D. Surface examinations from the Inside Diameter (I.D.) are not a practical alternative. Access to the inside of a pump casing is limited by its physical size (24 inch outside diameter for most of its approximately 49 feet 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 (several tons) 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 and is considered to be impractical as it allows access to only a small portion of the overall weld area inaccessible when the pump assembly is in place.

A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program by letter dated April 25, 2000, under TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Access for examination of the casing welds with a boroscope is not possible if the pump is fully assembled. (Unlike the Outside Recirculation Spray Pumps, the Low Head Safety Injection Pumps do not have a similar one-half inch opening). Even if the pump was partially disassembled to obtain limited access for a boroscope, the examination would still remain ineffective, because of the obstructions created by the installed pump components.

Licensee's Proposed Alternative Examination (as stated):
A surface examination of the accessible portions of the circumferential and accessible longitudinal welds will be performed to the extent and frequency described in IWC-2500. A remote visual examination (VT-1) of the ID of the pump casing welds will be performed only if the pump is disassembled for maintenance.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, requires that 100% surface examination of one pump casing welds from either the inside or outside surface is performed during each 10-year ISI interval. As stated by the licensee, 8 of the 10 welds are either completely encased in concrete or partially inaccessible. The inaccessibility of the welds, therefore, makes the Code-required surface examination impractical to perform. For complete examination coverage from the outside surface of the welds, redesign and modification of the plant layout for relocation of these two pumps would be necessary.

The licensee's proposed alternative is to perform the Code-required surface examination of the accessible portions of the outside surface of accessible pump casing welds to the extent and frequency allowed by the Code, and to perform a remote VT-1 visual examination of the interior surface of the pump casing welds when the pumps are disassembled for maintenance and their shafts are removed.

The Code-required 100% surface examination of the outside weld surfaces is impractical. Also, the disassembly and removal of pump internals only to perform an examination of the inside weld surfaces is impractical due to the difficulty of performing such extensive maintenance on a large number of pumps every interval. It is likely that such maintenance would negatively contribute to the overall performance of the pumps. Should the pump be disassembled for maintenance, the use of a remote VT-1 visual examination will provide an acceptable alternative to the Code-required examination, considering the accessibility and configuration of the internal weld surfaces. The licensee's proposed examination is acceptable given the extent of accessibility for examination and will provide reasonable assurance that inservice flaws, if developed in the pump casing welds, will be detected and removed or repaired prior to the return of the pumps to service. Also, these pump types have not experienced any history of degradation affecting the pressure boundary integrity. Therefore, it is recommended

that request for relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval.

2.4 Request for Relief No. NDE-005, for Pressure Retaining Welds in the Reactor Vessel and Vessel Nozzle Area Examined by the Automated Vessel Examination Tool.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, IWA-2600, "Weld Reference System," requires a reference system be established for all welds and areas subject to surface or volumetric examination. The system shall permit identification of each weld, location of each weld centerline, and designation of regular intervals along the length of the weld.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from establishing a new reference system that would be totally in compliance with guidelines delineated in IWA-2600 of the ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, for welds in the reactor vessel and vessel nozzle area.

Licensee's Basis for Requesting Relief (as stated):
The original construction activities of the North Anna Power Station did not establish a reference system for the reactor vessel and associated dissimilar metal welds as 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. This alternative referencing system is well established in the industry and provides an acceptable level of quality and safety.

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 operation [sic] 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 10CFR50.55a(3) (i) and (ii).

A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program by letter dated April 25, 2000, under TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Electronic encoding systems have been in use for the reactor vessel examinations by Dominion and the industry for over a decade. Dominion has not had any concern raised in the past over the use of the system from its own staff, the vendor, the ANII, or the regulator. 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 Vessel Examination Tool provides an acceptable level of quality and safety. The alternative system can locate welds with sufficient repeatability for future examinations. Therefore, it will satisfy the objectives of IWA-2600.

Relief is requested under the provisions of 10 CFR50.55a(a)(3)(i).

Licensee's Proposed Alternative Examination (as stated):
The automated vessel examination tool examinations will continue to establish its reference system based upon the existing zero reference and the electronic encoding system designed into tool. No other system is planned or deemed necessary.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, requires that a reference system as delineated in IWA-2600 be established for all welds and areas subject to surface or volumetric examination. The system shall permit identification of each weld, location of each weld centerline, and designation of regular intervals along the length of the weld, such that repeatability of examination can be ensured. However, guidelines of such a reference system did not exist during the early period after construction of the reactor vessel at NAPS-2, and as a result, the licensee did not establish a reference system for the reactor vessel welds as now required by the Code.

The licensee proposed an alternative to continue using the automated vessel tool examinations for establishing its reference system based upon the existing zero reference in the reactor vessel and the electronic encoding system designed into tool, and no other reference system is planned. Such an alternative will locate welds with sufficient repeatability for future examinations and serve the purpose and intent of IWA-2600 requirement, although not totally in conformance with the Code for a weld reference system.

The licensee's proposed alternative will provide reasonable assurance for consistent comparison of future examination results with the previous examination results. Thus, the licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, it is recommended that the request for relief be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third ISI interval.

2.5 Request for Relief No. NDE-006, for Shell-to-Flange Weld of the Reactor Vessel, Examination Category B-A, Item No. B1.30, Code Case N-623.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Note(3) to Table IWB-2500-1, Category B-A, Item Number B1.30 requires volumetric examination of 50% of the Code-required weld length from the flange face by the end of the first period, provided this same portion is examined from the shell face during the third period, and the remainder of the weld length by the end of the third period. IWB-2420(a) requires that the sequence of component examinations established during the first inspection interval shall be repeated to the extent practical during each successive inspection interval.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from performing the Code-required volumetric examination of essentially 100% of the shell-to-flange weld in the reactor vessel in accordance with Note(3) to Table IWB-2500-1 and IWB-2420(a).

Licensee's Basis for Requesting Relief (as stated):
On February 26, 1999 the ASME Boiler and Pressure Vessel Code approved Code Case N-623, "Deferral of Inspections of Shell-to-Flange and Head-to-Flange Welds of a Reactor Vessel," dated February 26, 1999. Although this code case allows deferral of both the shell-to-flange and head-to-flange welds, this request for relief is

applicable only to the shell-to-flange weld. The code case allows deferral of the shell-to-flange weld provided three conditions are met:

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

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

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

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

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

Having satisfied the requirements of Code Case N-623 and having performed an essentially 100% examinations of the shell-to-flange weld in the third period of the second inspection interval, NAPS 2 proposes that the deferral of the examination of 100% of the shell-to-flange weld to the third period of the third inspection interval provides an acceptable level of quality and safety. Approval for the deferral is requested under the provisions of 10CFR50.55a(3)(a) [sic].

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Relief is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

The correct reference is Table IWB-2500-1.

Clarification of relief: The licensee's Request for Relief NDE-006 requests permission to defer the examination required by Note 3 to Table IWB-2500-1 to the third period. Note 3 requires that, as a minimum, a partial examination of 50% of the shell-to-flange weld be performed in the first period. The remainder of the examination of the shell-to-flange weld may be deferred to the third period, if this partial examination is performed. The result of the licensee's request is to perform essentially 100% of the shell-to-flange weld in the third period.

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

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, requires that the RPV shell-to-flange weld be volumetrically examined once each inspection interval. The footnote 3 to Table IWB-2500-1 provides partial deferrals for this weld and requires volumetric examination of 50% of the Code-required weld length from the flange face by the end of the first period, provided this same portion is examined from the shell face during the third period, and the remainder of the weld length by the end of the third period.

The licensee proposes to follow the requirements of Code Case N-623. The licensee meets the requirements listed in Code Case N-623 and that deferral of the weld examinations to the end of the inspection interval is supported by the operating history of the industry. The industry experience to date indicates that examinations performed on the reactor pressure vessels shell-to-flange weld have not identified any detrimental flaws or relevant conditions and that changing the schedule for examining this weld to the end of the licensee's ten-year inservice inspection interval will provide a suitable frequency for verifying the integrity of the subject weld. The subject weld will still receive the same examinations that have been required by the ASME Code Section XI since the reactor was placed in commercial service. The only change is that the RPV shell-to-flange weld examinations will be deferred to the end of the inspection interval without conducting partial examinations from the flange face earlier in the inspection interval. No changes are being made to the volumes of material that are examined, nor to the nondestructive examination (NDE) personnel qualifications. This relief request does not involve changes to NDE methods or acceptance criteria.

In anticipation of proposing this alternative, the licensee has performed essentially 100% examination of the subject weld in the third period of the second inspection interval and there were no identified flaws or relevant conditions. Thus, this deferral will maintain the objective of IWB-2420(a). The licensee's proposed alternative to use Code Case N-623 provides an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third ten-year inservice inspection interval at NAPS-2, or until such time as Code Case N-623 is incorporated into a future revision of Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement Code Case N-623, the licensee must follow all provisions in Code Case N-623, including any exceptions or limitations discussed in the regulatory guide, if any.

2.6 Request for Relief No. NDE-007, for the Pressurizer Surge Line Nozzle-to-Vessel Weld and Nozzle Inside Radius Section, Examination Category B-D, Item Nos. B3.110 and B3.120.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category B-D, Item Nos. B3.110 and B3.120 require 100% volumetric examinations of the pressurizer surge line nozzle-to-vessel weld and nozzle inside radius section as defined by Figure IWB-2500-7.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code-required volumetric examination of pressurizer (2-RC-E-2) nozzle-to-vessel weld 9 and nozzle inner radius section 9NIR. These welds are shown on Figures NDE-007-1 and 2 attached with the relief request.

Licensee's Basis for Requesting Relief (as stated):
Access to the NAPS 2 pressurizer surge line nozzle is obstructed by multi-layered, stainless steel mirror insulation and the cables for the pressurizer heaters. Removal of the insulation and cables would be difficult as well as labor and time intensive. It is also likely that cable or heater pin damage would occur during removal. In addition, it is possible that the impingement shield would have to be removed to gain access to the examination area.

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. The exact scope of work to gain access cannot be fully determined until the unit is shutdown for the next refueling outage. Dose rates are predicted using a step approach to build the total projected exposure. There are four options possible. The worst case option assumes that all 78 heater cables have to be disconnected and pulled back. These cables have brazed connections that will be time consuming to remove and replace following the exam. This option carries a dose estimate of 56 rem. If the outer ring of heaters can be left intact during the examination (disconnect/reconnect 46 heater), then the does estimate is 35.2 rem. If only the first ring of heaters has to be dealt with (20 heaters), then the dose estimate is 18.3 rem. For the highly unlikely scenario of not having to disconnect any heater cables, the dose estimate is 5.3 rem. Separately, if the impingement shield has to be removed, than an additional 5.8 rem must be added to all these totals. It should be noted that the amount of heater cable work expected is likely to have a significant impact on overall outage manning requirements to accommodate the anticipated high dose.

Other personnel safety concerns potentially involved in 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. While actions would be taken to prevent any such events, the large dose rate gradients in the under-pressurizer area would challenge even the protection afforded by the best available technology. Temporary shielding is considered impractical in this regard because placement of the shielding material would obstruct and potentially preclude accessibility to the examination surface. Other issues include actual accessibility after removal of all interferences 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 has performed an evaluation to address the impact of operational transients for NAPS 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) after service equivalent to 60 years of operation for the lower head nozzle weld, is 0.32 for the inside surface and 0.07 for the outside surface. The CUFs for the nozzle inner radius are 0.17 (inside surface) and 0.09 (outside surface). These CUFs are considerably less than the design limit of 1.0 and are lower in magnitude than other locations on the pressurizer that are currently being inspected. For instance, the spray nozzle to safe-end has a CUF of 0.848 and the 6" safety and relief nozzle inside radius welds have a CUF of 0.148.

There are several uncertainties regarding an alternative examination of the inside surface of the pressurizer surge line area. An inspection may be able to be performed in which a boroscope could be fed through the manway and down through the middle of the pressurizer. Adding to the difficulty in performing such an exam, there is a screen device on the outlet of the surge line inner radius to control in-surge to the pressurizer. The boroscope would be positioned through the support plates, and then threaded through a screen inlet orifice, if possible, to the pressurizer surge line area. This examination could be partially obscured by the thermal sleeve. 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 inlet screen 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.

As such, we are also applying for relief per 10 CFR 50.55a(a)(3)(ii) due to the fact that compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. A similar relief for North Anna Power Station Unit 1 was granted for use during the second interval by NRC Letter No. 92-255 dated 4/7/92, and during the third interval by NRC letter No. 00-240 dated 4/25/00. This relief request was also approved for NAPS 2 for the second inspection interval by NRC letter dated 02/12/01. Similar relief was also granted for Surry Power Station Unit 1, Letter No. 95-404, dated 7/19/95, Surry Power Station Unit 2 Letter No. 95-480, dated 8/30/95, Byron Station dated 12/30/98, and Beaver Valley dated 10/8/97.

Licensee's Proposed Alternative Examination (as stated):
The pressurizer surge line nozzle-to-vessel will be examined as part of the normally scheduled Class 1 system leakage test each refueling. In addition the surveillance requirements of Technical Specifications that determine the reactor coolant system leak rate and the containment atmosphere radioactivity will be satisfied. These programs ensure that the overall level of plant quality and safety will not be compromised.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, requires 100% volumetric examinations of the pressurizer surge line nozzle-to-vessel weld and nozzle inside radius section as defined by Figure IWB-2500-7 during each inspection interval. As stated by the licensee, the pressurizer lower head design incorporates penetrations for heaters. The location of these heater penetrations and the lower head design limit the accessibility for performing a 100% volumetric examination of the surge line nozzle-to-vessel weld and the associated inside radius section. The licensee has estimated the percentage of the required volume that could be examined. However, even the limited examination is not commensurate with the personnel exposure that would be received. The inaccessibility of the welds, therefore, makes the volumetric examination impractical to perform to the extent required by the Code. Supporting the impracticality of conducting the complete inspection on the accessible portion are the ALARA considerations.

The licensee proposed an alternative in its request for relief to perform visual (VT-2) examination of the pressurizer surge line nozzle-to-vessel weld during the normally scheduled system leakage test in each refueling outage. The results of the fatigue evaluation show the cumulative usage factors (CUFs) for 60-years of operation are

considerably less than the design limit of 1.0 and therefore, the subject welds are less vulnerable to cracking due to fatigue. In addition, per Technical Specification requirements, reactor coolant leakage is monitored through periodic surveillance using a water inventory calculation and a containment atmosphere particulate radioactivity check. Thus, the proposed alternative to perform a VT-2 visual examination will provide reasonable assurance that unallowable reactor coolant leakage, if it occurred in the surge line welds, would be detected early. Therefore, considering the impracticality of performing the full examinations and the burden from potentially radiation exposure, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i) for the third 10-year ISI interval at NAPS-2.

2.7 Request for Relief No. NDE-010, for Reporting Repaired and Replacement Activities of Class 1, 2, and 3 Components and their Supports, Code Case N-532.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, IWA-4000 and IWA-6000 require the Owner to prepare and submit the Form NIS-1, “Owner’s Report for Inservice Inspection,” and the Form NIS-2, “Owner’s Report for Repair/Replacement Activity.”

Licensee’s Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from reporting the Code-required NIS-1 and NIS-2 forms for repair and replacement activities of Class 1, 2, and 3 components and their supports.

Licensee’s Basis for Requesting Relief (as stated):
NAPS 2 proposes to implement as part of its third interval inspection ISI Plan Code Case N-532, “Alternate Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000" as an alternative to the Section XI Code requirements referenced above. NAPS 2 has reviewed both the requirements of the Section XI Code and Code Case N-532 and has determined the Code Case provides an adequate level of quality and safety as required by 10CFR50.55a(3)(i). This conclusion is based on the following;

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

2) Code Case N-532 does not change the requirement that the documentation associated with a repair/replacement activity required by Section XI and the Owner’s Quality Assurance Program be maintained at the site for review by the NRC and/or representatives of the Owner or personnel from other appropriate organizations.

3) Code Case N-532 does not change the Owner's commitments to maintain an affective [sic] in-process quality assurance program to control applicable aspects of the repair/replacement activity under Appendix B to 10CFR50.

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

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

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

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

A similar relief request was approved for North Anna Unit 1 for the third interval inspection ISI Program and for the NAPS 2 second interval inspection ISI Program by letter dated October 6, 2000.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Relief is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

Licensee's Proposed Alternative Examination (as stated):
As an alternative to the requirements of IWA-4000 and IWA-6000, in accordance with 10CFR50.55a(3), NAPS 2 proposes to use Code Case N-532, with the clarification of the term "corrective action" as discussed in Section III, in the implementation of its Section XI third interval inspection ISI Plan.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, IWA-4000 and IWA-6000 require the Owner to prepare and submit the Form NIS-1, "Owner's Report for Inservice Inspection," and the Form NIS-2, "Owner's Report for Repair/Replacement Activity." The licensee proposed the alternative to use Code Case N-532 with a clarification of the term "corrective measure." The clarification is to differentiate how the Code uses the term "corrective measures." One distinction involves repair and replacement activities, and the second involves maintenance activities that are separate from repair and replacement activities. The licensee proposes to report repair and replacement activities and not report routine maintenance activities, such as tightening threaded fittings to eliminate leakage, torquing of fasteners to eliminate packing leakage, tightening connections, replacing valve packing due to unacceptable packing leakage, tightening loosened mechanical connections on supports, adjustment and realignment of supports, cleanup of corrosion on components from leakage, etc.

The BNL staff reviewed the proposed alternative documentation requirements of Code Case N-532 and determined that although the required forms have changed, the information required by the Code remains available for review. Code Case N-532 requires preparation of the Repair/Replacement Certification Record, Form NIS-2A. The completed Form NIS-2A shall be certified by an Authorized Nuclear Inservice Inspector (ANII) as defined in ASME Code, Section XI, IWA-2130 and shall be maintained by the Owner. Furthermore, the Owner's Activity Report Form, OAR-1 shall be prepared and certified by an ANII upon completion of each refueling outage. The OAR-1 form shall contain an abstract of applicable examinations and tests, a list of item(s) with flaws or relevant conditions that require evaluation to determine acceptability for continued service, and an abstract of repairs, replacements and corrective measures performed as a result of unacceptable flaws or relevant conditions. Hence, the information provided in the documentation required by Code Case N-532 can be used to assess the safety implications of Code activities performed during an outage.

A review using the information as prescribed by the subject code case will, therefore, provide the same or improved level of quality and safety as reviews that may be conducted using the Code reporting requirements. Therefore, it is recommended that the use of this alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third ten-year inspection interval, or until Code Case N-532 is approved for general use by reference in Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement Code Case N-532 the licensee must follow all provisions in the subject code case with the limitations (if any) listed in RG 1.147.

2.8 Request for Relief No. NDE-012, for Additional Examinations Based on the Conditions of Similar Material and Service for Class 1, 2, and 3 Piping and of Similar Type and Function for Class 1, 2, and 3 Supports, Code Case N-586.

In its response dated January 31, 2002 to the staff's RAIs, the licensee stated that based on the position of the NRC expressed in DG-1112, the licensee withdraws Request for Relief NDE-012. No further evaluation is necessary.

2.9 Request for Relief No. NDE-013, for Establishing the Minimum and Maximum Percentage of Examinations to be Completed Each Inspection Period for the Class 1, 2, and 3 Piping, Components and Supports, Code Case N-598.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, and IWF-2410-2 establish the maximum and minimum percent of items for each examination category to be completed each inspection period, except as modified by associated paragraphs of Section XI. Additionally, paragraphs IWB-2420(a), IWC-2420(a), IWD-2420(a), and IWF-2420(a) require "the sequence of component examinations that was established during the first inspection interval shall be repeated during successive inspection interval, to the extent practical."

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee requested relief from establishing Code-required maximum and minimum percent of items for each examination category to be completed each inspection period.

Licensee's Basis for Requesting Relief (as stated):
The reference tables establish both a minimum and maximum percent complete which allows more examinations to be performed in the second and third inspection periods. Only 34% of the examinations can be performed in the first inspection period while as much as 50% can be performed in either the second or third inspection period. However, a minimum of 16% must be performed in any one-inspection period.
Code Case N-598, "Alternative Requirements to Required Percentages of Examination," dated March 2, 1998, provides alternative examination percentages that allow more flexibility in scheduling of examinations to avoid unnecessary expenditure of both manpower and exposure. It changes the Section XI requirements by allowing more examinations to be performed in the first inspection period than the present requirements. The flexibility to perform more examinations in the second or third inspection period is still maintained by the code case. Also, the code case still maintains the 16% minimum in an inspection period. The IWX-2420(a) paragraphs are generally interpreted as requiring the examinations be performed within approximately 10 years of the previous examination. The application of the code case in conjunction with the objectives of the referenced IWX-2420(a) paragraphs would allow the flexibility to bring an examination forward into an earlier inspection period. This action would reestablish the examination schedule of the component, but in a manner that does not impact on the quality or safety afforded to the plant by the examination, because the duration between examinations would be less than 10-years. Therefore, the proposed alternative will provide an acceptable level of quality and safety as required by 10CFR50.55a(3) [sic] and affords the Owners more opportunity in scheduling examinations to minimize radiation exposure.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
Relief is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

Licensee's Proposed Alternative Examination (as stated):
NAPS 2 proposes to be allowed to use the requirements of Code Case N-598, "Alternative Requirements to Required Percentages of Examination," dated March 2, 1998. NAPS 2 will assure that the implementation of the code case will not result in the duration of time between the examination of a pipe, component, or support in the previous inspection interval and the third inspection interval exceeding 10 years, to the extend [sic] practical.

Evaluation:
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Tables IWB-2412-1, IWC-2412-1, IWD-2412-1, and IWF-2410-2 establish the maximum and minimum percent of items for each examination category to be completed each inspection period, except as modified by associated paragraphs of Section XI. In lieu of meeting the requirements of examination percentages for each inservice inspection period, the licensee proposed an alternative to use the percentages of examinations recommended in Code Case N-598 for all ASME Class 1, 2, 3 and MC Components and Supports.

The BNL staff finds that the completion range of examination percentages based on Code Case N-598 allows examinations to be distributed more evenly between refueling outages. The BNL staff also finds that this uniform distribution between outages is more conducive to performing quality examinations. On this basis, BNL concludes that the licensee's proposed alternative to use Code Case N-598 provides an acceptable level of quality and safety. However, the staff takes exception to NAPS 2's statement that it "will assure that implementation of the code case will not result ... in the duration of ... inspection ... exceeding 10years, **to the exten[t] practical**." [Bold added for emphasis]. It is the staff's position that NAPS 2 must assure that the implementation of the code case will not result in the duration of time between the examination of a pipe, component, or support in the previous inspection interval and the third inspection interval exceeding 10 years. On this basis, the BNL staff recommends that the use of this alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inservice inspection interval, or until such time Code Case N-598 is approved for general use by reference in Regulatory Guide 1.147. At that time, if the licensee intends to continue to implement Code Case N-598, the licensee must follow the conditions, if any, specified in the regulatory guide.

2.10 Request for Relief No. NDE-015, for Flaw Sizing Acceptance Criteria of Ultrasonic Examination of Class 1 Clad-to-Base Metal Interface of the Reactor Vessel, Code Case N-622.

In its response dated January 31, 2002 to the staff's RAIs, the licensee stated that based on the position of the NRC expressed in DG-1112, Dominion withdraws Request for Relief NDE-015. No further evaluation is necessary.

2.11 Request for Relief No. SPT-002, for System Leakage Testing of Class 1 Pressure Retaining Piping and Valves, Examination Category B-P, Item Numbers B15.50 and B15.70.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category B-P, Item Numbers B15.50 and B15.70 require system leakage testing and associated VT-2 visual examination of all Class 1 pressure retaining piping and valves in each refueling outage.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing Section XI required system leakage testing and associated VT-2 visual examination on the following components:

Drawing #	Line #	Class
12050-CBM-093A,	1"-RC-644-1502-Q2	1
Sheet 3	1"-RC-645-1502-Q2	1

These lines are located between the reactor head vent isolation valves 2-RC-SOV-201A1 and 201A2, and 2-RC-SOV-201B1 and 201B2. Each line is approximately 1.5 feet in length (Refer to attached Figure SPT-002-1[1]).

Licensee's Basis for Requesting Relief (as stated):
These piping segments are equipped with valves that provide for double isolation of the reactor coolant system (RCS) pressure boundary. The valves are generally maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The non-isolable, upstream portions of the RCS piping (lines 1"-RC-642-1502-Q1 and 1"-RC-643-1502-Q1) will be pressurized using RCS pressure and visually examined as required.

Opening the inboard isolation valves during the inspection of the upstream piping would pressurize lines 1"-RC-644-1502-Q2 and 1"-RC-645-1502-Q2. However, the inboard isolation valves should not be opened while the RCS is pressurized. Opening these valves could release reactor coolant into the reactor vessel refueling cavity. Stroking of the inboard valves has been performed while the RCS was pressurized. This test revealed that when the upstream valve was stroked, the downstream valve tended to lift due to the motive force of the steam. As long as the inboard and outboard valves remain closed under RCS pressure, they are an effective isolation boundary.

However, these valves should not be stroked for reason of routine operation while the RCS is pressurized.

The lines could be pressurized from the end of the discharge piping (1"-RC-646-1502) that leads to the refueling cavity using a pressure test pump. However, the burden of performing this system leakage test is not justified by a corresponding increase in safety.

Only a small portion of the ASME Code classed vent line will be excluded from leakage testing. Each pipe section between the isolation valves is approximately 1.5 feet in length. Also, a stress analysis review was performed on the two pipe sections. The review revealed that these lines are subject to stresses well below the applicable code allowable stresses. The lines have adequate flexibility to accommodate large differential displacement. A review of the support loads showed that these loads are small and within the design loads for the supports.

ASME Section XI Code, paragraph IWA-4540, provides the requirements for hydrostatic pressure testing of piping and components after repairs by welding to the pressure boundary. IWA-4500(b)(5) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Consequently, hydrostatic testing and the associated visual examination of these $\leq$1 inch RCS piping and valve bodies once each 10-year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI Code.

A similar relief request was approved for North Anna Unit 2 for the second interval inspection ISI Program by letter dated March 21, 2001, under TAC NO. MB0361.

1. Fig SPT-002-1 is part of the licensee's letter dated June 13, 2002, and is not included in this report.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated: The request for relief is from the system leakage test requirements of Category B-P, Item numbers B15.50 and B15.70.

The correct reference is to paragraph IWA-4540(b)(5).

The involved pilot-operated solenoid valves have a tendency to open spuriously in response to a sudden increase in supply side pressure at the valve inlet. The problem is most severe when the first of two valves mounted in series opens rapidly, permitting full supply pressure to be sharply introduced to the second valve. In the reactor vent application, if full RCS pressure is suddenly applied to the second valve, it may open suddenly and re-close after a few seconds.

In a closed de-energized valve, inlet pressure (Ps), entering radially, provides an upward force on the piston portion of the main disc. Control pressure (Pc) acting in opposition, negates this lifting force and additionally provides the valve a closure force by its effect on the disc port area. With the pilot valve closed, Pc equal Ps. At the introduction of a inlet pressure surge, supply pressure is momentarily higher than the control pressure, until control pressure re-establishes equality with supply pressure by flow of fluid through the inlet orifice. Consequently, there is a time delay in equalization of these pressures. Should the lifting force exceed the closure forces, the valve will lift. The valve will remain open until the downward force overcomes the lifting force, when upon the valve closes. This technical explanation establishes the cause of the opening of the second valve.

As stated in the request for relief, the opening of the second valve has been documented by testing. The testing and the explanation support the concern of the request for relief that testing of these valves may result in the inadvertent release of RCS coolant into the refueling cavity.

Between each of the two sets of valves and the RCS there is a 3/8-inch orifice that maintains the integrity of the Class 1 boundary i.e., the leakage through the 3/8-inch opening is within the makeup capacity of one charging pump. The SOV valves were added to maintain positive control of RCS inventory. The piping and valves downstream of the orifices were design [sic] as ASME Class 2. The ISI program upgraded these components to Class 1 as permitted by IWA 1320(c).

The stroking of these valves while the RCS is pressurized could result in an unnecessary release of reactor coolant into the reactor vessel refueling cavity.

<u>Licensee's Proposed Alternative Examination</u> (as stated):
As an alternative to Section XI required system leakage test of the subject Class 1 reactor vessel vent piping the following is proposed:

1. The reactor vessel vent piping 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 XI Class 1 System Leakage Test (IWB-5220).

2. The reactor vessel vent piping will also be visually examined with the isolation valves in the normally closed position during the 10-year ISI system pressure test. This examination will be performed with the RCS at nominal operating pressure and at near operating temperature after satisfying the required 4-hour hold time.

3. The surveillance requirements of Technical Specifications that determine the reactor coolant system leak rate and the containment atmosphere radioactivity will be satisfied.

These proposed alternatives will ensure that the overall level of plant quality and safety will not be compromised.

Evaluation:
The Code requires that all Class 1 components within the RCS boundary undergo a system leakage test once each refueling outage. The licensee has proposed an alternative to the Code-required system leakage test requirements for the subject line segments within the RCS pressure boundary.

The subject line segments are 1 inch lines and the line configuration provides double isolation of the RCS as indicated in attached Figure SPT-002-1. Under normal plant operation, the subject line segments would see RCS temperature and pressure only if leakage through the first normally closed valve occurs. To perform the Code-required test, it would be necessary to open the first valve at system normal operating pressure, thereby eliminating the double isolation of the RCS boundary. Pressurization by this method would cause significant safety concerns for the personnel performing the examination due to the close proximity to the primary RCS piping. The test could be done by pressurizing from the end of the discharge piping (1"-RC-646-1502) that leads to the refueling cavity using a pressure test pump. However, the burden of performing the test this way is not justified by a corresponding increase in safety because the possibilities of having a leakage failure in both valves or in the inboard valve plus the line section are small. Therefore, performing the Code-required testing on these isolated line segments results in a hardship for the licensee.

The subject isolation valves are closed during normal operation, and the outboard valve is not pressurized. Performing the Code-required test would identify any leakage in the 1.5-foot section of the pipe or in the outboard valves. However, a stress analysis performed shows that these lines are stressed well below the applicable code allowable stresses. The possibilities of having a leakage failure in both valves or in the inboard valve plus the line section are small. Thus, performing the Code-required test would not provide a significant increase in safety.

The proposed alternative by the licensee will examine the isolation valves in the normally closed positions for leaks and evidence of past leakage during the system leakage test each refueling outage. In addition, the RCS head vent connections will be visually examined with the isolation valves in the normally closed position during the 10-year pressure test. Thus, the licensee's proposed alternative will provide reasonable assurance that leakage integrity of the subject line segments is maintained. Accordingly, imposition of the Code requirement on the licensee would result in a hardship without a compensating increase in the level of quality and safety. Therefore, it is recommended that pursuant to 10 CFR 50.55a(a)(3)(ii) the staff authorizes the proposed alternative for the third 10-year ISI interval.

2.12 Request for Relief No. SPT-005, for System Leakage Testing of Partial Penetration Welds at the Bottom of the Reactor Vessel, Examination Category B-P, Item Number B15.10.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category B-P, Item Number B15.10 requires system

leakage test of IWB-5220 and associated VT-2 visual examination of the bottom of the reactor vessel in each refueling outage.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing VT-2 visual examination of the partial penetration welds at the bottom of the reactor vessel in the reactor coolant system during the system leakage test of IWB-5220.

Licensee's Basis for Requesting Relief (as stated):
In order to meet the Section XI pressure and temperature requirements for the system leakage tests of the reactor vessel, the reactor containment at NAPS 2 is required to be at a sub-atmospheric pressure. Station administrative procedures require that self-contained breathing apparatus 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 that the examiner to [sic] descend several levels by ladder and navigate a small entrance leading to the reactor vessel. In addition to these physical constraints, the examiner must contend with extreme environmental conditions: elevated temperatures due to reactor coolant at temperature above 500 degrees F and limited air circulation in the vessel cubicle. Also, the examiner is limited to the approximate 30-minute capacity of the breathing apparatus for containment entry, the VT-2 examination, and containment exit.

A similar relief request was approved for North Anna Unit 1 for that unit's third interval inspection ISI Program by letter dated April 25, 2000, under TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
The hardship arises less from the time constraint created by the use of bottled air or the involved radiation levels, but rather more from conditions that exist during Mode 3 of reactor operation. During Mode 3 the reactor coolant system is at the operational temperature of $\geq 350^{o}$F, and the containment is sub-atmospheric. Performing the examination at Mode 3 is complicated by the following factors:

1. The need to use a self-contained breathing apparatus (SCBA) with a full-face respirator. The weight of the bottle is approximately 25 pounds.

2. Having to access the bottom of the vessel under sub-atmospheric conditions which requires the examiner to descend several levels by ladders and to navigate a small, 2'-7.25" by 2'-0" hatch way wearing the SCBA.

3. The physical environment caused by the heat generated by a vessel elevated to a temperature of $\geq 350^{o}$F coupled with a lack of ventilation.

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

During operations, the Technical Specifications (TS) require the monitoring of reactor coolant leak rate. No identified pressure boundary leak can exist during operation and leakage from unidentified sources cannot exceed 1.0 GPM. Also, radiation monitors (gas and particulate) would respond to an increase in detectable leakage. These TS

requirements provide for ongoing monitoring for leakage during the operating cycle and for decisive correction [sic] action if an issue develops. As for direct visual examination, any leakage is expected to leave boron crystal residue that can be identified by a VT-2 visual examination performed during the refueling outage. The frequency of the proposed visual examination is the same as the system leakage test required by the Code.

The monitoring methods of the station and the VT-2 visual examination of the area each refueling outage provide an acceptable level of quality and safety. Because of the burden and potential safety challenges caused by the sub-atmospheric conditions of the containment, the Code required examinations at the bottom of the reactor vessel during system leakage tests, results in a hardship without a compensating increase in quality and safety over the proposed alternative.

Current Technical Specifications establish the following requirements and limits for leakage during modes of operation 1 through 4:

1. Every 72 hours, during steady state operation, the reactor coolant system leak rate is monitored to assure the limit of one gallon per minute unidentified leakage is maintained.

2. Every 12 hours the containment atmosphere particulate radioactivity is monitored.

3. No pressure boundary leakage is allowed and only one (1) gallon per minute of unidentified leakage is allowed.

Dominion has submitted a request to the NRC to change its current Technical Specifications to the plant specific Improved Technical Specifications (ITS). In this revision of the specifications, it is no longer required that the containment atmosphere particulate radioactivity be monitored every 12 hours. The proposed ITS requires that "One containment atmosphere radioactivity monitor (gaseous or particulate) be operable. The monitoring of the reactor coolant system leak rate every 72 hours during steady state operation remains a requirement of ITS, as do the limits on leakage.

Licensee's Proposed Alternative Examination (as stated):
Technical Specifications have surveillance requirements that monitor leakage and radiation levels. The applicable Technical Specification requirements will be satisfied through the third inspection interval. The incore sump room has a level alarm in the control room requiring operator action.

These actions would identify any integrity concerns associated with this area.

A VT-2 visual examination will be conducted when containment is at atmospheric conditions each refueling outage for evidence of boric acid corrosion."

Evaluation:
In accordance with the ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, a system leakage test and associated VT-2 visual examination of the bottom of the reactor vessel are required during each refueling outage. The BNL staff has reviewed the information concerning the ISI Program Request for Relief SPT-005 for the third 10-year ISI interval of NAPS 2 pertaining to visual VT-2 examination of the partial penetration welds at the bottom of the reactor vessel.

The containment building is at subatmospheric conditions during the system leakage test. Therefore, the examiner must wear a self-contained breathing apparatus with a full face respirator that limits his work duration and mobility. The examiner also has to descend several levels by ladders and to navigate a small, 2'-7.25" by 2'-0", hatch way wearing the breathing apparatus. In addition to these physical constraints, the examiner must contend with high ambient temperatures. Thus, the imposition of the examination requirements would cause a considerable burden on the licensee.

The licensee proposed, as an alternative, to perform a VT-2 examination for evidence of boric acid corrosion when the containment is at atmospheric conditions during refueling outage. The VT-2 examination for evidence of boric acid corrosion conducted during each refueling outage provides reasonable assurance of leaktight integrity of the subject welds. In addition, the licensee noted that the Technical Specifications require the monitoring of reactor coolant leak rate, containment atmospheric particulate radioactivity, and containment sump level. Thus, the Code-required VT-2 visual examinations at the bottom of the reactor vessel during system leakage test would result in a hardship without a compensating increase in the level of quality and safety. Therefore, it is recommended that the proposed alternative examination be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

2.13 Request for Relief No. SPT-006, for Removal of Insulation of the Class 1, 2, and 3 Bolted Connections in Systems Borated for the Purpose of Controlling Reactivity, Code Case N-533-1.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, paragraph IWA-5242(a) requires in part that for systems borated for the purpose of controlling reactivity, insulation shall be removed from the pressure retaining bolted connections for VT-2 visual examination.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from removing the insulation for the Code-required VT-2 visual examination of the pressure retaining bolted connections for the systems borated for the purpose of controlling reactivity.

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

The use of this code case is considered to be necessary by North Anna Power Station, because the majority of involved bolted connections are located inside containment, and potentially in high radiation areas. Additionally, many of the connections are required to be assessed by the Class 1 system leakage test conducted at Mode 3 as part of the start up testing following a refueling. NAPS 2 is required by Technical Specifications to be at sub-atmospheric conditions at this time. Because of the sub-atmospheric

condition any personnel inside containment must wear self-contained breathing apparatus. This would make the tear down and removal of scaffolding a task of significant hardship, one that provides essentially no gain in safety if the proposed alternative is utilized. (The scaffolding is required to gain access to the bolted connections and to reinstall the insulation after completion of the test.) Secondly, most of the involved connections are tested at elevated temperatures. The actions necessary to protect the personnel who must re-install insulation on these connections and remove scaffolding in the proximity of these connections represent another hardship with little or no justification from gains in safety.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
The reference sentence was meant to convey that if the testing is performed as part of the Code specified system leakage test, the test would be conducted during operation at nominal operating pressure and temperature. For many of the pipe sections or components, the operational temperature, and, therefore, the required test temperature, would be elevated (i.e., well above ambient and in a range that could be harmful to humans if accidental contact with the piping or component was to occur). The sentence does not describe an alternative test of the bolted connections. Dominion proposes no alternative testing other than that specified by Code Case N-533-1.

Request to use Code Case N-533-1 is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

<u>Licensee's Proposed Alternative Examination</u> (as stated):
NAPS 2 requests approval, as allowed by 10CFR50.55a(3) [sic], to use Code Case N-533-1, "Alternative Requirements for VT-2 Visual Examination of Class 1, 2, and 3 Insulated Pressure Retaining Bolted Connections," dated February 26, 1999, as an alternative to the requirements of IWA-5242(a) related to bolted connections in system [sic] borated for the purpose [of] controlling reactivity. In addition to complying with the requirements of paragraph[s] (a) and (b) of Code Case N-533-1, North Anna Power Station will also assure that a 4 hour hold time is required as part of the system pressure test required by paragraph (a) before the VT-2 visual examination is performed.

<u>Evaluation:</u>
ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, paragraph IWA-5242(a) requires removal of all insulation from pressure-retaining bolted connections in systems borated for the purpose of controlling reactivity when performing VT-2 visual examination during system pressure tests. In lieu of this, the licensee requested to use Code Case N-533-1.

The licensee stated in its relief request that the majority of involved bolted connections are located inside containment, and potentially in high radiation areas. Many of these connections are required to be assessed by the system leakage test conducted at Mode 3 as part of the start up testing following a refueling. At this time, by Technical Specifications, the containment remains at sub-atmospheric conditions and therefore, any personnel inside the containment is required to be equipped with self-contained breathing apparatus. This would make the insulation tear down and removal of scaffolding tasks of significant hardship without any gain in the safety. Also, most of the involved connections are tested at elevated temperatures. Thus, requiring the licensee to remove insulation during the pressure test would create a safety hazard due to these extreme conditions and would also result in excessive radiation exposure to plant personnel.

Code Case N-533-1 allows a system pressure test and associated VT-2 examination without the removal of insulation. It also provides an alternative to the requirements of IWA-5242(a) that allows for the removal of insulation and examination of the bolted connections without the connection being pressurized. Thus, the method proposed by the Code Case is well suited to the examination of borated system in that boron that leaks out of the system under pressurized condition leaves easily detectable residue or presence of boron crystal, which is later detected by a VT-2 visual examination. The approach specified in the Code Case provides reasonable assurance of the continued leakage integrity of Class 1, 2, and 3 bolted connections in borated systems.

Requiring the licensee to remove insulation at normal operating pressure (and elevated temperature) would present a significant safety hazard for plant personnel. The licensee's proposed use of Code Case N-533-1, in conjunction with its commitment to assure that a 4 hour hold time is required as part of the system pressure test before the VT-2 visual examination is performed, provides an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third inspection interval, or until Code Case N-533-1 is approved for general use by reference in Regulatory Guide 1.147. After that time, if the licensee intends to continue to implement Code Case N-533-1, the licensee must follow the conditions, if any, specified in the regulatory guide.

2.14 Request for Relief No. SPT-007, for System Leakage Testing of Class 1 Pressure Retaining Piping and Valves, Examination Category B-P, Item Numbers B15.50 and B15.70.

Code Requirement: ASME Section XI, 1995 Edition with addenda up to and including the 1996 Addenda, Examination Category B-P, Item Numbers B15.50 and B15.70 require system leakage testing and associated VT-2 visual examination of all Class 1 pressure retaining piping and valves.

Licensee's Code Relief Request: Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing Section XI required system leakage testing and associated VT-2 visual examination of approximately 30, small diameter ($\leq$ 1 inch), Class 1, reactor coolant system (RCS) pressure boundary vent and drain, sample, and instrumentation connections.

Licensee's Basis for Requesting Relief (as stated):
These piping segments are equipped with valves that provide for double isolation of the reactor coolant system (RCS) pressure boundary. These valves are maintained closed during normal operation and the piping outboard of the first isolation valve is, therefore, not normally pressurized. The proposed alternative provides an acceptable level of safety and quality based on the following:

1. ASME Section XI Code, 1995 Edition with addenda up to and including the 1996 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)(5) excludes component connections, piping, and associated valves that are 1 inch nominal pipe size and smaller from the hydrostatic test. Visual examination of these $\leq$1 inch diameter RCS vent/drain/sampling connections once each 10-year interval is unwarranted considering that a repair weld on the same connections is exempted by the ASME XI Code.

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

3. These piping connections are typically socket-welded and the welds received a surface examination after installation. The piping and valves are normally heavy wall (schedule 160 pipe and 1500# valve bodies). The vents, drains, and sample lines are not subject to high stresses or cyclic loads, and the design ratings are significantly greater than RCS operating or design pressure.

The Technical Specifications (TS) require RCS leakage monitoring during normal operation. Should any of the TS limits be exceeded, then appropriate corrective actions, which may include shutting the plant down, are 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 removing a flange and connecting a test rig. A majority of these piping segments are located in close proximity to the RCS main loop piping thus requiring personnel entry into high radiation areas within the containment. The dose associated with this testing was 1.5 man-Rem.

By a letter dated September 3, 1998 the NRC approved a similar relief request for Edwin I. Hatch Plant, Units 1 and 2. Also, on April 25, 2000, the NRC approved a similar relief request for North Anna Power Station, Unit 1, TAC NO. MA5750.

In its response dated January 31, 2002 to the staff's RAI, the licensee stated:
The request for relief is from the system leakage test requirements of Category B-P, Item Numbers B15.50 and B15.70.

Relief is requested under the provisions of 10 CFR 50.55a(a)(3)(i).

<u>Licensee's Proposed Alternative Examination</u> (as stated):
As an alternative to Section XI reguired [sic] system leakage test of the subject Class 1 RCS pressure boundary connections the following is proposed:

1. 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 XI Class 1 System Leakage Test (IWB-5221).

2. During modes 1 through 4 the RCS will be monitored for leakage and radiation levels in accordance with the requirements of the applicable Technical Specifications.

3. These alternative provisions will only be applied to the inservice testing performed to meet the requirements of Category B-P. They will not be applied to testing performed to satisfy requirements for post-repair/replacement activities related to these components.

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 above alternative examination requirements to those of Section XI stated above in Section I is requested under the provisions of 10CFR50.55a(3) [sic].

Evaluation:
The Code requires that all Class 1 small diameter (≤ 1 inch) components (including vent and drain, sample, and instrumentation connections) within the RCS boundary undergo a system leakage test once each refueling outage. The licensee has proposed an alternative to the system leakage test requirements for the subject RCS components. The proposed alternative is to conduct a visual examination for evidence of leakage each refueling outage during the RCS leakage test with the isolation valves normally in the closed position, and to monitor for leakage and radiation levels inside the containment in accordance with the Technical Specifications.

The RCS Class 1 vent and drain, sample, and instrumentation connection lines are 1 inch or smaller lines. The piping connections are typically socket-welded and the welds received a surface examination after installation. Testing with the isolation valves in their normally closed position means that the portions of those small diameter piping and connections located between double isolation valves, which are part of the Class 1 pressure boundary, will not have the Code-required pressurization during the pressure tests. The following justifications provide an acceptable basis for not including these unpressurized segments of the piping system in the Code-required pressure tests:

- The normally unpressurized piping segments are generally not subject to a harsh corrosive environment.

- With likely less severe pressure and thermal loadings, and generally ample design margins for the small diameter piping and connections, through-wall cracking due to flaw growth is unlikely. Fatigue loading due to vibration is unlikely to lead to failure given the age of these units.

- With routine monitoring of coolant leakage rate and containment air particulate radioactivity required by the plant Technical Specifications, any occurrence of leakage will likely be discovered in a timely manner and followed by appropriate corrective actions.

- As reported by the licensee, these components are located inside containment and in close proximity to the reactor coolant loop piping where radiation levels are high. Therefore, imposition of Code requirements will expose plant personnel to high doses of radiation.

When the system leakage test is performed at operating pressure and temperature, the portion of the piping beyond the first isolation valve up to the second valve is normally at a much lower pressure than RCS pressure. Opening the first isolation valve to extend the test boundary to the second valve would result in single valve protection of the reactor coolant boundary and may result in inadvertently pressurizing a lower pressure system to RCS pressure if the second valve allows sufficient leakage. By maintaining the test boundary at the first isolation valve, any seat leakage past this valve would pressurize the space between the isolation valves for which relief being sought but to a somewhat lower pressure than the RCS pressure. Thus, this will provide reasonable assurance of the leaktightness of the pressure boundary.

The licensee's proposed alternative to examine the isolation valves in the normally closed position for leaks and evidence of past leakage during the system leakage test each refueling outage will provide reasonable assurance that leakage integrity of the subject lines is maintained. Thus, the licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, it is recommended that pursuant to 10 CFR 50.55a(a)(3)(i) the staff authorizes the proposed alternative for the third 10-year ISI interval.

NORTH ANNA POWER STATION, UNIT 2
Third 10-Year ISI Interval

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	TLR Sec.	System or Component	Exam Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Status
NDE-001	2.1	Main Steam Relief Header	C-F-2	C5.81	Circumferencial Pipe Branch Connection Welds	Surface Examination	Surface Examination of reinforcement pads	Granted per 10 CFR 50.55a (g)(6)(i)
NDE-002	2.2	Outside Recirculation Spray Pumps	C-G	C6.10	Pump Casing Welds	Surface Examination	Remote visual, VT-1 when disassembled	Granted per 10 CFR 50.55a (g)(6)(i)
NDE-003	2.2	Low head Safety Injection Pumps	C-G	C6.10	Pump Casing Welds	Surface Examination	Remote visual, VT-1 when disassembled	Granted per 10 CFR 50.55a (g)(6)(i)
NDE-005	2.4	Reactor Vessel	IWA-2600	N/A	Vessel and Nozzle Welds	Weld Reference System	Automated Vessel Examination Tool	Authorized per 10 CFR 50.55a(a)(3)(i)
NDE-006	2.5	Reactor Vessel	B-A	B1.30	Shell-to-Flange Weld	Volumetric Examination	Code Case N-623	Authorized per 10 CFR 50.55a(a)(3)(i)
NDE-007	2.6	Pressurizer	B-D	B3.110 and B3.120	Surge Nozzle-to-Vessel Weld and Nozzle Radius Section	Volumetric Examination	System Leakage Test and Radiation Monitoring	Granted per 10 CFR 50.55a(g)(6)(i)
NDE-010	2.7	Class 1, 2, and 3 Components and Supports	IWA-4000 and IWA-6000	N/A	Repair and Replacement Activities	Preparation of Form NIS-1 and NIS-2: Report for Repair/ Replacement Activity	Code Case N-532	Authorized per 10 CFR 50.55a(a)(3)(i)
NDE-012	2.8	Class 1, 2, and 3 Components and Supports	IWX-2430(a)	N/A	Additional Examination	Additional Examinations	Code Case N-586	Relief Request Withdrawn
NDE-013	2.9	Class 1, 2, and 3 Components and Supports	IWX-2412-1	N/A	Percentage Inspection in each Period	Maximum and Minimum Percent of Items to be Examined	Code Case N-598	Authorized per 10 CFR 50.55a(a)(3)(i)
NDE-015	2.10	Class 1 Clad-to-Base Metal of RV	App. VIII, Suppl. 4, Para 3.2(c)	N/A	Clad-to-Base Metal Interface	Testing to Demonstrate Sizing Ability	Code Case N-622	Relief Request Withdrawn
SPT-002	2.11	Reactor head vent lines	B-P	B15.50 and B15.70	System Leakage	Valves open during system pressure tests	Visual examination while valves remain closed	Authorized per 10 CFR 50.55a(a)(3)(ii)
SPT-005	2.12	Reactor Vessel	B-P	B15.10	System Leakage	VT-2 visual examination during system leakage test	VT-2 visual examination for boric acid corrosion	Authorized per 10 CFR 50.55a(a)(3)(ii)
SPT-006	2.13	Class 1, 2, and 3 Bolted Connections	IWA-5242(a)	N/A	Bolted Connections	VT-2 visual examination during system leakage test	Code Case N-533-1	Authorized per 10 CFR 50.55a(a)(3)(i)
SPT-007	2.14	Class 1 small diameter lines	B-P	B15.50 and B15.70	System leakage	VT-2 visual examination during system leakage test	Visual examination while valves remain closed	Authorized per 10 CFR 50.55a(a)(3)(i)