CP&L
A Progress Energy Company

10 CFR 50.55a(a)(3)(i)
10 CFR 50.55a(a)(3)(ii)

JUL 16 2002

SERIAL: BSEP 02-0116

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
DOCKET NOS. 50-325 AND 50-324/LICENSE NOS. DPR-71 AND DPR-62
REQUEST FOR APPROVAL OF RELIEF REQUESTS FOR THE THIRD 10-YEAR INSERVICE INSPECTION PROGRAM – REACTOR PRESSURE VESSEL NOZZLE INNER RADIUS VOLUMETRIC EXAMINATIONS

Ladies and Gentlemen:

In accordance with 10 CFR 50.55a(a)(3)(i) and 10 CFR 50.55a(a)(3)(ii), Carolina Power & Light (CP&L) Company is requesting relief from the requirements of the 1989 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, for the Brunswick Steam Electric Plant (BSEP), Unit Nos. 1 and 2. CP&L is requesting relief from ASME Code, Section XI requirements to perform a volumetric examination of the inner radius of all reactor pressure vessel nozzles welded with full penetration welds.

Relief Requests RR-29 and RR-30 are provided in Enclosures 1 and 2, respectively. Relief Request RR-29 addresses reactor pressure nozzles for which only partial examination coverage can be achieved; therefore, this relief is based on hardship in accordance with 10 CFR 50.55a(a)(3)(ii). Relief Request RR-30 proposes an alternate examination, in accordance with 10 CFR 50.55a(a)(3)(i), for reactor pressure vessel nozzles where the plant configuration allows essentially 100 percent visual examination coverage inspection of the nozzle inner radius.

Approval of these relief requests are needed to support inservice inspection activities during next BSEP Unit 2 refueling outage, which is currently scheduled to begin March 1, 2003. Therefore, to support planning activities for this refueling outage, CP&L requests that NRC review and approval of these relief requests be completed no later February 1, 2003.

Two similar relief requests have been previously approved for the Detroit Edison Company's Fermi Unit 2 by letter dated October 5, 2001 (i.e., ADAMS Accession Number ML0126905460).

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Brunswick Nuclear Plant
P.O. Box 10429
Southport, NC 28461

Please refer any questions regarding this submittal to Mr. Leonard R. Beller, Supervisor - Licensing/Regulatory Programs, at (910) 457-2073.

Sincerely,

Edward T. O'Neil
Manager - Regulatory Affairs
Brunswick Steam Electric Plant

WRM/wrm

Enclosures:

1. Relief Request RR-29, Revision 0, "RPV Nozzle Inner Radius Volumetric Examinations"
2. Relief Request RR-30, Revision 0, "RPV Nozzle Inner Radius Volumetric Examinations"

cc (with enclosures):

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ENCLOSURE 1

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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REQUEST FOR APPROVAL OF RELIEF REQUESTS
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Relief Request RR-29, Revision 0,
"RPV Nozzle Inner Radius Volumetric Examinations"

RELIEF REQUEST: RR-29, Revision 0

SUBJECT: RPV Nozzle Inner Radius Volumetric Examinations

COMPONENTS FOR WHICH RELIEF IS REQUESTED:

This relief request is applicable to the inner radius sections for the following Reactor Pressure Vessel (RPV) nozzles:

Nozzle N5A (Core Spray)
Nozzle N5B (Core Spray)
Nozzle N2A (Reactor Recirculation System Inlet)
Nozzle N2B (Reactor Recirculation System Inlet)
Nozzle N2C (Reactor Recirculation System Inlet)
Nozzle N2D (Reactor Recirculation System Inlet)
Nozzle N2E (Reactor Recirculation System Inlet)
Nozzle N2F (Reactor Recirculation System Inlet)
Nozzle N2G (Reactor Recirculation System Inlet)
Nozzle N2H (Reactor Recirculation System Inlet)
Nozzle N2J (Reactor Recirculation System Inlet)
Nozzle N2K (Reactor Recirculation System Inlet)
Nozzle N8A (Jet Pump Instrumentation)
Nozzle N8B (Jet Pump Instrumentation)

ASME SECTION XI CODE REQUIREMENT:

The American Society of Mechanical Engineers (ASME) Code, Section XI, 1989 Edition, Table IWB-2500-1 for Examination Category B-D, requires a volumetric examination of the inner radius section of all RPV nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

REQUESTED RELIEF:

In accordance with 10 CFR 50.55a(a)(3)(ii), Carolina Power & Light (CP&L) Company is requesting relief from the ASME Code, Section XI requirements to perform the volumetric examination described below.

PROPOSED ALTERNATIVE:

CP&L will perform a 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 inservice examination of RPV nozzles identified in this relief request.

The resolution sensitivity for remote in-vessel examinations will be established using a 1-mil wire standard similar to that used for other RPV internal examinations intended to detect cracking.

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.

BASIS FOR REQUESTING RELIEF:

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 - 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 subject 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).

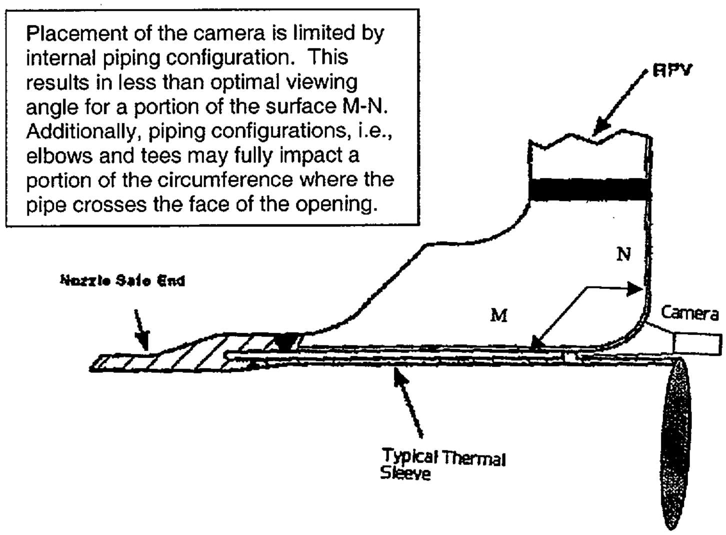

Figure 1
Typical Cross Section of BWR Nozzle with Internal Piping

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. Additionally, preservice

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

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.

- Fracture toughness tests performed at Oakridge 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.

- 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.

- 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 nozzles 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 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

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

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.

REFERENCES:

1. ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1989 Edition with no Addenda.

2. Conclusions made in ASME Non-Destructive Examination Subcommittee Report ISI-99-26, "Technical Basis for Elimination of Reactor Pressure Vessel Nozzle Inner Radius Inspections."

3. General Electric Topical Report GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle Inspection Requirements."

4. CP&L Relief Request RR-30, "RPV Nozzle Inner Radius Volumetric Examinations."

ENCLOSURE 2

BRUNSWICK STEAM ELECTRIC PLANT, UNIT NOS. 1 AND 2
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REQUEST FOR APPROVAL OF RELIEF REQUESTS
FOR THE THIRD 10-YEAR INSERVICE INSPECTION PROGRAM – REACTOR PRESSURE VESSEL NOZZLE INNER RADIUS VOLUMETRIC EXAMINATIONS

Relief Request RR-30, Revision 0,
"RPV Nozzle Inner Radius Volumetric Examinations"

RELIEF REQUEST: RR-30, Revision 0

SUBJECT: RPV Nozzle Inner Radius Volumetric Examinations

COMPONENTS FOR WHICH RELIEF IS REQUESTED:

This relief request is applicable to the inner radius sections for the following Reactor Pressure Vessel (RPV) Nozzles:

Nozzle N9 (Control Rod Drive Return Line Nozzle)
Nozzle N1A (Reactor Recirculation System Suction)
Nozzle N1B (Reactor Recirculation System Suction)
Nozzle N11A (Reactor Pressure Vessel Instrumentation)
Nozzle N11B (Reactor Pressure Vessel Instrumentation)
Nozzle N16A (Reactor Pressure Vessel Instrumentation)
Nozzle N16B (Reactor Pressure Vessel Instrumentation)

ASME SECTION XI CODE REQUIREMENT:

The American Society of Mechanical Engineers (ASME) Code, Section XI, 1989 Edition, Table IWB-2500-1 for Examination Category B-D, requires a volumetric examination of the inner radius section of all RPV nozzles welded with full penetration welds as shown in Figures IWB-2500-7(a) through (d).

REQUESTED RELIEF:

In accordance with 10 CFR 50.55a(a)(3)(i), Carolina Power & Light (CP&L) Company is requesting relief from ASME Code, Section XI requirements to perform the volumetric examination alternative described below by substituting a visual examination of the accessible surface M-N, as shown in Figures IWB-2500-7(a) through (d).

PROPOSED ALTERNATIVE:

CP&L will perform a 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 inservice examination of RPV nozzles identified in this relief request. 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 Figure IWB-2500-1. The resolution sensitivity for remote in-vessel examinations will be established using a 1-mil wire standard similar to that used for other RPV internal examinations intended to detect cracking.

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.

RELIEF REQUEST: RR-30, Revision 0

SUBJECT: RPV Nozzle Inner Radius Volumetric Examinations

This alternative is similar to the inspection alternative proposed in ASME Section XI Code Case N-648.

BASIS FOR REQUESTING RELIEF:

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.

- 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.

- 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.

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

REFERENCES:

1. ASME Code, Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 1989 Edition with no Addenda.

2. Code Case N-648, "Alternate Requirements for Inner Radius Examinations of Class 1 Reactor Pressure Vessel Nozzles, Section XI, Division 1."

3. General Electric Topical Report GE-NE-523-A71-0594-A, Revision 1, "Alternate BWR Feedwater Nozzle inspection Requirements."

4. CP&L Relief Request RR-29, "RPV Nozzle Inner Radius Volumetric Examinations."