March 21, 2003

Mr. J. S. Keenan
Vice President
Brunswick Steam Electric Plant
Carolina Power & Light Company
Post Office Box 10429
Southport, North Carolina 28461

SUBJECT: BRUNSWICK STEAM ELECTRIC PLANT, UNIT NO. 2 - REQUEST FOR RELIEF FROM THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN REQUESTS FOR RELIEF RR-29 AND RR-30 FOR BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2 (TAC NOS. MB5854 AND MB5855)

Dear Mr. Keenan:

In a letter dated July 16, 2002, as supplemented December 18, 2002, Carolina Power & Light Company (CP&L, the licensee) requested relief from American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2.

The NRC staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory, has reviewed and evaluated the information provided by CP&L. The licensee requested that the NRC staff approve Requests for Relief RR-29 and RR-30 for BSEP, Units 1 and 2. In response to an NRC request for additional information, the licensee provided additional information in its letter dated December 18, 2002.

The NRC staff's Safety Evaluation is enclosed. Pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative described in Relief Request RR-29, Revision 1, is authorized on the basis that the alternative testing provides a reasonable assurance of the continued structural integrity of the nozzle inner radius sections listed in the proposal. The proposed alternative is authorized for the third 10-year inservice testing interval, which ends on May 10, 2008.

Further, for Request for Relief RR-30, Revision 1, the proposed alternative described in the licensee's submittal provides an acceptable level of quality and safety. Therefore, RR-30, Revision 1, is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inspection intervals at BSEP, Units 1 and 2.

If you have any questions regarding this issue, please contact Ms. Brenda Mozafari at 301-415-2020 or by e-mail at blm@nrc.gov.

Sincerely,

/***RA***/

Allen G. Howe, Chief, Section 2
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-325 and 50-324

Enclosure: Safety Evaluation

If you have any questions regarding this issue, please contact Ms. Brenda Mozafari at 301-415-2020 or by e-mail at blm@nrc.gov.

Sincerely,

/***RA***/

Allen G. Howe, Chief, Section 2
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Docket Nos. 50-325 and 50-324

Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOR

THIRD 10-YEAR INTERVAL INSERVICE INSPECTION

REQUESTS FOR RELIEF RR-29 AND RR-30

FOR

BRUNSWICK ELECTRIC PLANT, UNITS 1 AND 2

CAROLINA POWER & LIGHT COMPANY

DOCKET NOS. 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated July 16, 2002, as supplemented by letter dated December 18, 2002, Carolina Power & Light Company (CP&L, the licensee) requested relief from American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, for the Brunswick Steam Electric Plant (BSEP), Units 1 and 2. The NRC staff, with technical assistance from Pacific Northwest National Laboratory (PNNL), has reviewed the information concerning inservice inspection (ISI) program Requests for Relief RR-29 and RR-30 submitted for the third 10-year interval for the BSEP, Units 1 and 2.

2.0 REGULATORY REQUIREMENTS

ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME B&PV Code and applicable addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if: (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 applicable Code of record for the third 10-year ISI for BSEP, Units 1 and 2, is the 1989 Edition of the ASME B&PV Code, Section XI.

3.0 TECHNICAL EVALUATION

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

For Request for Relief RR-29, the Code requires 100-percent volumetric examination for the inner radius sections of reactor pressure vessel (RPV) nozzles. The licensee proposed to perform enhanced VT-1 (EVT-1) visual examination on the accessible portions of the nozzle inner radius sections in lieu of ultrasonic testing (UT) on the Core Spray (N5A and N5B), Reactor Recirculation (N2A through N2K), and Jet Pump Instrumentation (N8A and N8B) Systems. There are a total of 14 nozzles included in this proposal. These will be remote examinations conducted from the vessel interior. “Enhanced” in this case refers to the ½-mil wire standard that is to be demonstrated by the examiners to assure acceptable sensitivity. The licensee has stated that flaw acceptance will be based upon ASME Table IWB-3512-1. The licensee also noted that visual examination of the inner radius region of the specified nozzles is limited because the reactor internal piping configuration prevents placement of the camera in all positions necessary to examine the subject nozzles and stated that estimated coverages will be in the range of 40-60 percent for the inner surfaces of the subject nozzle inner radius sections.

While the proposed visual examination on these components will be limited to about 40-60 percent estimated coverage, the NRC staff determined that flaws of significant size will be detected. The NRC staff also recognizes that the licensee, and other industry boiling-water reactors (BWRs), have experienced no reported cracking in the subject nozzle inner radius regions, and that these specific nozzles are not subjected to substantive thermal cycling. In addition, it is noted that approximately 50 percent of the total RPV nozzle population will receive the Code-required volumetric examinations during the current interval at BSEP, Units 1 and 2, and that where historical fatigue cracking has occurred, e.g., feedwater nozzles, the licensee will continue to perform volumetric examinations with procedures developed and qualified in accordance with the Code.

Furthermore, performance of the volumetric examinations results in significant personnel radiation exposure. Dose rates in the area for the specified nozzles, with shielding in place, are in the range of approximately 180 to 200 mr/hr. Performance of these examinations results in an estimated personnel exposure of approximately 3 rem per inspection interval.

The personnel radiation exposure that would be incurred if the licensee was required to perform the Code volumetric examinations would be a significant hardship without a compensating increase quality and safety. The potential of the EVT-1 visual examinations to detect thermal fatigue degradation in the inspected regions of the nozzle inner radius sections, the absence of any historical industry evidence for cracking in the subject components, and the volumetric examinations being performed on other similar nozzles, provides reasonable assurance of the continued structural integrity of the subject nozzles.

For Request for Relief RR-30, the Code requires 100-percent volumetric examination for the inner radius sections of RPV nozzles. In BWRs such as BSEP, Units 1 and 2, ultrasonic examinations are performed on nozzle inner radius sections from the outside surface of the vessel, which is accessed through penetrations, or “windows,” in the biological shield wall. For the subject components, the licensee proposed to perform an enhanced VT-1 (EVT-1) visual examination on “essentially 100%” of the nozzle inner radius section surfaces in lieu of UT on the Control Rod Drive Return Line (N9), Reactor Recirculation (N1A and N1B), and Reactor Pressure Vessel Instrumentation (N11A, N11B, N16A and N16B) Systems. There are a total of seven nozzles included in this proposal. These will be remote examinations conducted from the vessel interior. “Enhanced” in this case refers to the ½-mil wire standard that is to be demonstrated by the examiners to assure acceptable sensitivity. The licensee has stated that flaw acceptance will be based upon ASME Table IWB-3512-1. The NRC staff also recognizes that the licensee, and other industry BWRs, have experienced no reported cracking in the subject nozzle inner radius regions, and that these specific nozzles are not subjected to substantive thermal cycling. In addition, it is noted that approximately 50 percent of the total RPV nozzle population will receive the Code-required volumetric examinations during the current interval at BSEP, Units 1 and 2, and that where historical fatigue cracking has occurred, e.g., feedwater nozzles, the licensee will continue to perform volumetric examinations with procedures developed and qualified in accordance with the Code. Therefore, based on above, the NRC staff has determined that the licensee’s proposed alternative to perform EVT-1 visual examinations on the subject components provides an acceptable level of quality and safety.

4.0 CONCLUSION

The NRC staff adopts the evaluations and recommendations for authorizing alternatives contained in the TLR, included as Attachment 1, prepared by PNNL. Attachment 2 lists each relief request and the status of approval.

For Request for Relief RR-29, the NRC staff concludes that the Code requirements to perform the Code volumetric examinations would be a significant hardship without a compensating increase in quality and safety, and the proposed alternative provides reasonable assurance of the continued structural integrity of the subject nozzles. Therefore, the licensee's proposed alternative to use EVT-1 visual examination in lieu of volumetric for the subject nozzle inner radius sections is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for the third 10-year ISI interval.

For Request for Relief 30, the NRC staff concludes that the licensee’s proposed alternative provides reasonable assurance of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative to use EVT-1 visual examination in lieu of volumetric for the subject nozzle inner radius sections is authorized for the third 10-year ISI interval.

Attachments:
1. Technical Letter Report
2. Table

Principal Contributor: T. McLellan

Date:

TECHNICAL LETTER REPORT
ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION
REQUESTS FOR RELIEF NOS. RR-29 AND RR-30
FOR
CAROLINA POWER AND LIGHT COMPANY
BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 2
DOCKET NUMBERS: 50-325 AND 50-324

1.0 INTRODUCTION

By letter dated July 16, 2002, the licensee, Carolina Power and Light Company (CP&L), submitted Requests for Relief RR-29 and RR-30, proposing alternatives to certain requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*. In response to an NRC Request for Additional Information (RAI), the licensee revised these requests and provided further information in a letter dated December 18, 2002. The requests are for the third 10-year inservice inspection (ISI) interval at Brunswick Steam Electric Plant, Units 1 and 2 (Brunswick 1-2). The Pacific Northwest National Laboratory (PNNL) has evaluated the subject requests for relief below.

2.0 REGULATORY REQUIREMENTS

Inservice inspection of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME *Boiler and Pressure Vessel Code* (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). The regulation at 10 CFR 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the U.S. Nuclear Regulatory Commission (NRC), if the licensee demonstrates that (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) shall meet the requirements, except the design and access provisions and the preservice 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, which was 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 Brunswick 1-2 third 10-year interval inservice inspection programs, which began on May 11, 1998, is the 1989 Edition of Section XI of the ASME Boiler and Pressure Vessel Code, with no addenda.

3.0 TECHNICAL EVALUATION

The information provided by CP&L in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below.

3.1 Request for Relief RR-29, Revision 1, Examination Category B-D, Item B3.100, Full Penetration Welded Nozzles in Vessels, Reactor Pressure Vessel Nozzle Inside Radius Sections

Code Requirement: Examination Category B-D, Item B3.100, requires 100% volumetric examination, as defined by Figures IWB-2500-7(a) through (d), as applicable, of Class 1 reactor pressure vessel (RPV) nozzle inside radius sections.

Licensee's Proposed Alternative to Code: Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee is proposing to use a visual examination of the accessible inside surface in lieu of the Code-required volumetric examination for the inner radius sections of RPV nozzles on the Core Spray (N5A and N5B), Reactor Recirculation (N2A through N2K), and Jet Pump Instrumentation (N8A and N8B) Systems. There are a total of fourteen nozzles included in this proposal. The licensee stated:

CP&L will perform a remote visual examination of the accessible surface M-N, as shown in Figures IWB-2500-7(a) through (d), in lieu of the volumetric examinations required by Table IWB-2500-1, Examination Category B-D, Item B3.100, for the RPV nozzles identified in this relief request. The remote visual examination will be performed using the Enhanced VT-1 (i.e., EVT-1) requirements described in the Electric Power Research Institute (EPRI) technical report entitled *TR-105696-R4 (BWRVIP-03) Revision 4: BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines*. The sensitivity for remote in-vessel examinations will be established using the ½ mil wire standard described in that report.

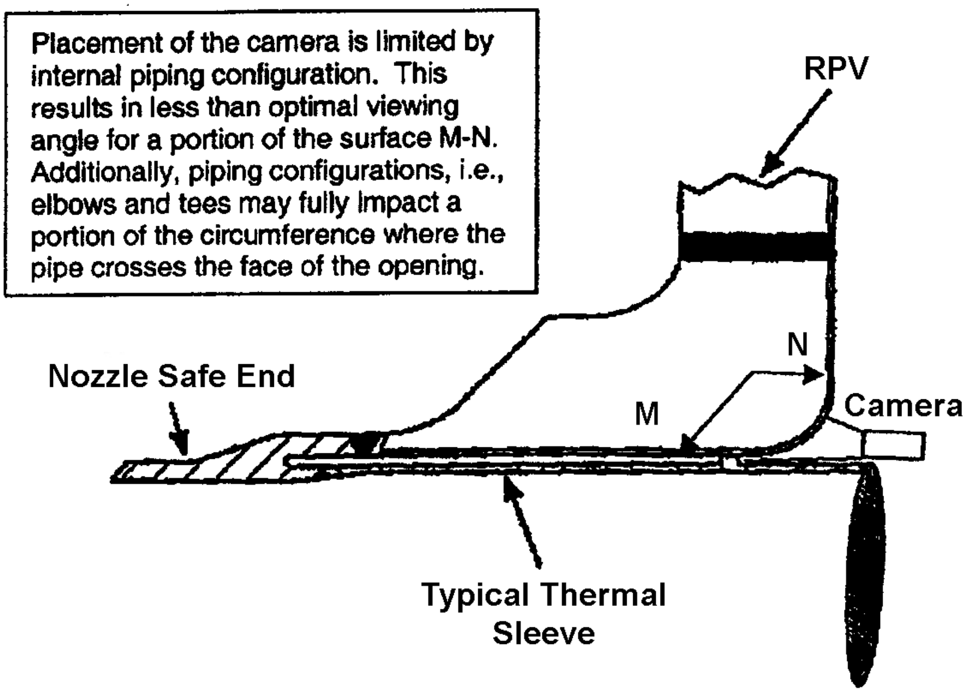
Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 will be deemed unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3, or IWB-3142.4. This alternative is similar to the inspection alternative proposed in ASME Section XI Code Case N-648.

Licensee's Bases for Alternative (as stated):

The volumetric examinations required by the ASME Code, Section XI will result in hardship without a compensating increase in the level of quality and safety, and the proposed visual examination alternative provides reasonable assurance of structural integrity of the subject components for the following reasons:

Performance of the volumetric examinations results in significant personnel radiation exposure without a commensurate increase in the level of plant quality or safety. Volumetric examination requires the examiner to enter and remain inside the biological shield penetration area around the nozzle for the duration of the ultrasonic examination, which takes approximately one hour. Dose rates in the area for the specified nozzles, with shielding in place, are in the range of approximately 180 to 200 mr/hr. Performance of these examinations results in an estimated personnel exposure of approximately 3 rem per inspection interval. Visual examination will reduce personnel radiation exposure consistent with the plant ALARA Program.

Visual examination of the inner radius region of the specified nozzles is limited because the reactor internal piping configuration prevents placement of the camera in all positions necessary to examine surface M-N over the full circumference (i.e., see Figure 1 below).



- Typical Cross-Section of BWR Nozzle with Internal Piping

The specific nozzle limitations and estimated coverage are as follows:

Nozzle Type/Number	Limitation	Estimated Coverage
Core Spray (2 nozzles)	Thermal Sleeve and Sparger	40%
Reactor Recirculation Inlet (10 nozzles)	Thermal Sleeve/Jet Pump Riser	50%
Jet Pump Instrumentation (2 nozzles)	Instrumentation Lines	60%

The limited visual examination does not significantly reduce the level of plant quality and safety for the following reasons:
There are no mechanisms of damage, other than fatigue, for the nozzle inner radius and, other than feedwater nozzles, there is no cause for significant thermal cycling. Therefore, the primary flaw of concern would be a flaw that was not detected during the manufacturing process[1]. For the Brunswick Steam Electric Plant (BSEP), the nozzles were examined during and after manufacturing by surface and volumetric techniques.

1. Conclusions made in ASME NDE Subcommittee Report ISI-99-26, “Technical Basis for the Elimination of Reactor Vessel Nozzle Inner Radius Inspections.”

Additionally, preservice and inservice ultrasonic examinations have detected no flaws. It is unlikely that flaws will be initiated by the fatigue mechanism.

After approximately 25 years of operation (i.e., over 1000 reactor years of industry experience), no cracking of any kind in the subject nozzle inner radius regions has ever been found.

1. Fracture toughness tests performed at Oak Ridge National Laboratories indicate there is a large flaw tolerance for boiling water reactor (BWR) nozzle inner radius regions. Even if flaw propagation was assumed, test results indicate a leak before break scenario would occur which would not result in a significant increase in core damage frequency[1]. Additionally, pressure testing continues to be performed each refueling outage, and during plant operation, containment is monitored for changes in unidentified leakage.

2. More than 50 percent of the total nozzle population receives a complete nozzle inner radius examination and 40 percent of the total nozzle population has already received a volumetric examination during the current interval.

3. Visual examination of the accessible nozzle inner radius surface (i.e., zone M-N) provides reasonable assurance that deep flaws are not present. Additionally, when flaws are initiated by the fatigue mechanism, they are typically encountered over a significant portion of the nozzle circumference as was the case for cracking of feedwater nozzles addressed in NUREG-0619.

In summary, fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region and for all nozzles other than feedwater nozzles. There is no significant thermal cycling of these nozzles during operation. Therefore, from a risk perspective, only volumetric examinations of the feedwater nozzles and operational control rod drive (CRD) return line nozzles are necessary. CRD nozzles at BSEP, Units 1 and 2 are cut and capped. This is supported by the fact that no service-related cracking has been discovered in any of the BWR fleet plant nozzles other than on feedwater or operational CRD return lines. The four (i.e., per unit) feedwater nozzle inner radius sections will continue to be examined with ultrasonic techniques developed and qualified in accordance with Topical Report GE-NE-523-A71-0594-A, Revision 1.

Additionally, Relief Request RR-30 [evaluated later in this report] provides for a full visual examination coverage (i.e., greater than 90 percent examination coverage, as defined by NRC Information Notice 98-42, “Implementation of 10 CFR 50.55a(g) Inservice Inspection Requirements”) of ten additional nozzles resulting in complete examination of more than 50 percent of the total nozzle population. CP&L believes that the partial visual examination alternative for the nozzle inner radius regions above results in a significant reduction in personnel dose and still ensures an acceptable level of quality and safety.

In response to an NRC request, the licensee provided further information:

The nozzle inner radius remote visual inspections will be performed using a site-specific procedure that incorporates the Enhanced VT-1 (EVT-1) requirements described in the EPRI technical report entitled *TR-105696-R4 (BWRVIP-03) Revision 4: BWR Vessel and Internals Project, Reactor Pressure Vessel and Intemals Examination Guidelines*. The Enhanced VT-1 described in this report is a visual inspection method with equipment and environmental conditions such that the inspection can achieve a ½ mil resolution [detection sensitivity]. This procedure also meets the requirements of the 1989 edition of the ASME Code, Section XI.

The resolution [detection sensitivity] check will be performed prior to, and at the completion of, each examination or series of examinations. The lens-to-object distance and lighting required to discern the wire will be at the maximum distance that will be used during the examination. The procedure to be used also addresses lens-to-object distances, adequacy of lighting, water clarity, and surface cleanliness. These attributes are determined by the level of detail to which the following conditions can be seen:

Grinding/machine marks
Undercut
Weld ripples/beads
Pre-existing indications

In addition, independent reviews are performed by CP&L, vendor, and Authored Nuclear Inservice Inspector (ANII) personnel to ensure procedural requirements are met.

The inspection system is capable of magnification; however, there is no minimum magnification capability required by the EVT-1 procedure. Adequate resolution [detection sensitivity] is determined using a ½ mil wire. Flaw lengths will be measured using any means that can be demonstrated under actual conditions to have repeatable accuracy (e.g., a ruler on a pole, ultrasonic testing). Table IWB-3512-l of the 1989 edition of the ASME Code, Section XI, provides limits for allowable planar flaws. Table IWB-3512-1 does not require flaw length determination for inside comer regions (i.e., for the nozzle inner radii, as shown in Figure IWB-2500-7(b)). Rather, only the a/t ratio, the flaw depth divided by the wall thickness, is required. Therefore, for the nozzle inner radius examinations, determination of an aspect (a/l) ratio is not required. If there is a need to measure length, the EVT-1 method will be used. The through-wall dimension will be measured using a volumetric (i.e., ultrasonic examination) method.

The EVT-1 inspection procedure requires that the surfaces be free from any conditions that would prevent detection of the smallest expected anomaly. The procedure also requires cleanliness to be discussed during a 4 hour indoctrination training session which is required for all inspection personnel. Cleaning methods, which have been used at Brunswick for previous shroud examinations, are also addressed in the procedure.

<u>Evaluation:</u> The Code requires 100% volumetric examination for the inner radius sections of RPV nozzles. In boiling water reactors (BWRs), such as Brunswick 1-2, ultrasonic examinations are performed on nozzle inner radius sections from the outside surface of the vessel, which is accessed through penetrations, or “windows,” in the biological shield wall. Examiners must remain inside the biological shield penetrations for the duration of the examinations, where dose rates are typically in the range of 200 mr/hr, or greater. The examinations contribute significantly to personnel exposures, with estimated cumulative dose rates of 3.0 rem per inspection interval for the subject components.

During the mid-1970s, fatigue-initiated cracking was discovered in the nozzle inner radius sections of feedwater nozzles at eighteen BWRs. The cracks were found during routine remote visual examinations of the reactor vessel internals that are performed each refueling outage. Ultrasonic testing (UT), with the technology of that era, failed to reveal the presence of these cracks. This prompted the NRC to prepare NUREG-0619, which modified inspection requirements for these components. In NUREG-0619, the NRC staff concluded that UT of vessel nozzle inner radius sections involves complex geometrical configurations, long examination metal paths, and inherent difficulties associated with UT beam spread, scatter, and attenuation. During the intervening years, improvements in UT technologies were introduced (e.g., computer modeling, tip diffraction, and phased array scanning), which improved the quality of the examination for these components. However, the areas remain difficult to examine completely.

The staff finds that while improved UT technology is applied from the outside surface, the complex geometries and extended sound paths for these RPV nozzle inner radius sections may prevent full volumetric coverage. For the components listed above, the licensee proposed to perform what has been identified as an enhanced VT-1 (EVT-1) visual examination on the accessible portions of the nozzle inner radius sections in lieu of UT. These will be remote examinations conducted from the vessel interior. Enhanced in this case refers to the ½-mil wire standard that is to be demonstrated by the examiners to assure acceptable sensitivity. The license stated that estimated coverages will be in the range of 40-60% for the inner surfaces of the subject nozzle inner radius sections. The licensee also indicated that measures described in the EVT-1 procedure will be taken to assure that examination parameters, including lighting, lens-to-object distances, water clarity and surface conditions will be adequate to detect the smallest expected anomaly, i.e., a fatigue crack whose surface opening is greater than ½-mil, for the duration of these examinations.

The primary degradation mode in the inner radius of RPV nozzles is thermal fatigue, which would typically produce a network of hairline surface indications along the circumference of the nozzle in the inner radius section. Further, the flaws are generally encountered over a significant portion of this circumference, as was the case for cracking of feedwater nozzles addressed in NUREG-0619. Given the resolution capability of the EVT-1 visual examination, it is expected that the licensee would detect such flaws if they are located on the accessible surface areas of the nozzle inner radius sections. The licensee has stated that flaw acceptance will be based upon ASME Table IWB-3512-1. Therefore, while the proposed visual examination on these components will be limited to about 40-60% estimated coverage, it is felt that flaws of significant size will be detected. The staff also recognizes that the licensee, and other industry BWRs, have experienced no reported cracking in the subject nozzle inner radius regions, and that these specific nozzles are not subjected to substantive thermal cycling. In addition, it is noted that approximately 50% of the total RPV nozzle population will receive the Code-required volumetric examinations during the current interval at Brunswick 1-2, and that where historical fatigue cracking has occurred, e.g., feedwater nozzles, the licensee will continue to perform volumetric examinations with procedures developed and qualified in accordance with the Code.

The staff determined that considering the personnel radiation exposure that would be incurred if the licensee was required to perform the Code volumetric examinations it would result in a significant hardship without a compensating increase in quality and safety. Furthermore, the staff determined that the potential of the EVT-1 visual examinations to detect thermal fatigue degradation in the inspected regions of the nozzle inner radius sections, the absence of any historical industry evidence for cracking in the subject components, and the volumetric examinations being performed on other similar nozzles, provides reasonable assurance of the continued structural integrity of the subject nozzles has been provided. Therefore, it is recommended that, pursuant to

10 CFR 50.55a(a)(3)(ii), the licensee's proposed alternative to use EVT-1 visual examination in lieu of volumetric for the nozzle inner radius sections listed in Request for Relief RR-29, Revision 1, should be authorized.

3.2 Request for Relief RR-30, Revision 1, Examination Category B-D, Item B3.100, Full Penetration Welded Nozzles in Vessels, Reactor Pressure Vessel Nozzle Inside Radius Sections

Code Requirement: Examination Category B-D, Item B3.100, requires 100% volumetric examination, as defined by Figures IWB-2500-7(a) through (d), as applicable, of Class 1 reactor pressure vessel (RPV) nozzle inside radius sections.

Licensee's Proposed Alternative to Code: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee is proposing to use a visual examination of the accessible inside surface in lieu of the Code-required volumetric examination for the inner radius sections of RPV nozzles on the Control Rod Drive Return Line (N9), Reactor Recirculation (N1A and N1B), and Reactor Pressure Vessel Instrumentation (N11A, N11B, N16A and N16B) Systems. There are a total of seven nozzles included in this proposal. The licensee stated:

CP&L will perform a remote visual examination of the accessible surface M-N, as shown in Figures IWB-2500-7(a) through (d), in lieu of the volumetric examinations required by Table IWB-2500-l, Examination Category B-D, Item B3.100, for the RPV nozzles identified in this relief request. The remote visual examination will be performed using the Enhanced VT-1 (EVT-1) requirements described in the Electric Power Research Institute (EPRI) technical report entitled *TR-105696-R4 (BWRVIP-03) Revision 4: BWR Vessel and Internals Project, Reactor Pressure Vessel and Internals Examination Guidelines*. Required coverage will include essentially 100 percent (i.e., greater than 90 percent examination coverage, as defined by NRC Information Notice 98-42, "Implementation of 10 CFR 50.55a(g) Inservice Inspection Requirements") of the surface M-N as shown in Figures IWB-2500-7 [as applicable]. The sensitivity for remote in-vessel examinations will be established using the ½ mil wire standard described in that report.

Crack-like surface flaws exceeding the acceptance criteria of Table IWB-3512-1 will be deemed unacceptable for continued service unless the reactor vessel meets the requirements of IWB-3142.2, IWB-3142.3, or IWB-3142.4. This alternative is similar to the inspection alternative proposed in ASME Section XI Code Case N-648.

Licensee's Bases for Alternative (as stated):

Performance of the volumetric examinations results in significant personnel radiation exposure without a commensurate increase in the level of plant quality or safety for the following reasons:

The subject nozzle forgings were nondestructively examined using inservice ultrasonic techniques specific to the nozzle configuration. No indication of fabrication defects or service related cracking has been detected by these examinations.

1. Nozzle inner radius examinations are the only non-welded area requiring examination on the RPV. This requirement was deterministically made early in the development of the ASME Code, Section XI, and applied to 100 percent of

nozzles welded with full penetration welds. Fatigue cracking is the only applicable degradation mechanism for the nozzle inner radius region. Other than feedwater nozzles, there is no significant thermal cycling during operation. Therefore, from a risk perspective, there is no need to perform volumetric examination on any nozzles other than feedwater and operational control rod drive (CRD) return line nozzles. The BSEP, Unit 1 and 2 CRD return line nozzles are cut and capped. No service induced cracking has been discovered in any of the boiling water reactor (BWR) fleet plant nozzles other than on feedwater nozzles or operational CRD return line nozzles.

2. The four feedwater (i.e., per unit) nozzle inner radius sections will continue to be examined with ultrasonic techniques developed and qualified in accordance with Topical Report GE-NE-523-A71-0594-A, Revision 1. The NRC has previously approved this topical report under TAC No. MA6787. Including the feedwater nozzles, 40 percent of the applicable nozzle inner radius sections have been ultrasonically examined during the current interval.

3. The visual examination will cover the same inspection surface as specified for the volumetric examination.

In response to an NRC request, the licensee provided further information:

The nozzle inner radius remote visual inspections will be performed using a site-specific procedure that incorporates the Enhanced VT-1 (EVT-1) requirements described in the EPRI technical report entitled *TR-105696-R4 (BWRVIP-03) Revision 4: BWR Vessel and Internals Project, Reactor Pressure Vessel and Intemals Examination Guidelines*. The Enhanced VT-1 described in this report is a visual inspection method with equipment and environmental conditions such that the inspection can achieve a ½ mil resolution [detection sensitivity]. This procedure also meets the requirements of the 1989 edition of the ASME Code, Section XI.

The resolution [detection sensitivity] check will be performed prior to, and at the completion of, each examination or series of examinations. The lens-to-object distance and lighting required to discern the wire will be at the maximum distance that will be used during the examination. The procedure to be used also addresses lens-to-object distances, adequacy of lighting, water clarity, and surface cleanliness. These attributes are determined by the level of detail to which the following conditions can be seen:

Grinding/machine marks
Undercut
Weld ripples/beads
Pre-existing indications

In addition, independent reviews are performed by CP&L, vendor, and Authored Nuclear Inservice Inspector (ANII) personnel to ensure procedural requirements are met.

The inspection system is capable of magnification; however, there is no minimum magnification capability required by the EVT-1 procedure. Adequate resolution [detection sensitivity] is determined using a ½ mil wire. Flaw lengths will be measured using any means that can be demonstrated under actual conditions to have repeatable accuracy (e.g., a ruler on a pole, ultrasonic testing). Table IWB-3512-l of the 1989 edition of the ASME Code, Section XI, provides limits for allowable planar flaws. Table

IWB-3512-1 does not require flaw length determination for inside comer regions (i.e., for the nozzle inner radii, as shown in Figure IWB-2500-7(b)). Rather, only the a/t ratio, the flaw depth divided by the wall thickness, is required. Therefore, for the nozzle inner radius examinations, determination of an aspect (a/l) ratio is not required. If there is a need to measure length, the EVT-1 method will be used. The through-wall dimension will be measured using a volumetric (i.e., ultrasonic examination) method.

The EVT-1 inspection procedure requires that the surfaces be free from any conditions that would prevent detection of the smallest expected anomaly. The procedure also requires cleanliness to be discussed during a 4 hour indoctrination training session which is required for all inspection personnel. Cleaning methods, which have been used at Brunswick for previous shroud examinations, are also addressed in the procedure.

<u>Evaluation:</u> The Code requires 100% volumetric examination for the inner radius sections of RPV nozzles. In boiling water reactors (BWRs), such as Brunswick 1-2, ultrasonic examinations are performed on nozzle inner radius sections from the outside surface of the vessel, which is accessed through penetrations, or "windows," in the biological shield wall.

During the mid-1970s, fatigue-initiated cracking was discovered in the nozzle inner radius sections of feedwater nozzles at eighteen BWRs. The cracks were found during routine remote visual examinations of the reactor vessel internals that are performed each refueling outage. Ultrasonic testing (UT), with the technology of that era, failed to reveal the presence of these cracks. This prompted the NRC to prepare NUREG-0619, which modified inspection requirements for these components. In NUREG-0619, the NRC staff concluded that UT of vessel nozzle inner radius sections involves complex geometrical configurations, long examination metal paths, and inherent difficulties associated with UT beam spread, scatter, and attenuation. During the intervening years, improvements in UT technologies were introduced (e.g., computer modeling, tip diffraction, and phased array scanning), which improved the quality of the examination for these components. However, the areas remain difficult to examine completely.

The staff finds that while improved UT technology is applied from the outside surface, the complex geometries and extended sound paths for these RPV nozzle inner radius sections may prevent full volumetric coverage. For the components listed above, the licensee proposed to perform what has been identified as an enhanced VT-1 (EVT-1) visual examination on "essentially 100%" of the nozzle inner radius section surfaces in lieu of UT. These will be remote examinations conducted from the vessel interior. Enhanced in this case refers to the ½-mil wire standard that is to be demonstrated by the examiners to assure acceptable sensitivity.

The primary degradation mode in RPV nozzles is thermal fatigue, which would typically produce a network of hairline surface indications along the circumference of the nozzle in the inner radius section. Given the visual sensitivity of the EVT-1 visual examination, and the ability to access essentially 100% of the inside surface, it is expected that the licensee would detect any significant fatigue degradation that may occur in accessible areas of the nozzle inner radius sections. The licensee has stated that flaw acceptance will be based upon ASME Table IWB-3512-1. The staff also recognizes that the licensee, and other industry BWRs, have experienced no reported cracking in the subject nozzle inner radius regions, and that these specific nozzles are not subjected to substantive thermal cycling. In addition, it is noted that approximately 50% of the total RPV nozzle population will receive the Code-required volumetric examinations during the current interval at Brunswick 1-2, and that where historical fatigue cracking has occurred, e.g., feedwater nozzles, the licensee will continue to perform volumetric examinations with procedures developed and qualified in accordance with the Code.

Therefore, the staff determined that based on above the licensee's proposed alternative to use EVT-1 visual examinations to detect thermal fatigue degradation in the inspected regions of the nozzle inner radius sections provides a reasonable acceptable level of quality and safety.

4.0 CONCLUSIONS

Based on the information provided in the licensee's submittal, it has been concluded that complying with the specified Code volumetric requirements would result in hardship without a compensating increase in quality and safety, and the proposed alternative described in Request for Relief RR-29, Revision 1, provides reasonable assurance of the continued structural integrity of the nozzle inner radius sections listed in the proposal. Therefore, it is recommended that, RR-29, Revision 1, be authorized pursuant to 10 CFR 50.55a(a)(3(ii) for the third 10-year inspection intervals at Brunswick 1-2, which ends on May 10, 2008. This authorization is limited to the components described in Section 3.1 above.

Further, for Request for Relief RR-30, Revision 1, it has been determined that the proposed alternative described in the licensee's submittal provides an acceptable level of quality and safety. Therefore, it is recommended that RR-30, Revision 1, be authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the third 10-year inspection intervals at Brunswick 1-2, which ends on May 10, 2008. This authorization is limited to the components described in Section 3.2 above.

ATTACHMENT 1

BRUNSWICK STEAM ELECTRIC PLANT, UNITS 1 AND 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 Disposition
RR-29, Revision 1	3.1	RPV nozzles: N5A, N5B, N2A thru N2K, N8A, N8B	B-D	B3.100	Inside radius sections	Volumetric	Use remote enhanced VT-1 from vessel interior on 40-60% accessible surfaces	Authorized 10 CFR 50.55a(a)(3)(ii)
RR-30, Revision 1	3.2	RPV nozzles: N9, N1A, N1B, N11A, N11B, N16A, N16B	B-D	B3.100	Inside radius sections	Volumetric	Use remote enhanced VT-1 from vessel interior on essentially 100% accessible surfaces	Authorized 10 CFR 50.55a(a)(3)(i)

ATTACHMENT 2

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