April 22, 2003

Mr. Michael Kansler, President
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

SUBJECT: REQUEST FOR RELIEF NOS. 2-CLOSEOUT-1 THROUGH 2-CLOSEOUT-6 TO THE INSERVICE INSPECTION PLAN, INDIAN POINT NUCLEAR GENERATING UNIT NO. 3 (TAC NO. MB2535)

Dear Mr. Kansler:

By letter dated July 16, 2001, as supplemented October 9, 2002, Entergy Nuclear Operations, Inc. (ENO or the licensee) submitted Request for Relief (RR) Nos. 2-Closeout-1 through 2-Closeout-6 which were associated with the closeout of the second 10-year interval for the inservice inspection (ISI) interval at Indian Point Nuclear Generating Unit No. 3 (IP3). Specifically, ENO requested in RR 2-Closeout-1 to use a proposed alternative to the augmented reactor vessel examination requirements specified in Section 50.55(g)(6)(ii)(A) of Title 10 of the *Code of Federal Regulations* (10 CFR). In RR 2-Closeout-2 through 6, the licensee sought relief from the 100% examination coverage requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," pursuant to 10 CFR 50.55a(g)(6)(i).

The U.S. Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, Pacific Northwest National Laboratory, has completed its review of the proposed relief requests. The results are provided in the enclosed safety evaluation.

The NRC staff finds that the alternative proposed in RR 2-Closeout-1, Revision 1, provides an acceptable level of quality and safety. The staff also finds that the completed inspections provide reasonable assurance of the continued structural integrity of the welds. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the NRC staff authorizes the proposed alternative for the second 10-year ISI interval at IP3.

For RR 2-Closeout-2 through 2-Closeout-6, the NRC staff concludes that it is impractical for the licensee to comply with the 100% volumetric examination requirements specified in the Code. The staff finds that reasonable assurance of the structural integrity of the specified components has been provided based on the examinations that were performed. Therefore, the licensee is granted relief pursuant to 10 CFR 50.55a(g)(6)(i) for the second 10-year ISI interval at IP3. The granting of relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

If you should have any questions, please contact the IP3 Project Manager, Patrick Milano, at 301-415-1457.

Sincerely,

***/RA/***

Richard J. Laufer, Chief, Section 1
Project Directorate 1
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosure: Safety Evaluation

cc w/encl: See next page

If you should have any questions, please contact the IP3 Project Manager, Patrick Milano, at 301-415-1457.

Sincerely,

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Enclosure: Safety Evaluation

cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

REQUEST FOR RELIEF NOS. 2-CLOSEOUT-1 THROUGH 2-CLOSEOUT-6

ENTERGY NUCLEAR OPERATIONS, INC.

INDIAN POINT NUCLEAR GENERATING UNIT NO. 3

DOCKET NO. 50-286

## 1.0 INTRODUCTION

By letter dated July 16, 2001 (Reference 1), as supplemented on October 9, 2002 (Reference 2), Entergy Nuclear Operations, Inc. (ENO or the licensee) submitted Request for Relief (RR) Nos. 2-Closeout-1 through 2-Closeout-6. In RR 2-Closeout-1, the licensee proposed an alternative to the augmented reactor vessel examination requirements specified in Section 50.55a(g)(6)(ii)(A)) of Title 10 of the *Code of Federal Regulations* (10 CFR). In RRs 2-Closeout-2 though RR 2-Closeout-6, the licensee sought relief from certain requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Each of the RRs are for the second 10-year inservice inspection (ISI) interval at Indian Point Nuclear Generating Unit No. 3 (IP3).

## 2.0 REGULATORY EVALUATION

ISI of the ASME Code Class 1, 2, and 3 components is performed in accordance with Section XI of the ASME Code and applicable addenda as required by 10 CFR 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 Nuclear Regulatory Commission (NRC), if the licensee demonstrated 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 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 (10-year) interval, subject to the limitations and modifications listed therein. The applicable Code of record for IP3 during the second 10-year ISI interval is the 1983 Edition of the ASME Code, Section XI, Summer 1983 Addenda.

## 3.0 TECHNICAL EVALUATION

The NRC staff, with technical assistance from its contractor, Pacific Northwest National Laboratory (PNNL), reviewed and evaluated the information submitted by the licensee in a letter dated July 16, 2001, as supplemented on October 9, 2002. The NRC staff adopts the evaluations and recommendations for the authorizing of alternatives and for the granting of relief, as applicable, contained in the attached Technical Letter Report (TLR), prepared by PNNL.

### 3.1 Request for Relief 2-Closeout-1

For RR 2-Closeout-1 (Revision 1), the 100% volumetric examination required in the Code is not achievable on reactor pressure vessel (RPV) shell longitudinal weld #5 due to a design structural discontinuity that restricts access. Scanning by ultrasonic testing (UT) of the portions of the long seam weld #5 near the nozzle is limited due to interference from the nozzle boss. The licensee estimated that 76% of the weld was inspected. It would be an excessive burden on the licensee to perform the Code required ultrasonic examinations, because a design modification or replacement of the nozzle to provide for complete coverage would be required. All other reactor vessel welds were examined in accordance with 10 CFR 50.55a(g)(6)(ii)(A). No reportable indications were found. Additionally, a visual examination has been performed in conjunction with the pressure testing on these components during scheduled refueling outages. The NRC staff determined the coverages and inspections obtained by the licensee provide reasonable assurance of the continued structural integrity of these welds, and an acceptable level of quality and safety.

### 3.2 Request for Relief 2-Closeout-2

For RR 2-Closeout-2 (Revision 0), the 100% volumetric examination required in the Code is impractical for the RPV outlet nozzle to vessel weld nos. 22, 23, 26, and 27. The licensee has provided documentation showing that for the outlet nozzles, the nozzle boss or protrusion completely obstructs scanning of the portion of the examination volume in the nozzle barrel. Coverage is estimated at 43% from the vessel inside surface and 100% from the bore. Imposing the requirement to perform the Code-required examination would be an excessive burden on the licensee because the RPV outlet nozzles would need to be replaced with a modified design. The licensee has also performed a visual examination in conjunction with pressure testing on these components during every refueling outage, with no evidence of leakage detected. The NRC staff finds the coverages and inspections obtained by the licensee, and the absence of any service degradation experience, provide reasonable assurance of the continued structural integrity of the RPV nozzle to vessel welds.

3.3 Request for Relief 2-Closeout-3

For RR 2-Closeout-3 (Revision 0), the 100% volumetric examination required in the Code is impractical for the steam generator nozzle inside radius sections (nos. 31-1A, 31-1B, 32-1A, 32-1B and 33-1A). The licensee has provided documentation that the support lugs limit the access on the flat transition and blended radius on the steam generator hot and cold leg nozzle inside radius sections. Coverage is estimated at 50%. Imposing the requirement to perform the Code-required examination would be an excessive burden on the licensee since the steam generator and adjacent reactor coolant system hot and cold leg integral support lugs would need to be replaced with a modified design. The NRC staff finds the coverages and inspections obtained by the licensee, and the absence of any industry service degradation experience, provide reasonable assurance of the continued structural integrity of the steam generator nozzle inside radius sections.

3.4 Request for Relief 2-Closeout-4

For RR 2-Closeout-4 (Revision 0), the 100% volumetric examination required in the Code is impractical for the following piping branch connection welds (16BC Accumulator Discharge, 33BC Reactor Coolant, and 13BC Accumulator Discharge), due to restricted access caused by various designs and geometry limitations. The limitations are caused primarily by the weld's proximity to other components, such as a valve, integrally welded support, or an adjacent nozzle. The licensee estimated that a maximum of 50% of the required weld examination coverage could be expected based on single-sided access for austenitic stainless steel weld material. The licensee has also conducted a visual inspection on these components during past refueling outages with no leakage detected. Imposing the requirement to perform the Code-required examination would be an excessive burden on the licensee since significant support redesign and plant modifications would need to be made for each branch connection to achieve 100% coverage. The NRC staff finds the coverages and inspections performed by the licensee, and the absence of any service degradation experience, provide reasonable assurance of the continued structural integrity of the piping branch connection welds.

3.5 Request for Relief 2-Closeout-5

For RR 2-Closeout-5 (Revision 1), the 100% surface or volumetric examination required by the Code is impractical for the 6 reactor coolant pump (RCP) integrally welded support attachments (3 on RCP No. 31 and 3 on RCP No. 32) because of obstructions created due to design configurations. The configuration restraints, which are created due to restrictions from the lower support structure, allow an estimated 75% of the support lugs to be examined. The licensee performed inspections on 6 of the 12 integrally welded attachments. A visual examination was also performed in conjunction with the pressure testing on these components during past fueling outages, with no leakage detected. Imposing the requirement to perform the Code-required examination would be an excessive burden on the licensee because significant support redesign and plant modifications would need to be made for each RCP. The NRC staff finds the coverages and inspections performed by the licensee, and the absence of any service degradation experience, provide reasonable assurance of the continued structural integrity of the integrally welded support attachments.

3.6 Request for Relief 2-Closeout-6

For RR 2-Closeout-6 (Revision 0), the 100% volumetric examination required by the Code is impractical for the welds identified in Table 6 of the TLR due to restricted access caused by interference from the close proximity of the vessel flange bolting, nozzles, and support clamps. The licensee estimates at least 50% volumetric coverage on these welds. A visual inspection was performed during the pressure testing on this component during every refueling outage, with no leakage detected. Imposing the requirement to perform the Code-required examination would be an excessive burden on the licensee because the components would have to be redesigned and modified. The NRC staff determined that reasonable assurance of the structural integrity of the subject components has been provided based on the examinations that were performed.

## 4.0 CONCLUSION

The NRC staff concludes that the proposed alternative for RR 2-Closeout-1 provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), the alternative in RR 2-Closeout-1 is authorized for the second 10-year ISI interval at IP3.

The NRC staff concludes that the Code examination requirements are impractical for RRs 2-Closeout-2, 2-Closeout-3, 2-Closeout-4, 2-Closeout-5, and 2-Closeout-6. Based on the examinations that were performed, there is reasonable assurance of the integrity of the components listed in the RRs has been provided. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), relief is granted for the second 10-year ISI interval at IP3.

The granting of relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility.

## 5.0 REFERENCES

1. Entergy Letter, Michael R. Kansler to NRC, dated July 16, 2001, “Second 10-Year ISI Interval Closeout and Associated Relief Requests,” enclosing “Indian Point Nuclear Generating Unit No. 3, Second 10-Year ISI Interval Closeout and Associated Relief Requests, Rev. 0.”

2. Entergy Letter, John Herron to NRC, dated October 9, 2002, “Response to Request for Additional Information Regarding Relief Requests for Second 10-Year ISI Interval Closeout.”

Attachment: Technical Letter Report

Principal Contributor: E. Reichelt

Date: April 22, 2003

**TECHNICAL LETTER REPORT**
**ON THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION**
**REQUESTS FOR RELIEF**
**FOR**
**ENTERGY OPERATIONS, INC.**
**INDIAN POINT NUCLEAR GENERATING UNIT 3**
**DOCKET NUMBER: 50-286**

## 1.0 INTRODUCTION

By letter dated July 16, 2001, as supplemented October 9, 2002, the licensee, Entergy Nuclear Operations, Inc. (Entergy), submitted Requests for Relief RR2-Closeout 1 through RR2-Closeout 6. Request for Relief RR2-Closeout 1 seeks approval for an alternative to the augmented reactor vessel examination requirements specified in 10CFR50.55a(g)(6)(ii)A. Requests for Relief RR2-Closeout 2 through 6 seek relief from 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*. These requests are for the second 10-year inservice inspection (ISI) interval at Indian Point Nuclear Generating Unit No. 3. The Pacific Northwest National Laboratory (PNNL) has evaluated the subject requests for relief in the following section.

## 2.0 EVALUATION

The information provided by Entergy in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below. The Code of record for Indian Point Nuclear Generating Unit No. 3, second 10-year interval, which started on August 30, 1986 and ended on July 20, 2000, is the 1983 Edition of ASME Section XI, including Summer 1983 Addenda.

### 2.1. Request for Relief RR 2-Closeout-1, Revision 1, 10 CFR 50.55a(g)(6)(ii)A, "Augmented Examination of Reactor Vessel"

Regulatory Requirement: 10 CFR 50.55a(g)(6)(ii)(A) *Augmented Examination of Reactor Vessel*, requires licensees to augment their reactor vessel examination by implementing a one time examination of all reactor vessel shell welds. The augmented examinations are to be performed in accordance with the requirements specified in Item B1.10 of Examination Category B-A, Table IWB-2500 of the 1989 Edition of the ASME Boiler and Pressure Vessel Code, Section XI. The examination requirement for the reactor vessel shell welds is a volumetric examination on essentially 100% of the welds. In accordance with 10 CFR 50.55a(g)(6)(ii)(A)(5), if a licensee determines they are unable to satisfy the augmented examination requirements, an alternative may be proposed that provides an acceptable level of quality and safety.

Licensee's Proposed Alternative: The licensee has proposed an alternative to the essentially 100% coverage requirement for RPV longitudinal Weld #5.

Licensee's Basis for the Proposed Alternative (as stated):

Entergy Nuclear Operations, Inc. (ENO) performed augmented volumetric examination of all the reactor vessel shell welds as required per 10 CFR 50.55a(g)(6)(ii)(A). However, for reactor vessel shell longitudinal weld #5, essentially 100% inspection was not achievable due to a design structural discontinuity that restricted access. The discontinuity was due to the cutout for the outlet nozzle, centered at 22 degrees. The effective length of the long seam weld #5 is 70 inches. Scanning of the portions of the long seam weld #5 near the nozzle is limited due to interference from the nozzle boss. Inspection coverage is estimated at 76%. (Ref. Technique drawings sheets 1, 14, & 15 attached). All other reactor vessel shell welds were examined as required per 10 CFR 50.55a(g)(6)(ii)(A). No reportable indications were found.

The RPV Shell Longitudinal Weld #5 has been examined to the maximum extent practical from the inside surface. No additional volumetric examinations will be performed on this weld.

A visual examination (VT-2) has been performed in conjunction with the pressure testing conducted on these components in every refuel outage in accordance with IWA-5000 and IWB-5000. This provides additional reasonable assurance of component integrity.

Evaluation: For compliance with the augmented reactor vessel examination requirements, licensee's must volumetrically examine essentially 100% (greater than 90%) of each of the Code Item B1.10 circumferential and longitudinal shell welds. The licensee submitted documentation showing that 100% volumetric coverage of RPV shell longitudinal Weld #5 is not feasible due to ultrasonic scanning interferences caused by the RPV outlet nozzle cut-out geometry. To achieve complete volumetric coverage, design modifications or replacement of the nozzle with one of a design providing for complete coverage would be required. Imposition of this requirement would cause a considerable burden on the licensee.

The volumetric examinations of the RPV shell welds were performed to the extent possible from the inside surface using mechanized inspection equipment. The effective length of the longitudinal seam Weld #5 is 70 inches. Scanning of the portions of this weld adjacent to the nozzle is limited due to interference from the nozzle boss. Inspection coverage is estimated at 76%. All other reactor vessel shell welds were fully examined as required per 10 CFR 50.55a(g)(6)(ii)(A), with no reportable indications being found. Based on the significant percentage of coverage obtained for Weld #5, in combination with essentially 100% coverage of the remaining reactor pressure vessel welds subject to examination, it is believed that significant degradation, if present, would have been detected. Based on review of the information submitted by the licensee, it is concluded that the licensee has maximized examination coverage for RPV shell weld #5, and that the coverage obtained by the licensee provide reasonable assurance of the continued structural integrity of the weld, and an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5), it is recommended that the licensee's alternative be authorized.

2.2 Request for Relief RR 2-Closeout-2, Revision 0, Code Category B-D, Item B3.90, Reactor Vessel Nozzle-to-Vessel Welds

Code Requirement: Examination Category B-D, Item B3.90, requires essentially 100% volumetric, as defined by Figure IWB-2500-7, of all reactor pressure vessel nozzle-to-vessel welds during each successive inspection interval ($2^{nd}$, $3^{rd}$, and $4^{th}$). "Essentially 100%," as clarified by ASME Code Case N-460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee's Code Relief Request: In accordance with the provisions of 10 CFR 50.55a(g)(5)(iii), the licensee has requested relief from the full extent of the Code-required volumetric examination for RPV outlet nozzle-to-vessel Weld Numbers 22, 23, 26, and 27.

Licensee's Basis for Relief Request (as stated):

Inspections of the Code-required accessible length of all nozzle-to-vessel welds were conducted from the bore in axial scans and from the vessel inside diameter surface to examine for transverse defects in accordance with RG 1.150, Rev. 1 and the Westinghouse Position Paper (Reference SER, TAC No. 72247, Page 8, dated November 7, 1991, Relief Request RR 2-1, Rev. 0 for which the NRC staff had previously determined no relief is required).

For the outlet nozzles, the nozzle boss or protrusion completely obstructs scanning of the portion of the examination volume in the nozzle barrel. Coverage is estimated at 100% from the bore, 43% from the vessel ID.

Licensee's Proposed Alternative Examinations (as stated):

No additional volumetric examinations will be performed. The RPV Nozzle-to-Vessel Welds #22, #23, #26, and #27 have been examined to the maximum extent practical from the inside surface.

A visual examination (VT-2) has been performed in conjunction with the pressure testing on these components every refuel outage (with no evidence of leakage detected) in accordance with IWA-5000 and IWB-5000, which provides reasonable assurance of component integrity.

Evaluation: The Code requires 100% examination of all RPV nozzle-to-vessel welds during each inspection interval. Where feasible, the volume shown in Figure IWB-2500-7 shall be examined (scanned) from both sides of the weld on the same surface. The licensee has provided documentation showing that for the outlet nozzles, the nozzle boss or protrusion completely obstructs scanning from one side of the weld such that only 43% of the volume can be examined from the vessel inside surface. Since the nozzle design configuration prevents full volumetric examination of these welds, the Code requirement is impractical. In order to gain access for 100% examination from both sides of the weld on the inside surface, the RPV outlet nozzles would need to be replaced with a modified design. This would place a significant burden on the licensee.

The licensee inspection results showed that essentially all of the required coverage (>98%) was obtained when the welds were inspected from the bore and approximately 43% of the weld was examined when inspected from the RPV inside surface. In addition, industry experience has not shown any history of integrity concerns associated with RPV nozzle-to-vessel welds nor have any unusual service loadings been identified which were not considered in the original design of these nozzle connections. Based on the volumetric examination coverage obtained, and the absence of any service degradation experience, reasonable assurance of the continued structural integrity of the RPV nozzle-to-vessel welds has been provided. Therefore, it is recommended that relief be granted pursuant to 10CFR50.55a(g)(6)(i).

2.3 Request for Relief RR 2-Closeout-3, Revision 0, Code Category B-D, Item B3.140, Steam Generator Nozzle Inside Radius Sections

Code Requirement: Examination Category B-D, Item B3.140, requires essentially 100% volumetric, as defined by Figure IWB-2500-7, of all steam generator nozzle inside radius sections during each successive inspection interval (2$^{nd}$, 3$^{rd}$, and 4$^{th}$). "Essentially 100%," as clarified by ASME Code Case N-460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee's Code Relief Request: In accordance with the provisions of 10 CFR 50.55a(g)(5)(iii), the licensee has requested relief from the full Code-required extent of volumetric examination for steam generator nozzle inside radius sections numbers 31-1A, 31-1B, 32-1A, 32-1B, and 33-1A.

Licensee's Basis for Relief Request (as stated):

Complete inspection of the Code-required volume is not possible due to restrictions caused by support lugs on the nozzle configuration [sketch and data sheets provided as Enclosure 3 of the licensee's request for relief submittal]. These welds have been examined to the maximum extent practical in accordance with standard industry practices.

Licensee's Proposed Alternative Examinations (as stated):

No additional volumetric examinations will be performed. The components listed in this relief request have been examined to the maximum extent practical.

A visual examination (VT-2) has been performed in conjunction with the pressure testing on these components every refuel outage (with no evidence of leakage detected) in accordance with IWA-5000 and IWB-5000, which provides reasonable assurance of component integrity.

Evaluation: The Code requires 100% examination of all steam generator nozzle inside radius sections during each inspection interval. Where feasible, the volume in Figure IWB-2500-7, a through d, as applicable, shall be examined (scanned) from both sides of the weld on the same surface. The licensee has provided documentation showing that for steam generator hot leg and cold leg nozzle inside radius sections, support lugs limit access on the flat transition and blended radius such that only 50% of the examination volume can be scanned. Since the steam generator hot/cold leg support configuration precludes full volumetric examination of these areas, the Code requirement is impractical. In order to gain access for 100% examination of the areas from the outside surface, the steam generator and adjacent reactor coolant system hot and cold leg integral support lugs would need to be replaced with a modified design. This would place a significant burden on the licensee.

The licensee inspection results showed that approximately 50% of each steam generator inside radius section was examined when inspected from the outside surface. In addition, industry experience has not shown any history of integrity concerns associated with these welds nor have any unusual service loadings been identified which were not considered in the original design of these nozzle connections. Based on the volumetric examination coverage obtained, and the absence of any service degradation experience, reasonable assurance of the continued structural integrity of the steam generator primary nozzle inside radius sections has been provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.4 Request for Relief RR2-Closeout-4, Rev.0, Examination Category B-J, Item B9.31, Pressure Retaining Welds in Piping

Code Requirement: Examination Category B-J, Item B9.31, requires volumetric and surface examinations, as defined by Figures IWB-2500-9, -10, or -11, for essentially 100% of the weld length of Class 1 piping branch connection welds to be performed during each inspection interval. “Essentially 100%,” as clarified by ASME Code Case N-460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from the Code volumetric examination requirement for the following three (3) branch connection welds:

| Weld ID | System |
|---|---|
| 16BC | Accumulator Discharge |
| 33BC | Reactor Coolant |
| 13BC | Accumulator Discharge |

Licensee's Basis for Relief Request: (as stated)

Complete inspection of the Code-required volume for the subject components was not possible due to restricted access caused by various design and geometry limitations.

Approximately two thirds (2/3) of the required examinations were performed during the 1st and 2nd Periods of the Interval (from 1989 to 1992) and do not have the detailed measurements necessary to estimate the percentage of examination coverage. This was standard practice in the industry at that time. The examinations were performed to the maximum extent possible using the available technologies. The type and location of limitations are also recorded on the data sheets, as approved for other B-J welds referenced by previous relief requests. In most cases, the limitation is caused by the weld's proximity to other components such as a valve, integrally support, or an adjacent nozzle. When a one-sided exam or a limited exam is confirmed as a one-sided exam (no other restrictions) through a review of examination data and drawing verification, a 50% exam coverage credit is taken. If a limited exam is noted but not enough information exists to derive a percentage of coverage, it will be so noted.

Licensee's Proposed Alternative Examination: (as stated)

No additional volumetric examinations will be performed on these welds. The components listed in this relief request have been examined to the maximum extent practical.

A visual inspection (VT-2) was performed during the pressure testing which is conducted on these components every refuel outage (no leakage detected) in accordance with IWA-5000 and IWC-5000, which provides reasonable assurance of component integrity."

Evaluation: The 1983 Section XI Code through Summer 1983 Addenda for Examination Category B-J "Pressure Retaining Welds in Piping," Items B9.30 and B9.31 states that, for 4-inch nominal pipe size or greater branch pipe connections, both surface and volumetric examinations are required. Essentially 100% of the weld length shall be examined. The volumetric examination shall be performed on the lower 1/3 thickness of the weld volume up to and including the adjacent base material extending 1/4 thickness from the weld toe on either side of the weld.

The licensee estimated that a maximum 50% of the required weld examination coverage could be expected based on single-sided access for austenitic stainless steel weld material. A review of the drawings and ultrasonic reports submitted by the licensee shows that access would be limited to one side of the weld. Although actual support dimensions were not available, the documentation appears to support the licensee's examination coverage estimates and shows that the Code-required 100% volumetric examination is impractical to perform. In order to gain access for 100% coverage, significant support redesign and plant modifications would need to be made for each branch connection. These changes would not be practical.

In addition, industry experience has not shown any history of integrity concerns associated with these welds nor have any unusual service loadings been identified which were not considered in the original design of these connections. Based on the volumetric examination coverage obtained, and the absence of any service degradation experience, reasonable assurance of the continued structural integrity of the three (3) piping branch connection welds has been provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

2.5 Request for Relief RR2-Closeout-5, Rev. 1, Examination Category B-K, Item B10.30, Integral Attachments for Piping, Pumps, and Valves

Code Requirement: ASME Code Section XI, 1983 Edition through Summer 1983 Addenda Examination Category B-K, Item B10.30, requires surface or volumetric examinations as applicable be performed on all pump welded attachments. The Code requires 100% of the attachment weld areas defined in Figures IWB-2500-13, 2500 -14, or 2500-15, to be examined during each inspection interval.

Licensee's Code Relief Request: Relief is requested from the ASME Boiler and Pressure Vessel Code Section XI requirement to examine, by surface or volumetric examination, 100% of the weld length or volume. Also, relief is requested from examining all of the pumps. In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee has requested relief from the 100% surface examination requirement for the six reactor coolant pump (RCP) integrally welded support attachments (three on RCP #31 and three on RCP #32) shown in Table 5. In addition, the licensee has requested relief from performing surface or volumetric examinations of the six remaining RCP integrally welded attachments.

**Table 5 - ASME Category B-K-1**

| Weld ID | System / Component | Extent Examined | Limitation | Remarks |
|---|---|---|---|---|
| 31-1SC | Reactor Coolant | At least 75%[1] | Lower support structure limits exam on bottom of weld | 31 RCP |
| 31-2SC | Reactor Coolant | At least 75%[1] | Lower support structure limits exam on bottom of weld | 31 RCP |
| 31-3SC | Reactor Coolant | At least 75%[1] | Lower support structure limits exam on bottom of weld | 31 RCP |
| 32-1SC | Reactor Coolant | At least 75%[1] | Lower support structure limits exam on bottom of weld | 32 RCP |
| 32-2SC | Reactor Coolant | At least 75%[1] | Lower support structure limits exam on bottom of weld | 32 RCP |

| | | | | |
|---|---|---|---|---|
| 32-3SC | Reactor Coolant | At least 75%(1) | Lower support structure limits exam on bottom of weld | 32 RCP |
| Note: Surface examination was limited by the lower support structure. Approximately 15-inches of the lower portion of each weld is inaccessible. It is estimated that more than 75% of each weld was inspected. | | | | |

Licensee's Basis for Relief Request: (as stated)

During the second interval, Code Case N-509 was approved by the NRC as an acceptable alternative to the 1983 Section XI Code requirements at stated, which allows for a 10% sample inspection of the applicable integrally welded attachments under Category B-K-1. This minimum 10% sample requirement also has been adopted into later editions of the Section XI Code (1995 Code with the 1995 Addenda, Examination Category B-K, Note (5), and later editions). Under the 10% sample, only two integrally welded attachments would be required. Both the Code Case and later editions of the Section XI Code require examination of essentially 100% of the weld length or volume.

During the second interval, six integrally welded attachments (three each on the 31 and 32 Reactor Coolant Pumps) were examined using the liquid penetrant method. The examinations were performed, in accordance with ASME Section V, 1983 and 1986 Code requirements respectively, by qualified inspectors using approved procedures. Minor indications were reported, and had been evaluated and determined as acceptable. None of the inspections indicated any deformation of the attachments. Due to restrictions from the lower support structure, a portion of the Reactor Coolant Pump support weld (about 15" on each welded support) could not be examined by either volumetric or surface examination method. This restriction exists at all the integrally welded attachments on the RCPs. Based on the attached "Limitation To Examination" sketches, it is estimated about 75% of the length of each welded attachment was examined. The weld length examined, on an individual weld basis, did not meet the minimum 90% of the weld length examined requirement.

Examination was performed on six (6) integrally welded attachments for two pumps (RCPs 31and 32) with 75% of coverage. While this does not meet the 1983 Code requirements, the number of attachments examined (6 out of 12) at 75% exceeded the minimum of a 10% sample 100% examination of 2 attachments selected for examination as required in Code Case N-509 and the later editions of the Section XI Code (1995 Code with the 1995 Addenda, and later editions) which have been approved by the NRC. ENO believes this reduction in population sample meets the intent of having at least 10% of the population examined as required. The minor indications were evaluated and found to be acceptable. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), ENO believes the proposed alternative examination performed provides an acceptable level of quality and safety.

Licensee's Proposed Alternative Examination: (as stated)

Surface examinations were performed on 31 and 32 Reactor Coolant Pumps to the maximum extent possible. No additional volumetric examinations will be performed.

A visual inspection (VT-2) was performed in conjunction with the pressure testing conducted on these components every refuel outage (with no leakage detected) in accordance with IWA-5000 and IWB-5000, which provides reasonable assurance of component integrity.

Evaluation: The 1983 Section XI Code, Examination Category B-K-1 Item 30.1 requires that all welded pump attachments be examined for essentially 100% of the length or volume of each attachment. At Indian Point, Unit 3 (IP3) there are four reactor coolant pumps (RCPs) in this examination category, each with three integrally welded attachments. The 1983 Code therefore requires that all 12 integrally welded attachments be examined each inspection interval. For the attachment weld configurations at IP3, surface examinations are required; however, the licensee has the option of performing volumetric examinations in lieu of the Code required surface examinations. Therefore an examination strategy based only on surface examinations is consistent with the Code and acceptable.

During the second inspection interval the licensee examined 75% of the weld area for 6 of the 12 RCP integrally welded attachments. A review of the drawings and ultrasonic reports, including component limitations,[1] submitted by the licensee showed that RCP support and welded attachment design configuration prevents the examination, by either surface or volumetric methods, of a portion of the RCP support lugs. Although actual support dimensions were not available, documentation appears to support the licensee's estimate that a maximum of 75% of the weld area is accessible for surface examination and the Code-required 100% volumetric examination is impractical to perform. In order to gain access for 100% coverage, significant support redesign and plant modifications would need to be made for each RCP. These changes would place a significant burden on the licensee.

Subsequent to the start of the licensee's second inspection interval, the NRC approved ASME Code Case N-509, which reduced the total number of integrally welded attachment examinations under Category B-K-1 to a 10% sampling. Under this alternative inspection strategy, the licensee would only be required to examine essentially 100% of the weld surface for 2, rather then 12, of the RCP integrally welded attachments required by the 1983 Edition of the Code. The 10% sampling requirement in Code Case N-509 was adopted into the ASME Section XI Code 1995 Edition. The extent of weld material examined by the licensee exceeds the amount of weld material that would be examined if the current Code inspection strategies were invoked.

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1 Drawings and limitations submitted by licensee are not included in this report.

Westinghouse PWR service experience has not identified any significant integrity issues associated with these supports. The licensee reported that minor indications were noted during the examinations of the 6 RCP support attachment welds, but that these indications were found to be acceptable, and none of the examinations showed any degradation to the attachments.

Based on the impracticality of achieving the Code-required 100% weld surface coverage, and that the extent of the examinations performed by the licensee exceed current Code requirements, it is recommended that the licensee's request for relief, to use the existing surface coverage on the 6 RCP integrally welded attachments for all 12 RCP integrally welded attachments, be granted, pursuant to 10 CFR 50.55a(g)(6)(i).

2.6 Request for Relief RR2-Closeout-6, Rev. 0, Examination Category C-A, Items C1.10, C1.20 and C1.30, Pressure Retaining Welds in Pressure Vessels

Code Requirement: Examination Category C-A, Items C1.10, C1.20, and C1.30 require essentially 100% volumetric examination, as defined by Figures IWB-2500-1 and 2, for shell and head circumferential welds and tubesheet-to-shell welds in Class 2 pressure vessels. "Essentially 100%," as clarified by ASME Code Case N-460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing the full extent of volumetric examinations for the welds shown in Table 6.

**Table 6 - ASME Category C-A**

| Code Item | Weld ID | System or Component | Extent Examined | Limitation (as stated) |
|---|---|---|---|---|
| C1.10 | 32-3 | 32 Accumulator Tank | 50% | Weld crown, welded lug, nozzle, head-to-shell configuration and long seam |
| C1.10 | 2 | Seal Water Return Filter | 50% | Welded Support and lower head configuration |
| C1.20 | 9 | 31 Regen Heat Exchanger | 50% | 2 adjacent nozzles |
| C1.20 | 2 | Volume Control Tank | 50% | Head configuration, weld crown, manway, nozzle |
| C1.20 | 32-4 | 32 Accumulator Tank | 50% | Weld crown, welded lug, nozzle, head-to-shell configuration and long seam |
| C1.20 | 1 | Seal Water Return Filter | 50% | Weld crown and flange configuration |

| | | | | |
|---|---|---|---|---|
| C1.20 | 3 | Seal Water Return Filter | 50% | Weld tag and flange configuration |
| C1.30 | 31-2 | 31 Steam Generator | 50% | 1 handhole and 2 nozzles |
| C1.30 | 10 | 31 Regen Heat Exchanger | 50% | 2 adjacent nozzles and support clamp |
| C1.30 | 11 | 31 Regen Heat Exchanger | 92% - 45°<br>93% - 2 scan<br>73% - 5 scan | 1 adjacent nozzle and support clamp |

Licensee's Basis for Relief Request (as stated):

Complete inspection of the Code-required volume is not possible based on restricted access caused by interference due to the close proximity of the vessel flange bolting, nozzles and support clamps. Drawings and sketches which illustrate the restricted conditions encountered that limit examination coverage are included in Enclosure 6 *[not included in this report]*.

Licensee's Proposed Alternative Examination (as stated):

No additional volumetric examinations will be performed on these welds. The components listed in this relief request have been examined to the maximum extent practical.

A visual inspection (VT-2) was performed during the pressure testing which is conducted on this component during every refuel outage (no leakage detected) in accordance with IWA-5000 and IWC-5000, which provides reasonable assurance of component integrity.

Evaluation: The Code requires 100% volumetric examination of the Class 2 vessel welds shown in Table 6 during each inspection interval. However, geometrical configurations, nozzle locations, vessel supports and other interference sources cause access restrictions that preclude ultrasonic scans of the full volume of these welds. Therefore, the Code-required 100% volumetric examination is impractical to perform. To gain access for 100% coverage, these components would have to be redesigned and modified. This would place a significant burden on the licensee.

A review of the drawings and ultrasonic reports, including component limitations,[2] submitted by the licensee indicate that these components have been examined to the maximum extent practical, with the licensee acquiring at least 50% volumetric coverage(s) for the subject welds. In addition, the welds are visual VT-2 examined during system leakage and hydrostatic tests, as required by the Code. Based on the volumetric coverage(s) obtained and the VT-2 examinations performed during system pressure tests, reasonable assurance of the detection of any existing patterns of

2 Drawings and limitations submitted by licensee are not included in this report.

degradation has been provided. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that the licensee's request be granted.

## 3.0 CONCLUSIONS

The PNNL staff has reviewed the licensee's submittal and concludes that reasonable assurance of the structural integrity of the subject components has been provided by the examinations that were performed. For Request for Relief RR2-Closeout 1, Rev. 1, it is recommended that the licensee's proposed alternatives be authorized pursuant to 10 CFR 50.55a(g)(6)(ii)(A)(5). For Requests for Relief RR2-Closeout 2, Rev.0, RR2-Closeout 3, Rev. 0, RR2-Closeout 4, Rev. 0, RR2-Closeout 5, Rev.1 and RR2-Closeout 6, Rev. 0, it is concluded that the Code examination coverage requirements are impractical for the subject welds. Therefore, for these requests, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).