February 18, 2005

Mr. Craig W. Lambert
Site Vice President
Kewaunee Nuclear Power Plant
Nuclear Management Company, LLC
N490 Highway 42
Kewaunee, WI 54216-9511

SUBJECT: KEWAUNEE NUCLEAR POWER PLANT - FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL PROGRAM REQUESTS FOR RELIEF (TAC NOS. MC2497, MC2498, MC2499, MC2500, MC2501, MC2503, MC2504, MC2505, MC2506, MC2507 AND MC2509)

Dear Mr. Lambert:

By letter dated December 16, 2003 (ML033580734), as supplemented September 17 (ML042720366) and September 30, 2004 (ML042890373), Nuclear Management Company, LLC (the licensee) submitted requests for relief from certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code), Section XI, for the fourth 10-year interval inservice inspection (ISI) program at Kewaunee Nuclear Power Plant (KNPP). The ASME Code Section XI of record for KNPP for the fourth 10-year ISI interval is the 1998 Edition with 2000 Addenda.

Based on the information provided in the relief requests, the U. S. Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), concluded that the following requests for relief were acceptable: RR-1-1, RR-1-2, RR-1-3, RR-1-4, RR-1-5, RR-1-7, RR-1-8, RR-2-1, and RR-G-2.

Relief Requests RR-1-1, RR-1-2, RR-1-4, and RR-2-1 may be granted on the basis that the staff concludes that the alternatives proposed by the licensee provide reasonable assurance of structural integrity and that complying with the specified Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternatives are authorized for the fourth 10-year ISI interval at Kewaunee, which ends on June 16, 2014.

Relief Requests RR-1-3 and RR-1-5 may be granted on the basis that the staff concludes that it is impractical for the licensee to comply with the subject ASME Code requirements, the proposed alternative inspections provide reasonable assurance of structural integrity, and that granting 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. Therefore, the subject reliefs are granted for the fourth 10-year ISI interval at Kewaunee, which ends on June 16, 2014.

Relief Requests RR-1-7 and RR-1-8 may be granted on the basis that the staff concludes that the alternative proposed by the licensee provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the fourth 10-year ISI interval at Kewaunee, which ends on June 16, 2014. The licensee withdrew Relief Request RR-1-9 by supplemental letter dated September 17, 2004. Also, the staff determined that Relief Request RR-1-10 was no longer required. The recent amendment of 10 CFR 50.55a as noticed in the *Federal Register* on October 1, 2004 (69 FR 58804), which became effective on November 1, 2004, included a modification to paragraph 50.55a(b)(2)(xv)(C)(*1*). The staff finds the licensee's proposed alternative in RR-1-10 is the same as the requirement in 10 CFR 50.55a(b)(2)(xv)(C)(*1*) and, therefore, relief is no longer required.

All other requirements of the ASME Code Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Two additional relief requests submitted in the letter dated December 16, 2003, are being reviewed separately by the NRC staff and will be addressed via separate correspondence. RR-1-6 relates to limited examination coverage of examinations selected in the risk-informed ISI program and RR-G-1 requests the implementation of a risk-informed ISI program.

The detailed results of the staff's review are provided in the safety evaluation in Enclosure 1. Enclosure 2 is the PNNL Technical Letter Report. Enclosure 3 is a table that provides a summary and the status of approval for the relief requests. If you have any questions concerning this matter, please call Mr. F. Lyon of my staff at (301) 415-2296.

Sincerely,

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-305

Enclosures: 1. Safety Evaluation
2. PNNL Technical Letter Report
3. Summary Table

cc w/encl: See next page

Relief Requests RR-1-7 and RR-1-8 may be granted on the basis that the staff concludes that the alternative proposed by the licensee provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the fourth 10-year ISI interval at Kewaunee, which ends on June 16, 2014. The licensee withdrew Relief Request RR-1-9 by supplemental letter dated September 17, 2004. Also, the staff determined that Relief Request RR-1-10 was no longer required. The recent amendment of 10 CFR 50.55a as noticed in the *Federal Register* on October 1, 2004 (69 FR 58804), which became effective on November 1, 2004, included a modification to paragraph 50.55a(b)(2)(xv)(C)(*1*). The staff finds the licensee's proposed alternative in RR-1-10 is the same as the requirement in 10 CFR 50.55a(b)(2)(xv)(C)(*1*) and, therefore, relief is no longer required.

All other requirements of the ASME Code Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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Docket No. 50-305

Enclosures: 1. Safety Evaluation
2. PNNL Technical Letter Report
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cc w/encl: See next page

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**Previously concurred

Accession Number: **ML050350225** ***NLO w/comments *SE dated 1/25/05

OFFICE	PM:PDIII-1	LA:PDIII-1	SC:EMCB	OGC	SC:PDIII-1
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DATE	02/17/05	2/9/05	1/25/05	2/16/05	02/18 /05

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM

REQUESTS FOR RELIEF

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

1.0 INTRODUCTION

The U. S. Nuclear Regulatory Commission (NRC) staff, with technical assistance from its contractor, the Pacific Northwest National Laboratory (PNNL), has reviewed and evaluated the information provided by the Nuclear Management Company, LLC (the licensee) in its letter dated December 16, 2003, as supplemented by letters dated September 17 and September 30, 2004, requesting relief from certain requirements of the American Society of Mechanical Engineers (ASME) *Boiler and Pressure Vessel Code* (Code) for the Kewaunee Nuclear Power Plant (KNPP). The requests for relief are identified as follows: RR-1-1, RR-1-2, RR-1-3, RR-1-4, RR-1-5, RR-1-7, RR-1-8, RR-1-9, RR-1-10, RR-2-1, and RR-G-2.

By its supplemental letter dated September 17, 2004, the licensee withdrew RR-1-9. The licensee's proposed alternative in RR-1-10 is the same as the current requirement in 10 CFR 50.55a(b)(2)(xv)(C)(*1*) and, therefore, relief is not required.

Two additional relief requests submitted in the letter dated December 16, 2003, are being reviewed separately by the NRC staff and will be addressed via separate correspondence. RR-1-6 relates to limited examination coverage of examinations selected in the risk-informed inservice inspection (ISI) program and RR-G-1 requests the implementation of a risk-informed ISI program.

2.0 REGULATORY REQUIREMENTS

ISI of 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 Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). In addition, 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 alternative 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 preservice 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 interval, subject to the limitations and modifications listed therein. The applicable Code of record for the fourth 10-year ISI interval for KNPP is the 1998 Edition of ASME Code Section XI, through the 2000 Addenda.

3.0 TECHNICAL EVALUATION

The NRC staff adopts the evaluations and recommendations for granting or authorizing reliefs as contained in the Technical Letter Report (TLR) prepared by PNNL, included as Enclosure 2. Enclosure 3 is a summary table listing each relief request by ASME Code examination category and the status of approval.

For Relief Request RR-1-1, the licensee requested to use a VT-2 visual examination during pressure tests in lieu of the 100 percent volumetric examination required once each interval of the inner radius section of the integrally-cast pressurizer surge nozzle. The licensee provided justification of the hardship to gain access to the inner radius section of the nozzle and the difficulties in inspecting the cast material. This type of nozzle does not have a shell-to-nozzle weld. Industry experience has shown no history of degradation or failures for integrally-cast pressurizer surge nozzle inner radius sections. The staff determined that the licensee's proposed alternative to perform a VT-2 visual examination during system pressure tests provides reasonable assurance of structural integrity. In addition, complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-2, the licensee proposed to conduct the system leakage tests for an 8-inch long section of nominal pipe size 2-inch, Schedule 160, stainless steel pipe between control valve CVC-15 and check valve CVC-16 that connects the auxiliary spray line to the normal pressurizer spray line in accordance with Code Paragraph IWB-5222(a). The subject piping would be included in the visual VT-2 examination, although the pressure of the subject segment of piping would be subject to the pressure it is exposed to during normal operation as opposed to the operating pressure of the reactor coolant system. The staff finds this approach is sufficient to ensure the leakage integrity of the subject piping segment. To require the licensee to open control valve CVC-15 and pressurize the subject 8-inch long pipe segment could cause an off-normal plant transient to occur and presents the licensee with an unusual difficulty with no compensating increase in quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-3, the staff determined that it is impractical to perform the Code-required surface examination of the reactor pressure vessel Welded Attachments RV-CS5 and RV-CS6 because of accessability. To gain access to perform surface examinations, extensive modifications to the bio-shield wall and reactor pressure vessel (RPV) insulation would be necessary. The licensee proposes to perform a remote ultrasonic examination of these attachments from the vessel inner surface in conjunction with the RPV shell weld examinations. In addition, the sand plugs (located above the safety injection nozzles) will be removed and allow limited access visual VT-3 examinations to be performed on the outside surface of the subject attachments. The staff finds that the alternative examinations proposed by the licensee provide reasonable assurance of continued structural integrity of the subject welds. Therefore, relief may be granted on the basis that the staff concludes that it is impractical for the licensee to comply with the subject ASME Code requirements and that granting 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. Therefore, the subject relief is granted for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-4, ASME Section XI, Examination Category B-P, Items B15.50 and B15.70, requires visual VT-2 examinations be performed in conjunction with system leakage tests for Class 1 piping and valves. The leakage tests must be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power. Portions of the residual heat removal system and safety injection system operate at pressures lower than the pressure corresponding to 100 percent rated reactor power. To require the licensee to pressurize the subject segments to the pressure corresponding to 100 percent rated reactor power could cause an off-normal plant transient to occur, over-pressurize certain components beyond design limits, or expose personnel to increased safety hazards, and presents the licensee with an unusual difficulty with no compensating increase in quality and safety. The staff finds that the alternative proposed by the licensee is sufficient to ensure the leakage integrity of the subject piping segments. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-5, ASME Section XI, Examination Category B-P, Item Number B15.50 requires that a visual VT-2 examination be performed in conjunction with a system leakage test for Class 1 piping. The system leakage tests must be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power. The licensee requested relief from the Code pressure requirement specified in IWB-5221(a), for the RPV flange leak-off piping. The subject reactor vessel flange leak-off lines are designed to only experience reactor coolant pressure if the O-ring seal on the RPV flange fails during power operations. There is no method to pressurize these small 3/4-inch outside diameter lines without removing the O-ring in the flange seal region, which would prevent the reactor coolant system from being pressurized. Therefore, the staff finds the leakage testing of the subject flange leak-off lines to normal reactor coolant pressure (2235 psig) is impractical. The licensee proposed to perform visual VT-2 examinations of this piping during normal system pressure tests at the end of each refueling outage. At this time, if any through-wall leakage has occurred, boric acid residue will be visible on the outside surface of the piping, providing a method to ensure that any potential leakage will be detected prior to compromising the intended function of the subject flange leak-

off lines. The staff finds that the alternative examinations proposed by the licensee provide reasonable assurance of continued structural integrity. Therefore, relief may be granted on the basis that the staff concludes that it is impractical for the licensee to comply with the subject ASME Code requirements and that granting 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. Therefore, the subject relief is granted for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request Number RR-1-7, the licensee proposed using Supplement 10, as administered by Electric Power Research Institute (EPRI)-Performance Demonstration Initiative (PDI) program, in lieu of the selected requirements of ASME Section XI, Appendix VIII, Supplement 10. The staff determined that the licensee's proposed alternative as administrated by EPRI-PDI will provide a comparatively challenging process for qualification in the detection and sizing of degradation in the subject components. Since the licensee's proposed alternative will provide an acceptable level quality and safety, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-8, the licensee proposed using a Supplement 2 add-on to a Supplement 10 qualification, as administrated by the EPRI-PDI program. The staff determined that the licensee's proposed alternative use of the EPRI-PDI administrated program in lieu of the selected requirements of ASME Section XI will provide a comparatively challenging process for qualification in the sizing and detection of flaws in the subject components. Therefore, the licensee's proposed alternative provides an acceptable level of quality and safety. Since the licensee's proposed alternative will provide an acceptable level quality and safety, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-1-9, ASME Section XI requires that examination volumes associated with RPV nozzle-to-vessel welds include the full weld thickness and base material adjacent to each side of the weld equal to one-half of the vessel shell thickness. The licensee proposed to implement the alternative found in Code Case –613-1, "Ultrasonic Examinations of Full Penetration Nozzles in Vessels," which provides a reduced examination volume for the subject welds. By letter dated September 17, 2004, the licensee withdrew RR-1-9.

For Relief Request RR-1-10, the licensee proposed to use the root-mean-square (RMS) values of 10 CFR 50.55a(b)(2)(xv)(C)(*1*), in lieu of the depth and length sizing criteria of ASME Code Section XI, Appendix VIII, Supplement 4, Subparagraphs 3.2(a), 3.2(b) and 3.2(c). The final rule, issued in the *Federal Register* on October 1, 2004 (69 FR 58804) and effective on November 1, 2004, states in paragraph 50.55a(b)(2)(xv)(C)(*1*):

> A depth sizing requirement of 0.15 inch RMS must be used in lieu of the requirements in Subparagraph 3.2(a) and 3.2(c), and a length sizing requirement of 0.75 inch RMS must be used in lieu of the requirements in Subparagraph 3.2(b).

Therefore, the licensee's proposed alternative is the same as the requirement in 10 CFR 50.55a(b)(2)(xv)(C)(*1*) and relief is no longer required.

For Relief Request RR-2-1, ASME Code Section XI, Examination Category C-H, Item Number C7.10 requires that a visual VT-2 examination be performed in conjunction with a system leakage test for Class 2 piping. The system leakage tests must be conducted at a pressure not less than the pressure corresponding to 100 percent rated reactor power (2235 psig). The subject reactor coolant vent system lines are small 1-inch and ½-inch outside diameter piping designed to vent non-condensible gases from the high points of the reactor coolant system. The licensee's technical specifications do not allow pressurizing the head vent lines above 200 degrees F using normal reactor coolant system pressure. Isolation valves are required to be closed when the reactor coolant system is at normal operating pressure and temperature. Therefore, the staff finds the leakage testing of the subject reactor vessel head vent lines to normal reactor coolant pressure (2235 psig) would present an unusual difficulty without a compensating increase in quality or safety. The licensee proposed to perform (1) visual VT-2 examinations of this piping each period at approximately 380 psig and less than 200 degrees F, and (2) visual VT-2 examination of the head vent lines during the Class 1 reactor coolant system pressure test each refueling outage. During the part 2 examinations, head vent lines downstream of the normally closed isolation valves will not be pressurized. However, these lines are normally filled with borated water during each refueling outage. If any through-wall leakage has occurred, boric acid residue will be visible on the outside surface of the piping, providing a method to ensure that any potential leakage will be detected prior to compromising the intended function of the head vent lines. The staff finds that the licensee's two-part alternative satisfies the intent of the Code requirement, and provides reasonable assurance that the structural adequacy of the subject head vent lines will be maintained. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

For Relief Request RR-G-2, the Code requires the removal of all insulation from pressure-retaining bolted connections in systems borated for the purpose of controlling reactivity when performing VT-2 visual examinations during system pressure tests. As an alternative, the licensee proposed a two-part approach. First, by performing the leakage test at operating pressure with the insulation in place, any significant leakage will be detected when the leakage either penetrates the insulation, or is detected at joints or low points of the pipe. Second, by removing the insulation each refueling outage for the subject components, the licensee will be able to detect minor leakage indicated by the presence of boric acid crystals or residue. This two-phase approach will provide reasonable assurance of the continued structural integrity of the subject bolted connections in borated systems. The staff finds the licensee's alternative provides a thorough approach to ensuring the leak tight integrity of systems borated for the purpose of controlling reactivity. Therefore, the licensee's proposed alternative provides an acceptable level of quality and safety. Since the licensee's proposed alternative will provide an acceptable level quality and safety, the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval at KNPP, which ends on June 16, 2014.

4.0 CONCLUSION

The staff adopts the evaluations and recommendations for granting reliefs and authorizing alternatives contained in the TLR prepared by PNNL, included as Enclosure 2, with the exception of RR-1-10. Due to the recent amendment of 10 CFR 50.55a as provided in the *Federal Register* on October 1, 2004 (69 FR 58804) that became effective on November 1, 2004, which included a modification to paragraph 50.55a(b)(2)(xv)(C)(*1*), the staff finds the

licensee's proposed alternative in RR-1-10 is the same as the requirement in 10 CFR 50.55a(b)(2)(xv)(C)(*1*) and, therefore, relief is no longer required. Enclosure 3 is a summary table listing each relief request and the status of approval.

For Relief Requests RR-1-1, RR-1-2, RR-1-4, and RR-2-1, the staff concludes that alternatives proposed provide reasonable assurance of structural integrity and that complying with the specified Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii) the proposed alternatives are authorized for the fourth 10-year ISI interval at Kewaunee.

For Relief Requests RR-1-3 and RR-1-5, the staff concludes that it is impractical for the licensee to comply with the subject ASME Code requirements, the proposed alternative inspections provide reasonable assurance of structural integrity, and that granting 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. Therefore, the subject reliefs are granted for the fourth 10-year ISI interval at KNPP.

For Relief Requests RR-1-7, RR-1-8 and RR-G-2, the staff concludes that the alternative proposed provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i) the licensee's proposed alternative is authorized for the fourth 10-year ISI interval at KNPP.

Relief Request Number RR-1-9 was withdrawn by letter dated September 17, 2004.

All other requirements of the ASME Code Section XI for which relief has not been specifically requested remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: A. Keim

Date: February 18, 2005

TECHNICAL LETTER REPORT

FOURTH 10-YEAR INSERVICE INSPECTION INTERVAL

REQUESTS FOR RELIEF

NUCLEAR MANAGEMENT COMPANY, LLC

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NUMBER 50-305

1.0 SCOPE

By letter dated December 16, 2003, the licensee, Nuclear Management Company, submitted several requests for relief from the 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 provided further information in a letter dated September 30, 2004. The requests are for the fourth 10-year inservice inspection (ISI) interval at Kewaunee Nuclear Power Plant (Kewaunee). Pacific Northwest National Laboratory (PNNL) has evaluated the requests for relief and supporting information submitted by the licensee in Section 3.0 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 the

Kewaunee fourth 10-year interval ISI program, which began on June 16, 2004, is the 1998 Edition of ASME Section XI, through the 2000 Addenda.

3.0 TECHNICAL EVALUATION

The information provided by the Nuclear Management Company in support of the requests for relief from Code requirements has been evaluated and the bases for disposition are documented below. For clarity, the request has been evaluated in multiple parts, according to ASME Code Examination Category.

3.1 Request for Relief RR-1-1, Examination Category B-D, Item B3.120, Full Penetration Welded Nozzles in Vessels

Code Requirement: Examination Category B-D, Item B3.120 requires 100% volumetric examination, as defined in Figures IWB-2500-7(a) through (d), as applicable, of Class 1 pressurizer (PZR) nozzle inner radius sections during each inspection interval. Code Case –460, as an alternative approved for use by the NRC Staff, states that a reduction in examination coverage due to part geometry or interference for any Class 1 and 2 weld is acceptable provided that the reduction is less than 10%, i.e., greater than 90% examination coverage is obtained.

Licensee's Code Relief Request: In accordance with 10CFR50.55a(g)(5)(iii), the licensee requested relief from the Code volumetric examination requirement for the Kewaunee PZR surge nozzle inner radius section identified by the licensee as ISI item P-IR7.

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

Ultrasonic examination of the pressurizer surge nozzle inner radius section is undesirable for the following reasons:

a. Coarse grain found in castings causes sound to be attenuated.
b. Difficult to differentiate flaws from normal geometry (clad roll).
c. Access restrictions caused by the pressurizer heater penetrations and associated wiring. Due to the complexity of work on and around the heater penetrations, there is a possibility of damaging this equipment and a potential to adversely impact the outage duration due to scheduling conflicts.
d. Difficulty in removal and replacement of insulation around the heater penetrations and wiring.
e. Increased personnel exposure to radiation and high cost of examination.
f. There is not a history of industry failures in this area.

Licensee's Proposed Alternative Examination (as stated):

The surge line (at the bottom of the pressurizer) is inaccessible for visual examination even when the manway (at the top of the pressurizer) is removed; therefore, no alternative examination on the surge nozzle can be performed.

The integrity of this nozzle will be verified during the Class 1 system leakage test which is performed after each refueling outage during startup as required by Table IWB-2500-1, Category B-P, Item B15.20.

Response to Request for Additional Information (as stated):

The KNPP pressurizer surge nozzle is integrally cast with the pressurizer; thus, there is no nozzle to vessel shell weld to be performed. In review of industry operability assessments, KNPP has found no examples of pressurizer nozzle inner radius failures.

Exposure in the general area of the pressurizer surge line nozzle inner radius at the bottom of the pressurizer is 50 mR to 300 mR. The estimated time for the examination of pressurizer nozzle inner radius, P-IR7, would include for the area at the bottom of the pressurizer:

Insulation removal:	3 men X 8 hours
Inner radius prep:	2 men X 1 hour
Ultrasonic examination:	2 men X 1 hour
Insulation replacement:	3 men X 10 hours
Total man-hours:	58 man-hours X 100mR = 5.8R

Evaluation: The Code requires a 100% volumetric examination of the inner radius sections of Class 1 vessel shell-to-nozzle welds and integrally cast nozzles to be performed once during each ISI interval. In order to examine the inner radius section for the pressurizer surge nozzle at Kewaunee, the licensee would have to carefully remove insulation and prepare the vessel outside surface for ultrasonic examination, while not damaging the wiring or connections to the pressurizer heaters. This places a significant hardship on the licensee, based on radiation exposure that would be incurred.

The pressurizer surge nozzle at Kewaunee is integrally cast into the bottom head so that no vessel-to-nozzle weld exists. The inner radius section is the transition zone from the nozzle cylinder to the bottom head bowl in this large casting. The inner radius ultrasonic examination on the pressurizer surge nozzle is difficult to perform due to large grain structures in the cast material, the complex geometry of this transition region, and irrelevant indications that may be reflected from the clad-to-base metal interface. No access to the inside surface is possible for a visual examination of this area of the pressurizer.

To gain access for examination from the outside surface, insulation on the pressurizer bottom head would have to be carefully removed and the outside surface would then have to be prepared by removing loosely adhering oxides or other residue to allow for ultrasonic transducer coupling. After examining the item, the insulation would have to be carefully replaced. Performing these activities could result in damage to the heater wiring and connections. The licensee estimates that a total radiation exposure of approximately 5.8 man-rem would be incurred during these activities. As stated above, there is no shell-to-nozzle weld at Kewaunee; the inner radius section examination is the only Code requirement for the pressurizer surge nozzle. Therefore, to require ultrasonic

examination of the surge nozzle inner radius section would place a considerable hardship on the licensee.

A brief review of industry experience has shown that no history of generic degradation or failures has been observed for integrally cast pressurizer surge nozzle inner radius sections. To require the licensee to remove the bottom head insulation and prepare the surface for examination, which could potentially damage heater wiring or connections, solely for the purpose of examining the surge nozzle inner radius section would cause an undue hardship with no compensating increase in quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Request for Relief RR-1-1 be authorized for the fourth interval at Kewaunee.

3.2 Request for Relief RR-1-2, Examination Category B-P, Item B15.50, All Pressure Retaining Components

Code Requirement: Table IWB-2500-1, Examination Category B-P, Item B15.50, requires a system leakage test be performed on piping systems prior to plant start-up during each refueling outage. Requirements for system leakage test boundaries, pressures, and temperatures are listed in paragraph IWB-5220.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing system leakage tests at normal reactor coolant pressure for a segment of PZR Auxiliary Spray piping located between valves CVC-15 and CVC-16.

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

Pressurizer pressure is maintained by the reactor coolant pumps via normal pressurizer spray. Normal pressurizer spray is controlled by the pressurizer pressure control system which automatically controls the pressurizer environment. The primary purpose of the auxiliary spray line is for pressure control when the reactor coolant pumps are not running (i.e., during a post accident condition when it is desired to decrease reactor coolant system pressure.) The use of the auxiliary spray line at hot standby or power may lead to an unnecessary plant transient. Implementing this Code requirement requires that the plant open valve CVC-15 to pressurize the subject pipe. Opening of valve CVC-15 at hot standby or power increases pressurizer spray which will cause an adverse reduction in reactor coolant system pressure.

Licensee's Proposed Alternative Examination (as stated):

Perform VT-2 visual examination during the Class 1 system leakage pressure test in accordance with requirements specified in Table IWB-2500-1, Examination Category B-P, Item Number B15.50, Note 2 IWB-5222(a). This requires the pressure retaining boundary to correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup.

Evaluation: The Code requires that licensees perform a system leakage test during each refueling outage, just prior to starting up the plant. These leakage tests must be conducted at a system pressure not less than the pressure corresponding to 100% rated

reactor power. For the subject portion of the pressurizer auxiliary spray line piping between valves CVC-15 and CVC-16, this corresponds to normal reactor coolant pressure (approximately 2235 psig). However, if auxiliary spray valve CVC-15 is opened while the reactor coolant system (RCS) is at normal pressure and temperature, cold water would be injected into the pressurizer unnecessarily, causing a drop in pressurization of the RCS. This could cause an off-normal plant transient and place a considerable hardship on the licensee.

Paragraph IWB-5222(a) of the Code states:

> *The pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.*

The subject piping segment at Kewaunee is an 8-inch long section of NPS 2-inch, Schedule 160, stainless steel pipe between motor control valve CVC-15 and check valve CVC-16 that connects the auxiliary spray line to the normal pressurizer spray line. At normal operating modes, the pressurizer spray line (downstream of CVC-16) is pressurized by the reactor coolant pumps. Piping in the auxiliary spray system (upstream of CVC-15) is pressurized by charging pumps. Thus, the 8-inch line segment between these valves would be pressurized if the auxiliary system is activated, which only occurs when it is necessary to decrease RCS pressure. To require the licensee to activate auxiliary spray during normal startup operations with the RCS at temperature and pressure could cause the plant to shutdown and challenge important safety systems unnecessarily. This would present an unusual difficulty for the licensee.

The licensee has proposed to conduct the system leakage tests for this piping segment in accordance with Code Paragraph IWB-5222(a), as stated above. The subject piping segment would be included in the visual VT-2 examination, although the ambient pressure would be as found during normal operation of the RCS. This approach should be sufficient to ensure the leakage integrity of the subject piping segment, and meets the intent of the Code requirements.

To require the licensee to open control valve CVC-15 and pressurize the subject 8-inch long pipe segment, which could potentially cause an off-normal plant transient to occur, presents the licensee with an unusual difficulty with no compensating increase in quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Request for Relief RR-1-2 be authorized for the fourth interval at Kewaunee.

3.3 Request for Relief RR-1-3, Examination Category B-K, Item B10.10, Welded Attachments for Vessels, Piping, Pumps, and Valves

Code Requirement: Examination Category B-K, Item B10.10, requires “essentially 100%” surface examination, as defined by Figures IWB-2500-13, -14, and -15, as applicable, of the length of each weld for Class 1 integrally welded attachments to pressure vessels. “Essentially 100%,” as clarified by ASME Code Case –460, is greater than 90% coverage of the examination volume, or surface area, as applicable.

Licensee's Code Relief Request: In accordance with 10CFR50.55a(g)(5)(iii), the licensee requested relief from the Code surface examination requirement for reactor pressure vessel (RPV) Welded Attachments RV-CS5 and RV-CS6.

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

Surface examination of the Reactor Vessel Welded Attachments cannot be performed due to restricted access. RV-CS5 and RV-CS6 are located on the O.D. of the Reactor Vessel and between the Reactor Vessel and Biological Shield. Restricted area prohibits removal of permanent, Reactor Vessel insulation and inability to properly clean welds for surface examination.

Licensee's Proposed Alternative Examination:

Perform ultrasonic examination of Welded Attachments RV-CS5 and RV-CS6 from the Reactor Vessel I.D. using remotely operated automated equipment. Perform examination at the end of the interval when core barrel is removed for remainder of Reactor Vessel shell circumferential welds. Perform VT-3 Visual examination from Reactor O.D. of accessible areas of RV-CS5 and RV-CS6.

Response to Request for Additional Information (as stated):

The reactor vessel welded attachments are RV-CS5 located at 88.5° and RV-CS6 located at 268.5°. The KNPP configuration is similar to ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, 2000 Addenda, Figure IWB-2500-15. Access to the six reactor vessel nozzle safe-end welds is provided by removal of the sand plugs (concrete blocks) located directly above each nozzle to safe-end. However, no access is directly available to RV-CS5 and RV-CS6, which are restricted by the reactor vessel insulation and bioshield. A partial visual examination can be performed on RV-CS5 and RV-CS6 due to sand plug removal for the two to four inch reactor vessel safe-ends.

The ultrasonic technique utilized will be EPRI's Performance Demonstration Initiative (PDI) qualified method for examination of reactor vessel shell welds for Appendix VIII, Supplement 4 and Supplement 6 performed by WesDyne International. The scan volume will be extended to include the integrally welded attachment documented in the WesDyne International scan program for the reactor vessel. Scanning performed will utilize a 45° shear angle beam from the ID of the reactor vessel. The WesDyne International performance of ultrasonic examinations from the reactor vessel would be expected to locate fabrication type defects in the weld attached to the reactor vessel. Additionally, cracking from the toe of the weld of the integrally welded attachment into the base metal of the reactor vessel could also be detected. Defects would be located and sized based on a C scan presentation of the reflectors in the U-shaped configuration of the integrally welded attachment. During the 1995 refueling outage, ultrasonic examinations were performed by WesDyne International of integrally welded attachments, RV-CS5 and RV-CS6 from the ID of the reactor vessel. Examinations revealed no recordable indications.

Evaluation: The Code requires that surface examinations of integrally welded attachments to the reactor pressure vessel (RPV) be performed once during each ISI interval. Welded Attachments RV-CS5 and RV-CS6 provide primary support for the RPV and consist of plate assemblies integrally welded to the vessel external surface. These attachments cannot be accessed for surface examination because of their location in the annulus region that exists between the radiological bio-shield wall and the RPV outside surface. To gain access to perform surface examinations, extensive modifications to the bio-shield wall and RPV insulation would be necessary. Therefore, the Code-required surface examinations are impractical at Kewaunee.

Welded Attachments RV-CS5 and RV-CS6 are located on the outside surface of the RPV upper shell near the safety injection nozzles. Access to perform surface examinations of these integral attachment welds is not possible due to the limited space available between the permanent RPV insulation and radiological bio-shield wall. This small annulus region does not permit removal of insulation and preparation of the RPV outside surface for the required examinations. In lieu of surface examinations on the integral attachment welds, the licensee has proposed to perform a remote ultrasonic examination of these attachments from the vessel inner surface in conjunction with the RPV shell weld examinations. A 45 degree shear wave examination will be performed to detect any flaws that may have originated in the attachment welds and progressed into the RPV shell. In addition, the sand plugs (located above the safety injection nozzles) will be removed and will allow a limited access visual VT-3 examination to be performed on the outside surface of these attachments. The alternative examinations proposed by the licensee should provide reasonable assurance of the continued structural integrity of these welds.

Therefore, based on the impracticality of performing the Code-required surface examinations, and considering the alternative examinations proposed by the licensee, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that Request for Relief RR-1-3 be granted for the fourth interval at Kewaunee.

3.4 Request for Relief RR-1-4, Examination Category B-P, Items B15.50 and B15.70, All Pressure Retaining Components

Code Requirement: Examination Category B-P, Items B15.50 and B15.70, require that visual VT-2 examinations be performed in conjunction with system leakage tests for Class 1 piping and valves, respectively. The system leakage tests must be performed during each refueling outage, and meet the system pressure and test boundary conditions specified in Paragraph IWB-5220. Specifically, IWB-5221(a) states that system leakage tests must be conducted at a pressure not less that the pressure corresponding to 100% rated reactor power.

Licensee's Code Relief Request: In accordance with 10CFR50.55a(g)(5)(iii), the licensee requested relief from the Code pressure requirement specified in IWB-5221(a), for the piping and valves segments shown in Table 3.4 below.

Table 3.4 Class 1 Piping and Valves Included in RR-1-4	
Item	**Description**
A	8-inch and 3/4-inch piping in the residual heat removal (RHR) system between valves RHR-1A and RHR-2A, up to and including valves RHR-1A, RHR-2A, RHR-30A, RHR-31A, RHR-32A, RHR-32A-1 and rupture disc
B	8-inch and 3/4-inch piping in the residual heat removal (RHR) system between valves RHR-1B and RHR-2B, up to and including valves RHR-1B, RHR-2B, RHR-30B, RHR-31B, RHR-32B, RHR-30B-1 and rupture disc
C	12-inch and 3/4-inch piping in the safety injection (SI) system between valves SI-21A and SI-22A, up to and including valves SI-21A, SI-22A, SI-44A, SI-45A, and SI-201A
D	12-inch, 10-inch, and 3/4-inch piping in the safety injection (SI) system between valves SI-21B and SI-22B, up to and including valves SI-21B, SI-22B, RHR-11, SI-44B, SI-45B, and SI-201B
E	6-inch, 2-inch, and 3/4-inch piping in the SI system between valves SI-12A and SI-13A, up to and including valves SI-12A, SI-13A, and SI-42
F	6-inch, 2-inch, and 3/4-inch piping in the SI system between valves SI-12B and SI-13B, up to and including valves SI-12A, SI-13A, and SI-62
G	6-inch, 2-inch, and 3/4-inch piping in the SI system between valves SI-303A and SI-304A up to and including SI-303A, SI-304A, SI-16A, SI-46 and SI-48
H	6-inch, 2-inch, and 3/4-inch piping in the SI system between valves SI-303B and SI-304B up to and including SI-303B, SI-304B, SI-16B, SI-47, SI-49 and SI-50

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

The affected components listed above consist of piping that is either located between two (2) shut valves, located between two (2) check valves and/or classified as parts of systems not required to operate during normal plant operation. This piping is operated at a pressure lower than the nominal operating pressure associated with 100% rated reactor power. The piping and valves including operating pressure are as follows:

Items A and B: Train A and Train B of Residual Heat Removal (RHR) Inlet Piping

Items A and B have the following characteristics:

Section XI Required System Leakage Test Pressure:	2235 psig
Operating Pressure:	450 psig
Proposed System Leakage Test Pressure:	450 psig

Both trains consist of two motor-operated valves in series and are located off the hot legs of the RCS loops. These trains are the inlet piping to the RHR system that are used for cooling the core during plant shutdown, refueling and startup. At 100% rated reactor power, this piping can be pressurized to RCS pressure by either of the following methods:

- The interlocks associated with valves RHR-1A(B) and RHR-2A (B) could be modified permit pressurization from the RCS. Overriding the interlocks

  associated with RHR-1A and RHR-2A (RHR-1B and RHR-2B) to pressurize the piping between these valves could result in challenging the piping on the downstream side of valves RHR-2A(B). This piping is classified as Section XI code Class 2 and designed for 600 psig. This method could result in reducing the margin of safety of the plant since failure of either valve RHR-2A(B) would result in an inter-system LOCA outside of containment.
- A hydrostatic pressure pump could be used to pressurize the piping between these two motor-operated valves through an existing drain valve. Use of a hydrostatic pressure pump in this application poses the possibility of overpressurizing the downstream Class 2 piping due to leakage or failure of RHR-2A or RHR-2B.

Items C and D: Train A and B of Accumulator Injection Piping

Items C and D have the following characteristics:
Section XI-Required System Leakage Test Pressure: 2235 psig
Operating Pressure: 2200 psig at SI pump discharge
Proposed System Leakage Test Pressure: 2200 psig at SI pump discharge

This piping is located at the discharge of the SI accumulator tanks and is maintained at approximately 750 psig when the plant is operating at 100% rated reactor power. At 100% rated reactor power, this piping can be pressurized to RCS pressure by either of the following methods:

- The piping configuration would require the installation of jumpers to existing drain valves located between check valves SI-21A&B and SI-22A&B to pressurize the piping from the RCS.
- Installation and use of a hydrostatic pressure pump.

Items E and F: Train A and B High Pressure SI Piping

Items E and F have the following characteristics:
Section XI-Required System Leakage Test Pressure: 2235 psig
Operating Pressure: 2200 psig at SI pump discharge
Proposed System Leakage Test Pressure: 2200 psig at SI pump discharge

This piping is connected to the cold legs of the RCS loops. This piping provides SI fluid to the core under high-pressure conditions following an accident. At 100% rated reactor power, this piping can be pressurized to RCS pressure by either of the following two methods:

- Installation of jumpers to the drain valves located between the check valves.
- Installation of a hydrostatic pressure pump.

Items G and H: Train A and B SI to Reactor Vessel

Items G and H have the following characteristics:
Section XI-Required System Leakage Test Pressure: 2235 psig

Operating Pressure:	2200 psig at SI pump discharge
Proposed System Leakage Test Pressure:	2200 psig at SI pump discharge

This piping is connected to the SI nozzles attached to the reactor vessel. At 100% rated reactor power, this piping can be pressurized to RCS pressure by either of the following two methods:

- Installation of jumpers to existing drain valves located between the check valves.
- Installation of a hydrostatic pressure pump.

A hydrostatic pressure pump could be used to pressurize each of these segments of piping through an existing drain valve. When a hydrostatic pump is used as a pressure source, the affected system is not available to perform its intended safety function during the period of time it has been declared inoperable to conduct the test. Although hydrostatic pressure testing is performed with the utmost of care using detailed procedures and trained personnel, there is a small possibility of equipment damage or human error. Hydrostatic pressure testing also delays availability of the system by several shifts to establish test conditions, perform the test and recover from testing.

The use of a hydrostatic pressure pump poses various operational challenges depending on the plant mode when testing is performed. The testing poses operational concerns and personnel and plant safety issues because the plant is placed in a configuration requiring an operating pressure greater than normal operating pressure for either hydrostatic or system pressure testing. Connecting the RCS to the SI system and RHR system through the use of jumpers poses similar challenges.

Licensee's Proposed Alternative Examination (as stated):

Perform the Code-required VT-2 visual examinations of the affected components at the normal operating pressure of each of the systems, as discussed below:

Items A and B: Train A and Train B RHR Inlet Piping

Since this piping is within the RCS test boundary, it is VT-2 visually inspected following each refueling outage when the plant is in hot shutdown. Although the motor-operated valves are shut at this time, the piping is pressurized from operation of the RHR system. This section of piping is also VT-2 visually examined as part of the Class 2 RHR system once during each inspection period (every 40 months). A test pressure of 450 psig (pump discharge pressure) is used for testing the RHR system. During refueling shutdown, except when fuel is removed from the reactor vessel, the RHR system is in continuous operation at pressures that vary between approximately 450 psig and atmospheric pressure. At this time, the integrity of RHR system is verified via available instrumentation and personnel observations. The combination of plant monitoring equipment such as leak detection systems and increased maintenance and surveillance activities provides a high degree of confidence that through-wall leakage would be detected and corrected.

The alternative test pressure of 450 psig fulfils the same purpose as the test pressure required by Paragraph IWB-5221 in that it accomplishes a check for component leakage at a reduced cost while enhancing plant safety. Plant safety is enhanced when pressure

testing is performed at the normal operating pressure of 450 psig because the affected system is available to perform its intended safety function during testing, the possibility of challenging the pressure integrity of the downstream Class 2 piping is reduced, the possibility of damage to pipe connections is eliminated if a hydrostatic pressure pump need not be installed.

Items C and D: Train A and B Accumulator Injection Piping

This section of piping is pressurized to approximately 750 psig and VT -2 visually inspected as part of the RCS following each refueling outage when the plant is in hot shutdown. This section of piping is also VT-2 visually examined as part of the SI system at or near the end of the inspection interval to satisfy the hydrostatic pressure test requirement. A test pressure of approximately 2200 psig (pump discharge pressure) is used to test the SI system.

The alternative test pressure of 2200 psig at the SI pump discharge fulfills the same purpose as the test pressure required by Paragraph IWB-5221 because a check for component leakage is performed at a reduced cost while enhancing plant safety. Plant safety is enhanced when pressure testing is performed at the normal operating pressure of approximately 2200 psig (pump discharge pressure). The affected system is available to perform its intended safety function during testing, the probability of challenging the pressure integrity of an affected component or causing an inadvertent actuation of a safety/relief valve or safety feature is reduced, and the possibility of damage to pipe connections is eliminated that could cause system leakage or valve inoperability.

Items E and F: Train A and B High Pressure Safety Injection Piping and Items G and H: Train A and B Safety Injection to Reactor Vessel

Since this piping is within the RCS test boundary, it is VT-2 visually inspected following each refue1ing outage when the plant is in hot shutdown. This section of piping is also VT -2 visually examined as part of the SI system at or near the end of the inspection interval to satisfy the hydrostatic pressure test requirement. A test pressure of approximately 2200 psig (pump discharge pressure) is used for testing the SI system.

The alternative test pressure of 2200 psig at the SI pump discharge fulfills the same purpose as the test pressure required by Paragraph IWB-5221 in that a check for component leakage is accomplished at a reduced cost while plant safety is enhanced. Plant safety is enhanced when pressure testing is performed at the normal operating pressure. The affected system is available to perform its intended safety function during testing; the possibility of challenging the pressure integrity of an affected component or causing an inadvertent actuation of a safety/relief valve or safety feature is reduced; and the possibility of damage to pipe connections is eliminated that could cause system leakage or valve inoperability.

Evaluation: The Code requires that licensees perform a system leakage test during each refueling outage, just prior to starting up the plant. These leakage tests must be conducted at a system pressure not less than the pressure corresponding to 100% rated reactor power. In addition, Paragraph IWB-5222(a) of the Code states:

The pressure retaining boundary during the system leakage test shall correspond to the reactor coolant boundary, with all valves in the position required for normal reactor operation startup. The visual examination shall, however, extend to and include the second closed valve at the boundary extremity.

The piping segments listed in Table 3.4 above do not normally experience full reactor coolant system (RCS) pressure. In fact, the subject piping segments and valves are only designated as Class 1 because they are between the first and second isolation valves; beyond the second isolation valve the systems change to Class 2 or 3. In order to pressurize these piping segments to RCS pressure, the licensee would have to defeat the first isolation valve via hose jumpers or use a hydrostatic pump. These approaches may challenge the safety function of these systems, over-pressurize downstream piping beyond design loads, or subject personnel to safety hazards. Thus, pressurization of the subject piping segments to RCS pressure would require the licensee to incur a considerable hardship.

The licensee has proposed alternatives to the Code pressure requirements for the various piping system segments identified in Table 3.4. For segments in Train A and B of the Residual Heat Removal (RHR) system, the piping will be pressurized to 450 psig during Class 2 pressure testing every 40 months. This corresponds to the maximum pressure that will be experienced for these piping segments. In addition, the RHR system is in continuous operation during plant shutdown (except for fuel movement), and the leakage integrity of the system is verified by plant instrumentation and personnel observations. Of course, the RHR Class 1 segments are also visual VT-2 inspected during the leakage tests of the RCS system at each refueling outage, but at ambient pressure and temperature.

For segments in Trains A and B of the Accumulator Injection piping, the piping is pressurized to approximately 750 psig and visual VT-2 examined each refueling outage. In addition, these piping segments are visual VT-2 examined at or near the end of each inspection interval with the safety injection pumps (2200 psig pump discharge pressure) running.

For piping segments in Trains A and B of the Safety Injection Piping and Trains A and B of the Safety Injection to Reactor Vessel, a visual VT-2 is performed at hot shutdown during each refueling outage, but with these isolated segments at normal pressure. In addition, these piping segments are visual VT-2 examined at or near the end of each inspection interval with the safety injection pumps (2200 psig pump discharge pressure) running.

The alternatives described above should be sufficient to ensure the leakage integrity of the subject piping segments, and meet the intent of the Code requirements for these isolated portions of the piping.

To require the licensee to pressurize the subject segments to RCS pressure, which could potentially cause an off-normal plant transient to occur, over-pressurize certain components beyond design limits, or expose personnel to increased safety hazards, presents the licensee with an unusual difficulty with no compensating increase in quality

and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that Request for Relief RR-1-4 be authorized for the fourth interval at Kewaunee.

3.5 Request for Relief RR-1-5, Examination Category B-P, Item 15.50, All Pressure Retaining Components

Code Requirement: Examination Category B-P, Item B15.50, requires that a visual VT-2 examination be performed in conjunction with a system leakage test for Class 1 piping. The system leakage tests must be performed during each refueling outage, and meet the system pressure and test boundary conditions specified in Paragraph IWB-5220. Specifically, IWB-5221(a) states that system leakage tests must be conducted at a pressure not less than the pressure corresponding to 100% rated reactor power.

Licensee's Code Relief Request: In accordance with 10CFR50.55a(g)(5)(iii), the licensee requested relief from the Code pressure requirement specified in IWB-5221(a), for reactor pressure vessel (RPV) flange leak-off piping.

Basis for Requesting Relief (as stated):

The reactor flange leak-off lines are not pressurized to 2235 psig when the RCS is operated at 100% power. The design of the reactor flange leak-off lines does not allow for pressurization using reactor coolant as a pressurizing medium. The purpose of the reactor vessel O-rings is to provide a seal between the reactor vessel and head. The reactor vessel flange leak-off lines would only experience a pressure of 2235 psig if the reactor O-rings leaked. These lines are classified as parts of the system not required to operate during normal plant operation. The lines normally see a pressure of approximately 50 psig when the reactor O-rings are removed and the reactor cavity is flooded for refueling activities.

Licensee's Proposed Alternative Examination (as stated):

Perform the required VT-2 visual examinations for the reactor vessel flange leak-off lines during the regularly scheduled Class 1 system pressure test that is performed following each refueling outage. The reactor vessel flange leak-off lines will not be pressurized during the VT-2 visual examinations to RCS pressure (2235 psig) using reactor coolant as a pressurizing medium. However, the reactor vessel flange leak-off lines are filled with borated water at a pressure of approximately 50 psig, which corresponds to the static head in the reactor cavity during refueling operations. Since borated water leaves a crystalline residue, the proposed VT-2 visual examination provides reasonable assurance that through-wall leakage in the reactor vessel flange leak-off lines will be detected and corrected.

Evaluation: The Code requires that licensees perform system leakage tests during each refueling outage, just prior to starting up the plant. These leakage tests must be conducted at a system pressure not less than the pressure corresponding to 100% rated reactor power. The subject reactor vessel flange leak-off lines are designed to only experience reactor coolant pressure if the O-ring seal on the RPV flange fails during power operations. There is no method to pressurize these small 3/4-inch outside

diameter (OD) lines without removing the O-ring in the flange seal region, which would prevent the reactor coolant system from being pressurized.
Therefore, leakage testing of these flange leak-off lines to normal reactor coolant pressure (2235 psig) is impractical.

The Class 1 RPV flange seal leak-off lines at Kewaunee are 3/4-inch OD piping segments that connect to 3/8-inch OD reducers which route potential flange leakage through valves RC-40A and RC-40B. Portions of the piping, including the reducers and downstream segments and valves, are non-ASME class components. The purpose of these lines is to allow detection of RPV O-ring seal failure during normal plant operation. There is no method to pressurize the 3/4-inch OD portions of the subject lines to normal reactor coolant pressure without compromising the RPV flange seal. During shutdown conditions with the RPV head (and O-ring seal) removed, these piping segments are exposed to borated water in the refueling canal and experience line pressure of approximately 50 psig. The licensee has proposed to perform visual VT-2 examinations of this piping during normal system pressure tests at the end of each refueling outage. At that time, if any through-wall leakage has occurred, boric acid residue will be visible on the outside surface of the piping, providing a method to ensure that any potential leakage will be detected prior to compromising the intended function of these flange leak-off lines.

Therefore, based on the impracticality of pressuring the RPV flange seal leak-off lines to normal reactor coolant system pressure, and considering the alternative visual examination proposed by the licensee, it is recommended that, pursuant to 10 CFR 50.55a(g)(6)(i), Request for Relief RR-1-5 be granted for the fourth interval at Kewaunee.

3.6 Request for Relief RR-1-7, Pressure Retaining Welds in Piping Subject to Appendix VIII, Supplement 10, Qualification Requirements for Dissimilar Metal Piping Welds

Code Requirement: Performance demonstration requirements for qualifying procedures, personnel and equipment to inspect dissimilar metal piping welds are listed in the 1998 Edition/2000 Addenda of ASME Section XI, Appendix VIII, Supplement 10. Licensees may 1) elect to use the requirements of Supplement 10 as listed, 2) seek NRC approval for new ASME code cases currently being reviewed by Code Committees, or 3) propose an alternative to Code requirements. The licensee proposed to use the industry's Performance Demonstration Initiative (PDI) program as an alternative to the following paragraphs of Supplement 10:

- Paragraph 1.1(b) states in part - Pipe diameters within a range of 0.9 to 1.5 times a nominal diameter shall be considered equivalent.

- Paragraph 1.1(d) states - All flaws in the specimen set shall be cracks.

- Paragraph 1.1(d)(1) states - At least 50% of the cracks shall be in austenitic material. At least 50% of the cracks in austenitic material shall be contained wholly in weld or buttering material. At least 10% of the cracks shall be in ferritic material. The remainder of the cracks may be in either austenitic or ferritic material.

- Paragraph 1.2(b) states in part - The number of unflawed grading units shall be at least twice the number of flawed grading units.

- Paragraph 1.2(c)(1) and 1.3(c) state in part - At least 1/3 of the flaws, rounded to the next higher whole number, shall have depths between 10% and 30% of the nominal pipe wall thickness. Paragraph 1.4(b) distribution table requires 20% of the flaws to have depths between 10% and 30%.

- Paragraph 2.0 first sentence states - The specimen inside surface and identification shall be concealed from the candidate.

- Paragraph 2.2(b) states in part - The regions containing a flaw to be sized shall be identified to the candidate.

- Paragraph 2.2(c) states in part - For a separate length-sizing test, the regions of each specimen containing a flaw to be sized shall be identified to the candidate.

- Paragraph 2.3(a) states - For the depth sizing test, 80% of the flaws shall be sized at a specific location on the surface of the specimen identified to the candidate.

- Paragraph 2.3(b) states - For the remaining flaws, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region.

- Table VIII-S2-l provides the false call criteria when the number of unflawed grading units is at least twice the number of flawed grading units.

Licensee's Proposed Alternative to Code: Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed using the PDI program in lieu of the requirements of ASME Section XI, 1998 Edition with 2000 Addenda, Appendix VIII, Supplement 10. The Electric Power Research Institute (EPRI) PDI program is described in the submittal as supplemented. Dissimilar metal welds affected by this alternative are listed below.

SI-W112DM	RC-1DM	RC-W26DM
SI-W54DM	RC-W30DM	RC-W58DM
PS-W61DM	RCW67DM	PR-1DM
PR-W16DM	PR-W26DM	RC-W76DM
RC-W77DM	RC-W78DM	RC-W79DM

Licensee's Bases for Alternative (as stated):

The proposed alternative is based on forthcoming Code action and was generated from a PDI model prepared by EPRI.

Item 1- The proposed alternative to Paragraph 1.1(b) states:

"The specimen set shall include the minimum and maximum pipe diameters and thicknesses for which the examination procedure is applicable. Pipe diameters within a range of 1/2 in. (13 mm) of the nominal diameter shall be considered equivalent. Pipe diameters larger than 24 in. (610 mm) shall be considered to be flat. When a range of thicknesses is to be examined, a thickness tolerance of ±25% is acceptable."

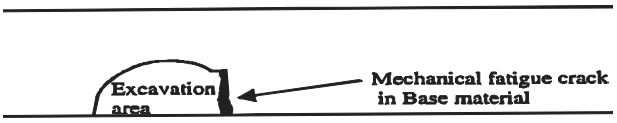
Technical Basis - The change in the minimum pipe diameter tolerance from 0.9 times the diameter to the nominal diameter minus 0.5 inch provides tolerances more in line with industry practice. Though the alternative is less stringent for small pipe diameters they typically have a thinner wall thickness than larger diameter piping. A thinner wall thickness results in shorter sound path distances that reduce the detrimental effects of the curvature. This change maintains consistency between Supplement 10 and the recent revision to Supplement 2.

Item 2 - The proposed alternative to Paragraph 1.1 (d) states:

"At least 60% of the flaws shall be cracks; the remainder shall be alternative flaws. Specimens with IGSCC shall be used when available. Alternative flaws, if used, shall provide crack-like reflective characteristics and shall be limited to the case where implantation of cracks produces spurious reflectors that are uncharacteristic of actual flaws. Alternative flaw mechanisms shall have a tip width of less than or equal to 0.002 in. (.05 mm).

Note, to avoid confusion the proposed alternative modifies instances of the term "cracks" or "cracking" to the term "flaws" because of the use of alternative flaw mechanisms."

Technical Basis - As illustrated below, implanting a crack requires excavation of the base material on at least one side of the flaw. While this may be satisfactory for ferritic materials, it does not produce a useable axial flaw in austenitic materials because the sound beam, which normally passes only through base material, must now travel through weld material on at least one side, producing an unrealistic flaw response. In addition, it is important to preserve the dendritic structure present in field welds that would otherwise be destroyed by the implantation process. To resolve these issues, the proposed alternative allows the use of up to 40% fabricated flaws as an alternative flaw mechanism under controlled conditions. The fabricated flaws are isostatically compressed which produces ultrasonic reflective characteristics similar to tight cracks.



Item 3- The proposed alternative to Paragraph 1.1(d)(1) states:

“At least 80% of the flaws shall be contained wholly in weld or buttering material. At least one and a maximum of 10% of the flaws shall be in ferritic base material. At least one and a maximum of 10% of the flaws shall be in austenitic base material."

Technical Basis - Under the current Code, as few as 25% of the flaws are contained in austenitic weld or buttering material. Recent experience has indicated that flaws contained within the weld are the likely scenarios. The metallurgical structure of austenitic weld material is ultrasonically more challenging than either ferritic or austenitic base material. The proposed alternative is therefore more challenging than the current Code.

Item 4 - The proposed alternative to Paragraph 1.2(b) states:

"Detection sets shall be selected from Table VIII-S10-1. The number of unflawed grading units shall be at least one and a half times the number of flawed grading units."

Technical Basis - Table VIII-S10-1 provides a statistically based ratio between the number of unflawed grading units and the number of flawed grading units. The proposed alternative reduces the ratio to 1.5 times to reduce the number of test samples to a more reasonable number from the human factors perspective. However, the statistical basis used for screening personnel and procedures is still maintained at the same level with competent personnel being successful and less skilled personnel being unsuccessful. The acceptance criteria for the statistical basis are in Table VIII-S10-1.

Item 5 - The proposed alternative to the flaw distribution requirements of Paragraph 1.2(c)(1) (detection) and 1.3(c) (length) is to use the Paragraph 1.4(b) (depth) distribution table (see below) for all qualifications.

Flaw Depth Minimum (% Wall Thickness)	Number of Flaws
(10 - 30)	20%
(31 - 60)	20%
(61 - 100)	20%

Technical Basis - The proposed alternative uses the depth sizing distribution for both detection and depth sizing because it provides for a better distribution of flaw sizes within the test set. This distribution allows candidates to perform detection, length, and depth sizing demonstrations simultaneously utilizing the same test set. The requirement that at least 75% of the flaws shall be in the range of 10 to 60% of wall thickness provides an overall distribution tolerance yet the distribution uncertainty decreases the possibilities for testmanship that would be inherent to a uniform distribution. It must be noted that it is possible to achieve the same distribution utilizing the present requirements, but it is preferable to make the criteria consistent.

Item 6 - The proposed alternative to Paragraph 2.0 first sentence states:

"For qualifications from the outside surface, the specimen inside surface and identification shall be concealed from the candidate. When qualifications are performed from the inside surface, the flaw location and specimen identification shall be obscured to maintain a "blind test"."

Technical Basis - The current Code requires that the inside surface be concealed from the candidate. This makes qualifications conducted from the inside of the pipe (e.g., PWR nozzle to safe end welds) impractical. The proposed alternative differentiates between ID and OD scanning surfaces, requires that they be conducted separately, and requires that flaws be concealed from the candidate. This is consistent with the recent revision to Supplement 2.

Items 7 and 8 - The proposed alternatives to Paragraph 2.2(b) and 2.2(c) state:

"...Containing a flaw to be sized may be identified to the candidate."

Technical Basis - The current Code requires that the regions of each specimen containing a flaw to be length sized shall be identified to the candidate. The candidate shall determine the length of the flaw in each region (Note, that length and depth sizing use the term "regions" while detection uses the term "grading units" - the two terms define different concepts and are not intended to be equal or interchangeable). To ensure security of the samples, the proposed alternative modifies the first "shall" to a "may" to allow the test administrator the option of not identifying specifically where a flaw is located. This is consistent with the recent revision to Supplement 2.

Items 9 and 10 - The proposed alternative to Paragraph 2.3(a) and 2.3 (b) state:

"... Regions of each specimen containing a flaw to be sized may be identified to the candidate."

Technical Basis - The current Code requires that a large number of flaws be sized at a specific location. The proposed alternative changes the "shall" to a "may" which modifies this from a specific area to a more generalized region to ensure security of samples. This is consistent with the recent revision to Supplement 2. It also incorporates terminology from length sizing for additional clarity.

Item 11 - The proposed alternative modifies the acceptance criteria of Table VIII-S2-1.

Technical Basis - The proposed alternative is identified as new Table S-10-1. It was modified to reflect the reduced number of unflawed grading units and allowable false calls. As a part of ongoing Code activities, PNNL has reviewed the statistical significance of these revisions and offered the revised Table S-10-1.

<u>Evaluation:</u> The licensee proposed to use the program developed by PDI that modifies selected aspects of the Code requirements. The differences between the Code and the PDI program are discussed below.

Paragraph 1.1(b)
The Code requirement of “0.9 to 1.5 times the nominal diameter are equivalent” was established for a single nominal diameter. When applying the Code-required tolerance to a range of diameters, the tolerance rapidly expands on the high side. Under the current code requirements, a 5-inch OD pipe would be equivalent to a range of 4.5-inch to 7.5-inch diameter pipe. Under the proposed PDI guidelines, the equivalent range would be reduced to 4.5-inch to 5.5-inch diameter pipe. With current Code requirements, a 16-inch nominal diameter pipe would be equivalent to a range of 14.4-inch to 24-inch diameter pipe. The proposed alternative would significantly reduce the equivalent range to between 15.5-inch and 16.5-inch. The difference between Code and the proposed alternative for diameters less than 5-inches is not significant because of shorter metal path and beam spread associated with smaller diameter piping. The proposed alternative is considered more conservative than current Code requirements, and, therefore, provides an acceptable level of quality and safety.

Paragraph 1.1(d)
The Code requires all flaws to be cracks. Manufacturing test specimens containing cracks free of spurious reflections and telltale indicators is extremely difficult in austenitic material. To overcome these difficulties, PDI developed a process for fabricating flaws that produce UT acoustic responses similar to the responses associated with real cracks. PDI presented its process for discussion at public meetings held June 12 through 14, 2001 and January 31 through February 2, 2002 at the EPRI NDE Center, Charlotte, NC. The staff attended these meetings and determined that the process parameters used for manufacturing fabricated flaws resulted in acceptable acoustic responses. PDI is selectively installing these fabricated flaws in specimen locations that are unsuitable for real cracks. The proposed alternative paragraph provides an acceptable level of quality and safety.

Paragraph 1.1(d)(1)
The Code requires that at least 50% of the flaws be contained in austenitic material, 50% of the flaws in the austenitic material shall be contained fully in weld or buttering material. This means that at least 25% of the total flaws must be located in the weld or buttering material. Field experience shows that flaws identified during ISI of dissimilar metal welds are more likely to be located in the weld or buttering material. The grain structure of austenitic weld and buttering material represents a much more stringent ultrasonic scenario than that of a ferritic material or austenitic base material. Flaws made in austenitic base material are difficult to create free of spurious reflectors and telltale indicators. The proposed alternative of 80% of the flaws in the weld metal or buttering material provides a challenging testing scenario reflective of field experience and minimizes testmanship associated with telltale reflectors common to placing flaws in austenitic base material. The proposed alternative paragraph provides an acceptable level of quality and safety.

Paragraph 1.2(b) and Paragraph 3.1
The Code requires that detection sets meet the requirements of Table VIII-S2-1 which specifies the minimum number of flaws in a test set to be 5 with 100% detection. The current Code also requires the number of unflawed grading units to be two times the number of flawed grading units. The proposed alternative would follow the same pass/fail screening criteria of the table beginning with a minimum number of flaws in a test set being 10, and reducing the number of false calls to one and a half times the number of flawed grading units, while maintaining the same statistical design basis as the Code. The proposed alternative paragraphs satisfy the pass/fail objective established for Appendix VIII performance demonstration acceptance criteria, and, therefore, provide an acceptable level of quality and safety.

Paragraph 1.2(c)(1) and Paragraph 1.3(c)
For detection and length sizing, the Code requires at least 1/3 of the flaws be located between 10 and 30% through the wall thickness and 1/3 located greater than 30% through the wall thickness. The remaining 40% would be located randomly throughout the wall thickness. The proposed alternative sets the distribution criteria for detection and length sizing to be the same as the depth sizing distribution, which stipulates that at least 20% of the flaws be located in each of the increments of 10-30%, 31-60% and 61-100%. The remaining 40% would be located randomly throughout the pipe thickness. With the exception of the 10-30% increment, the proposed alternative is a subset of the current Code requirements. The 10-30% increment would be in the subset if it contained at least 30% of the flaws. The change simplifies assembling test sets for detection and sizing qualifications and is more indicative of conditions in the field. The proposed alternative paragraphs provide an acceptable level of quality and safety.

Paragraph 2.0
The Code requires the specimen inside surface be concealed from the candidate. This requirement is applicable for test specimens used for qualification performed from the outside surface. With the expansion of Supplement 10 to include qualifications performed from the inside surface, the inside surface must be accessible while maintaining the specimen integrity. The proposed alternative requires that flaws and specimen identifications be obscured from candidates, thus maintaining blind test conditions. The proposed alternative paragraph provides an acceptable level of quality and safety.

Paragraph 2.2(b) and 2.2(c) -
The Code requires that the location of flaws added to the test set for length sizing shall be identified to the candidate. The proposed alternative is to make identifying the location of additional flaws an option. This option provides an additional element of difficulty to the testing process because the candidate would be expected to demonstrate the skill of detecting and sizing flaws over an area larger than a specific location. The proposed alternative paragraph is more conservative than Code requirements and, therefore, provides an acceptable level of quality and safety.

Paragraph 2.3(a)
The Code requirement is that 80% of the flaws be sized in a specific location that is identified to the candidate. The proposed alternative permits detection and depth sizing to be conducted separately or concurrently. In order to maintain a blind test, the

location of flaws cannot be shared with the candidate. For depth sizing that is conducted separately, allowing the test administrator the option of not identifying flaw locations makes the testing process more challenging. The alternative is more conservative than the Code requirements and, therefore, provides an acceptable level of quality and safety.

Paragraph 2.3(b)
The Code requires that the location of flaws added to the test set for depth sizing shall be identified to the candidate. The proposed alternative is to make identifying the location of additional flaws an option. This option provides an additional element of difficulty to the testing process because the candidate would be expected to demonstrate the skill of finding and sizing flaws in an area larger than a specific location. The alternative is more conservative than the Code requirements and, therefore, provides an acceptable level of quality and safety.

Pursuant to 10 CFR 50.55a(a)(3)(i), and based on the evaluations above, it is recommended that Request for Relief RR-1-7 be authorized for the fourth interval inservice inspection at Kewaunee.

3.7 Request for Relief RR-1-8, Pressure Retaining Welds in Piping Subject to Appendix VIII, Supplements 2 and 10, Qualification Requirements for Dissimilar Metal Piping Welds

Code Requirement: Performance demonstration requirements for qualifying procedures, personnel and equipment to inspect dissimilar metal piping welds are listed in the 1998 Edition/2000 Addenda of ASME Section XI, Appendix VIII, Supplement 10. Licensees may 1) elect to use the requirements of Supplement 10 as listed, 2) seek NRC approval for new ASME code cases currently being reviewed by Code Committees, or 3) propose an alternative to Code requirements.

Licensee's Proposed Alternative to Code:

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed to use a modification of the industry's Performance Demonstration Initiative (PDI) program as an alternative to the requirements listed in ASME XI, Appendix VIII, Table VIII-3110-1 for Supplement 2 Wrought Austenitic Piping Welds, as coordinated with the proposed alternative for the Supplement 10 Dissimilar Metal Piping Welds implementation program. The Electric Power Research Institute (EPRI) PDI program is described in the submittal as supplemented. This alternative applies to examinations performed from the inside surface of PWR piping using automated inspection systems. (For reference, this PDI alternative is being routed through the Code approval process as proposed Supplement 14).

Licensee's Bases for Alternative (as stated):

The Kewaunee Nuclear Power Plant reactor vessel nozzles (4) to main coolant piping and reactor nozzles (2) to safety injection piping are fabricated using ferritic components and assembled using austenitic or dissimilar metal welds. Additionally, differing combinations of these assemblies may be in close proximity, which typically means the same ultrasonic essential variables are used for each weld and the most challenging

ultrasonic examination process is employed (e.g., the ultrasonic examination process associated with a dissimilar metal weld would be applied to an austenitic weld).

Separate qualifications to Supplements 2 and 10 are redundant when done in accordance with the PDI program. For example, during a personnel qualification to the PDI program, the candidate would be exposed to a minimum of 10 flawed grading units for each individual supplement. Personnel qualification to Supplements 2 and 10 would therefore require a total of 20 flawed grading units. Test sets this large and tests of this duration are impractical. Additionally, a full procedure qualification (i.e. 2 personnel qualifications) to the PDI Program requirements would require 60 flawed grading units. This is particularly burdensome for a procedure that will use the same essential variables or the same criteria for selecting essential variables for the two supplements.

To resolve these issues, the PDI program recognizes the Supplement 10 qualification as the most stringent and technically challenging ultrasonic application. The essential variables used for the examination of Supplement 2 and 10 are the same. A coordinated add-on implementation would be sufficiently stringent to qualify Supplement 2 if the requirements used to qualify Supplement 10 are satisfied as a prerequisite. The basis for this conclusion is the fact that the majority of the flaws in Supplement 10 are located wholly in austenitic weld material. This configuration is known to be challenging for ultrasonic techniques due to the variable dendritic structure of the weld material. Conversely, flaws in Supplement 2 initiate in the fine-grained base materials.

Additionally, the proposed alternative is more stringent than current Code requirements for a detection and length sizing qualification. For example, the current Code would allow a detection procedure, personnel, and equipment to be qualified to Suppliment 10 with 5 flaws and Supplement 2 with 5 flaws, a total of only 10 flaws. The proposed alternative of qualifying Supplement 10 using 10 flaws and adding on Supplement 2 with 5 flaws result in a total of 15 flaws will be multiplied by a factor of 3 for the procedure qualification.

Based on the above, the use of a limited number of Supplement 2 flaws is sufficient to access the capabilities of procedures and personnel who have already satisfies Supplement 10 requirements. The statistical basis used for screening personnel and procedures is still maintained at the same level with competent personnel being successful and less skilled personnel being unsuccessful The proposed alternative is consistent with other coordinated qualifications currently contained in Appendix VIII.

The proposed alternative program is attached[1] and is identified as Supplement 14. It has been submitted to the ASME Code for consideration as new Supplement 14 to Appendix VIII and as of February 2002 has been approved by Subcommittee on Nuclear Inservice Inspection.

Evaluation: The licensee requested relief from the qualification requirements of ASME Section XI, Appendix VIII, Supplement 2 criteria. The Code currently requires separate qualifications for Supplements 2 (austenitic piping welds), 3 (ferritic piping welds), and

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1 The PDI alternative submitted as part of the licensee's request is not included in this report.

10 (dissimilar metal piping welds). Qualifications for each supplement would entail a minimum of 10 flaws each for a total of 30 flaws minimum. The minimum number of flaws per supplement established a statistical-based pass\fail objective. The process of a single qualification for each supplement would greatly expand the minimum number of ferritic and austenitic flaws required to be identified which would also raise the pass\fail acceptance criteria.

The Code recognized that flaws in austenitic materials are more difficult to detect and size than flaws in ferritic material. In addition, the prevailing reasoning concluded that a Supplement 3 qualification following a Supplement 2 qualification had diminishing returns on measuring personnel skills and procedure effectiveness. Therefore, in lieu of separate Supplements 2 and 3 qualifications, the ASME Code developed proposed Supplement 12 which provides for a Supplement 2 add-on to a Supplement 3 qualification. The add-on consists of a minimum of 5 flaws in austenitic material. A statistical evaluation of Supplement 12 acceptance criteria satisfied the pass\fail objective established for Appendix VIII performance demonstration acceptance criteria.

The licensee's proposed alternative builds upon the experiences of Supplement 12 by starting with the most challenging Supplement 10 qualifications, as implemented by the PDI program (PDI Supplement 10), and adding a sufficient number of flaws to demonstrate the personnel skills and procedure effectiveness to satisfy Supplement 2 qualifications. A PDI Supplement 10 performance demonstration has at least 1 flaw with a maximum of 10% of the total number of flaws being in the ferritic material. The rest of the flaws are in the more challenging austenitic material. When expanding the PDI Supplement 10 qualification to include Supplement 2, the proposed alternative would add a minimum of 5 flaws in austenitic material to the performance demonstration. The performance demonstration results added to the appropriate PDI Supplement 10 results must satisfy the acceptance criteria of the PDI Supplement 10. A statistical evaluation performed by the Pacific Northwest National Laboratories, showed that the proposed alternative acceptance criteria satisfied the pass\fail objective established for Appendix VIII for an acceptable performance demonstration.

It has been determined that use of a limited number of flaws to qualify Supplement 2 as coordinated with the PDI developed alternative to Supplement 10, will provide equivalent flaw detection performance to that of the Code-required qualification for piping welds. As such, the licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), it is recommended that the licensee's proposed alternative contained in Request for Relief RR-1-8 be authorized for the fourth interval at Kewaunee.

3.8 Request for Relief RR-1-9, Examination Category B-D, Item B3.90, Full Penetration Welded Nozzles in Vessels

By letter dated September 17, 2004, the licensee withdrew Request for Relief RR-1-9.

3.9 Request for RR-1-10, Examination Category B-A, Pressure Retaining Welds in Reactor Vessel, Examinations Subject to Qualification Requirements of Appendix VIII, Supplement 4, Flaw-Sizing

Code Requirement: ASME Code Appendix VIII, Supplement 4, Sub-paragraphs 3.2(a), (b) and (c) require that ultrasonic performance demonstration flaw-sizing results be graphed by plotting the candidate's reported size along the ordinate and the true depth along the abscissa. For successful qualification, the plotted data must satisfy the following statistical parameters: (1) the slope of the linear regression line is not less than 0.7; (2) the mean deviation of the flaw depth is less than 0.25 in.; and (3) the correlation coefficient is less than 0.70.

Licensee's Proposed Alternative to Code: In accordance with 10 CFR 50.55a(a)(3)(i), the licensee proposed an alternative to the requirements listed in Supplement 4, Subparagraph 3.2(c), for determining a successful qualification for flaw depth sizing. The alternative consists of two parts:

1) Use the flaw-sizing acceptance criteria of 0.15-inch Root Mean Square (RMS) Error (depth) and 0.75-inch (length) as listed in 10 CFR 50.55a(b)(2) (xv)(C)(1), which modifies Code Sub-paragraphs 3.2(a) and 3.2)b), and

2) Perform the qualification analysis for flaw-sizing capability by determining RMS error in lieu of the statistical parameters of ASME Boiler and Pressure Vessel Code Section XI 1998 Eddition 2000 Addenda Appendix VIII, Supplement 4, Subparagraph 3.2(c).

The licensee will implement this alternative for the inspection of RPV shell Welds RV-W2, RV-W3, RV-W4, and RV-W5.

Licensee's Basis for Alternative (as stated):

On September 22, 1999, the NRC published a final rule in the Federal Register (64 FR 51378) to amend 10 CFR 50.55a(b)(2), to incorporate by reference the 1995 Edition and Addenda through the 1996 Addenda, of Section XI of the ASME Code. The change included the provisions of subparagraph 3.2(a), 3.2(b), and 3.2(c) of Section XI of the ASME Code, 1995 Edition with 1996 Addenda, Appendix VIII, Supplement 4.

The September 22, 1999, Federal Register amended 10 CFR 10.55a(b)(2)(xv)(C)(1). The amended 10 CFR 10.55a(b)(2)(xv)(C)(1) requires a depth sizing acceptance criterion of 0.15 inch RMS to be used in lieu of the requirements of Subparagraph 3.2(a) and 3.2(b) of Section XI of the ASME Code, Appendix XI, Supplement 4.

On March 26, 2001, the NRC published a correction to the September 22, 1999, final rule in the Federal Register (66 FR 16390). The NRC identified that an error had occurred in the published wording of 10 CFR 50.55a(b)(2)(xv)(C)(1). The corrected 10 CFR 10.55a(b)(2)(xv)(C)(1) requires a depth sizing acceptance criterion of 0.15 inch RMS be used in lieu of the requirements of Subparagraph 3.2(a) and the length sizing requirement of 0.75 inch RMS to be used in lieu of the requirements of 3.2(b) of Section XI of the ASME Code, Appendix VIII, Supplement 4.

The Nuclear utilities created the Performance Demonstration Initiative (PDI) to implement performance demonstration requirements contained in Appendix VIII of Section XI of the ASME Code. To this end, PDI has developed a performance

demonstration program for qualifying UT equipment, procedures, and personnel. During the development of the performance demonstration for Supplement 4, the PDI determined that the code criteria for flaw sizing was unworkable.

Evaluation: Supplement 4, Subparagraph 3.2(c) imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is a best fit line obtained by the least-square method using data points of UT measured flaw depth versus actual flaw depth. For Supplement 4 performance demonstrations, a best fit line acquired by the linear regression method would be calculated from data points that come from the inner 15% of the wall thickness. Plotting the data, UT measured flaw depth versus true flaw depth, produce closely grouped data points that resemble a shotgun pattern. The slope of a line calculated by linear regression from data points that are so close together would not produce meaningful results because the line would be extremely sensitive to small variations in depth measurements. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The Code currently requires a mean deviation flaw depth of less than 0.25-inch versus the licensee proposed 0.15 RMS value. The licensee's proposal to use the more restrictive criterion of 0.15 RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion is more conservative than Code and follows the PDI protocol. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1). In addition, the licensee's use of 0.75-inch RMS for flaw length-sizing acceptance is consistent with the requirements stated in 10 CFR 50.55a(b)(2)(xv)(C)(1).

It has been determined that the proposed alternative to Supplement 4, as administered by the PDI program will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), it is recommended that Request for Relief RR-1-10 be authorized for the fourth interval at Kewaunee.

3.10 Request for Relief RR-2-1, Examination Category C-H, Item C7.10, All Pressure Retaining Components

Code Requirement: Examination Category C-H, Item C7.10, requires a system leakage test be performed on piping systems once each ISI period. Requirements for system leakage test boundaries, pressures, and temperatures are listed in paragraph IWC-5220.

Licensee's Code Relief Request: In accordance with 10 CFR 50.55a(g)(5)(iii), the licensee requested relief from performing system leakage tests at normal reactor coolant pressure (2235 psig) and temperature for Class 2 portions of Reactor Vessel Head Vent piping.

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

The purpose of the Reactor Coolant Vent System is to vent non-condensible gases from the high points of the Reactor Coolant System to assure that core cooling during natural circulation will not be inhibited and to vent the vessel head during a plant startup.

The reactor vessel head vent lines downstream of PR-33A, PR-33B, RC-45A, and RC-45B, are not pressurized to 2235 psig when the RCS is operated at 100% rated power and approximately 547° F. The Kewaunee Nuclear Power Plant Technical Specifications do not permit pressurization of the reactor vessel head vent lines above 200° F using reactor coolant system pressure and thus valves PR-33A, PR-33B, RC-45A and RC45B are required to be maintained closed.

Licensee's Proposed Alternative Examination (as stated):

Perform the required Class 2 VT-2 visual examinations for the reactor vessel head vent lines prior to 200° F once each 3 1/3 year period using Reactor Coolant as a pressurizing medium when the pressure will be approximately 380 psig. Perform VT-2 visual examinations for the reactor vessel vent lines. During the regularly scheduled Class 1 system pressure test (Table IWB-2500-1; Category B-P; Item Number B15.50 and 15.70) that is performed following each refueling outage. The reactor vessel head vent lines downstream of PR-33A, PR-33B, RC-45A, and RC-45B will not be pressurized, during the Class 1 VT-2 visual examinations, to RCS pressure (2235 psig) using reactor coolant as the pressurizing medium. However, the reactor vessel head vent lines are filled with borated water following each Refueling Outage when performing static and dynamic testing of PR-33A, PR-33B, RC-45A, and RC-45B. Since borated water leaves a crystalline residue, the proposed VT-2 visual examination provides reasonable assurance that through-wall leakage in the reactor head vent lines will be detected and corrected.

Evaluation: The Code requires that licensees perform system leakage tests of all Class 2 piping systems during each ISI period. These leakage tests must be conducted at a system pressure not less than the pressure corresponding to 100% rated reactor power (2235 psig). The subject reactor coolant vent system lines are small 1-inch and ½-inch outside diameter (OD) piping designed to vent non-condensible gases from the high points of the reactor coolant system (RCS) to ensure that core cooling during natural circulation will not be inhibited, as well as, to vent the vessel head when filling the RCS during a plant startup. Kewaunee Technical Specifications (TS) do not allow pressurizing the head vent lines above 200 degrees F using RCS normal system pressure, thus isolation valves are required to be maintained closed when the RCS is at normal operating pressure and temperature. In order to pressurize these lines to normal RCS pressure and temperature, the plant would be forced to violate the TS, and design modifications may be necessary to the valves and piping in this system. This presents the licensee with a considerable difficulty.

The licensee has proposed an alternative that consists of two parts: 1) perform the VT-2 visual examinations of reactor vessel head vent lines each period using RCS at approximately 380 psig and < 200 degrees F, and 2) perform a VT-2 examination of the head vent lines during the Class 1 RCS system pressure test during each refueling outage. During the part 2 examinations, head vent lines downstream of the normally closed isolation valves will not be pressurized, however, these lines are normally filled during each refueling outage with borated water. If any through-wall leakage has occurred, boric acid residue will be visible on the outside surface of the piping, providing a method to ensure that any potential leakage will be detected prior to compromising the intended function of these head vent lines. The licensee's two-part alternative satisfies

the intent of the Code requirement, and provides reasonable assurance that the subject head vent lines will continue to operate as designed.

To require the licensee to pressurize the reactor vessel head vent lines to normal RCS pressure and temperature would present an unusual difficulty with no compensating increase in quality or safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), it is recommended that the licensee's alternative described in Request for Relief RR-2-1 be authorized for the fourth ISI interval at Kewaunee.

3.11 Request for Relief RR-G-2, Examination Categories B-P and C-H, All Pressure Retaining Components, Insulated Bolted Connections

Code Requirement: Examination Categories B-P and C-H, require system leakage tests to be performed on all Class 1 and 2 pressure retaining components once each refueling outage, and ISI period, respectively. In addition, Paragraph IWA-5242 requires that, for systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure retaining bolted connections for the purpose of performing visual VT-2 examinations.

Licensee's Proposed Alternative to Code (as stated):

A. Perform the VT-2 visual examination required by Table IWB-2500-1 and IWC-2500-1 without the removal of insulation. A 4-hour hold time shall be established prior to the VT-2 visual examination to allow leakage from the subject bolted connections to migrate through the insulation. Any evidence of leakage will be evaluated in accordance with IWA-5250(a)(2) through utilization of ASME Boiler and Pressure Vessel Code Section XI Code Case N-566-1. During the inservice leak test, the exposed insulation surfaces and joints at bolted connections shall be VT-2 visually examined.

B. For Pressure retaining bolted connections in Class 1 Valves, Class 1 Flanges, Class 2 Valves and the pressurizer manway, perform a supplemental VT-3 visual examination once every refueling outage without disassembly and without the system under operating pressure and temperature, during cold shutdown or refueling shutdown. No supplemental examinations are required to ensure integrity of the pressure retaining studs in the reactor vessel flange each outage. The steam generator primary side manway bolting insulation is removed, due to the ease of replacing, during the Class 1 System Pressure Test so no supplemental examinations are needed to ensure their integrity.

Performing the VT-3 visual examinations during cold shutdown or refueling shutdown will significantly reduce the plant operational concerns, personal radiation, personal safety. Since borated water leaves a crystalline residue, the proposed supplemental VT-3 visual examination (in addition to the Class 1 system pressure test, area radiation monitors, and RCS leakage detective system) provides reasonable assurance that leakage at pressure retaining bolted connections will be detected and corrected. The proposed VT-3 visual examination at cold or refueling shutdown will permit a more thorough

examination than during the Class 1 and Class 2 system pressure test due to better accessability.

Licensee's Basis for Alternative (as stated):

Satisfying the Code requirement of removing insulation from pressure retaining bolted connections for visual examination of borated systems will require significant planning and scheduling due to operational concerns, personnel radiation, and personnel safety. VT-2 examinations of the Class 1 System at the Kewaunee Nuclear Power Plant are performed at a system operating pressure of 2235 psig and a system temperature of 547° F. Area radiation levels range from 5 mr/hr to 100 mr/hr. Re-insulating and the removal of access equipment after the VT-2 examination will require additional staff to be exposed to higher system pressure, system temperature, and radiation levels than would be experienced during cold shutdown or refueling shutdown.

Additionally, the time required to replace insulation and remove the access equipment after the VT-2 examination may delay plant startup for an anticipated short time duration between performance of the Class 1 system pressure test and placing the reactor into critical operation. This relief request is intended to cover all pressure retaining bolted connections that are insulated and require VT-2 visual examination under Table IWB-2500-1 and IWC-2500-1. Representative components listed below are insulated, are part of or connected to the reactor coolant system, contain pressure retaining bolting, and are pressurized during the Class 1 system pressure test and class 2 system pressure test.

Pressure Retaining Components with Bolted Connections that are Insulated		
Reactor Vessel Closure Head Flange Studs	8" Valve RHR-1A	3" FE-459
Reactor Vessel Closure Head 40 CRDM's and 1-3/4" Head Vent	8" Valve RHR-1B	2" Valve LD-4A
Pressurizer Manway	6" Valve SI-13A	2" Valve LD-4B
Steam Generator Primary Side Manway	6" Valve SI-13B	2" Valve LD-4C
2" Valve LD-2	12" Valve SI-22A	8" Valve SI-2A
2" Valve LD-3	12" Valve SI-22B	8" Valve SI-2B
3" Valve PS-1A	6" Valve SI-304A	8" Valve SI-3
3" Valve PS-1B	6" Valve SI-304B	3" Valve RC-103A
3" Valve RC-103A	3" Valve RC-103B	3" FE-458

Response to Request for Additional Information (as stated):

KNPP relief request follows the alternative requirements located in ASME Boiler and Pressure Vessel Code, Section XI, Code Case N-533-1, which was approved for use by the Nuclear Regulatory Commission in Regulatory Guide 1.147 Rev. 13, January 13, 2004, as a conditionally acceptable Section XI Code Case. An added requirement was

"Prior to conducting the VT-2 examination, the provisions of the IWA-5213 'Test Condition Holding Times', 1989 Edition, are to be followed".

KNPP's concern with Code Case N-533-1 was the performance of a VT-2 examination of the bolted connection when the insulation is removed each refueling outage. Per ASME Boiler and Pressure Vessel Code, Section XI, 1998 Edition, 2000 Addenda, IWA-5211 test description, a VT-2 examination is performed for a system leakage test while the system is in operation, during a system operability test, or while the system is at rest conditions using an external pressurization source. Since the bolted connections in question will not be under pressure, a VT-2 examination cannot be performed. Thus, a VT-3 was suggested under relief request RR-G-2, instead of a VT-2, as required by the ASME Code Case N-533-1. A VT-3 examination will be a more thorough examination, as the lighting requirements, examination angle, and examination distance are more stringent. Additional experience requirements are also necessary for VT-3 examiners' qualifications. Previous examinations of the bolting performed during the third 10-year in-service inspection interval were performed to that requested in relief request RR-G-2. This is based on the requested relief that was submitted in December 1993, for the third 10-year interval, approved by the Nuclear Regulatory Commission, and incorporated in the in-service inspection program. Thus, no man hours have been involved with previous examinations and no radiation exposure has been accrued by personnel having been accrued during the tests for examinations with the insulation removed, scaffolding in place, and under test conditions of 2235 psig and 547°F prior to reactor startup following a refueling outage.

KNPP is not invoking NRC approved ASME Code Case N-533-1 for relief request RR-G-2. Reference to ASME Code Case N-533-1 was for discussion with NRC reviewers to indicate where KNPP gathered data for relief request RR-G-2 and the current industry practice. The initial relief request RR-G-2 submitted December 16, 2003 is intended as a stand alone relief request which does not reference ASME Code Case N-533-1.

Evaluation: The Code requires the removal of all insulation from pressure-retaining bolted connections in systems borated for the purpose of controlling reactivity when performing VT-2 visual examinations during system pressure tests. As an alternative, the licensee has proposed a two-part approach:

1) Perform the VT-2 visual examinations required by Code Tables IWB- and IWC-2500 with the insulation in place. The licensee has committed to a four-hour hold time for insulated bolted components during this examination. Any evidence of leakage will be evaluated in accordance with Code Case N-566-1, *Corrective Action for Leakage Identified at Bolted Connections*; N-566-1 has been unconditionally accepted by the Staff in Regulatory Guide 1.147, Revision 13, June 2003.

2) A separate, direct VT-3 visual examination of bolted connections at cold shutdown with the insulation removed, but without disassembly or pressurizing the system, will be performed during each refueling outage for Class 1 valves, Class 1 flanges, pressurizer man-way bolting, and Class 2 valves. This part of the licensee's alternative is not required for the RPV closure head flange bolts because these are removed and cleaned during each refueling outage. In

addition, the insulation on steam generator primary side man-way bolting, due to ease of replacement at power, is not covered under this alternative and will be examined in accordance with Code.

The licensee's proposed alternative provides a thorough approach to ensuring the leak-tight integrity of systems borated for the purpose of controlling reactivity. First, by performing the leakage test at operating pressure with the insulation in place, any significant leakage will be detected when the leakage either penetrates the insulation, or is detected at joints or low points. Second, by removing the insulation each refueling outage for the subject components, the licensee will be able to detect minor leakage indicated by the presence of boric acid crystals or residue. This two-phase approach will provide reasonable assurance of the continued structural integrity of bolted connections in borated systems.

Based on the above evaluation, it is concluded that the proposed alternative will provide an acceptable level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i), for the fourth ISI interval at Kewaunee.

4.0 CONCLUSIONS

Pacific Northwest National Laboratory has reviewed the licensee's submittal and concludes that the Code examination requirements are impractical for the subject components listed in Requests for Relief RR-1-3 and RR-1-5. Further, reasonable assurance of the leakage or structural integrity of the subject components has been provided by the examinations that are being performed. Therefore, for these requests, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

Based on the information provided in the licensee's submittal, it has been concluded that the alternatives proposed in Requests for Relief RR-1-7, RR-1-8, RR-1-10, and RR-G-2 provide an acceptable level of quality and safety. Therefore, it is recommended that these alternatives be authorized, pursuant to 10 CFR 50.55a(a)(3)(i), for the fourth 10-year inspection interval at Kewaunee.

For Requests for Relief RR-1-1, RR-1-2, RR-1-4, and RR-2-1, it has been shown that compliance with the Code requirements would result in a hardship or unusual difficulty with no compensating increase in quality or safety. The alternatives proposed by the licensee provide reasonable assurance of the continued leakage or structural integrity of the subject components. Therefore, for these requests, it is recommended that the licensee's alternatives be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

By letter dated September 17, 2004 the licensee withdrew Request for Relief RR-1-9.

KEWAUNEE NUCLEAR POWER PLANT
Fourth 10-Year ISI Interval

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	PNNL TLR Sec.	System or Component	Exam. Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Disposition
RR-1-1	3.1	Pressurizer Surge Nozzle	B-D	B3.120	100% of the inner radius section of integrally-cast PZR surge nozzle	Volumetric	Use VT-2 visual during pressure tests	Authorized 10 CFR 50.55a(a)(3)(ii)
RR-1-2	3.2	Auxiliary Spray Piping	B-P	B15.50	100% of pressure-retaining boundary is required to be pressure tested	Visual VT-2	Conduct pressure test of pipe segment between CVC-15 and CVC-16 with valves in normal position	Authorized 10 CFR 50.55a(a)(3)(ii)
RR-1-3	3.3	RPV Supports	B-K	B10.10	100% of integral attachment welds on RPV	Surface	Use volumetric from vessel inner surface	Granted 10 CFR 50.55a(g)(6)(i)
RR-1-4	3.4	RHR and SI Piping	B-P	B15.50 B15.70	100% of pressure-retaining boundary is required to be pressure tested	Visual VT-2	Conduct pressure testing and VT-2 examination at normal system operating pressures	Authorized 10 CFR 50.55a(a)(3)(ii)
RR-1-5	3.5	RPV Flange Leak-off Piping	B-P	B15.50	100% of pressure-retaining boundary is required to be pressure tested	Visual VT-2	Use VT-2 at atmospheric pressure to detect boric acid residue	Granted 10 CFR 50.55a(g)(6)(i)
RR-1-7	3.6	Vessel Nozzles	B-F	Multiple	100% of dissimilar metal nozzle welds in Class 1 vessels	Volumetric and Surface	Use PDI alternative to Appendix VIII, Supplement 10 for ultrasonic qualifications	Authorized 10 CFR 50.55a(a)(3)(i)
RR-1-8	3.7	Piping	B-J C-F-1	Multiple	Pressure retaining circumferential piping welds	Volumetric and Surface	Use PDI alternative to Appendix VIII, Supplements 2 and 10 for ultrasonic qualifications	Authorized 10 CFR 50.55a(a)(3)(i)
RR-1-9	3.8	RPV Nozzles	B-D	B3.90	100% of nozzle-to-vessel weld volumes per Figure IWB-2500-7(a -c)	Volumetric	Use Code Case N-613-1 to define volumes to be examined	Withdrawn
RR-1-10	3.9	RPV Shell Welds	B-A	Multiple	100% of RPV shell and head welds	Volumetric	Use RMSE for flaw-sizing qualification in lieu of statistics	Not required
RR-2-1	3.10	RPV Head Vent Piping	C-H	C7.10	100% of pressure-retaining boundary is required to be pressure tested	Visual VT-2	Perform VT-2 at 380 psig and <200F; also perform VT-2 at atmospheric pressure to detect boric acid residue	Authorized 10 CFR 50.55a(a)(3)(ii)
RR-G-2	3.11	Insulated Bolted Components	B-P C-H	Multiple	Remove insulation during pressure testing to examine 100% of bolted connection exposed surfaces	Visual VT-2	Perform VT-2 visual with insulation in-place and four hour hold time; also perform VT-3 each outage at shutdown after removing insulation	Authorized 10 CFR 50.55a(a)(3)(i)