Attachment 2

INEL-95/0400

Technical Evaluation Report on the Second 10-year Interval Inservice Inspection Program Plan: Washington Public Power Supply System, WPPSS Nuclear Project, Unit 2, Docket Number 50-397

M. T. Anderson
E. J. Feige
K. W. Hall

Published August 1995

Idaho National Engineering Laboratory
Materials Physics Department
Lockheed Idaho Technologies Company
Idaho Falls, Idaho 83415

Prepared for the
Division of Engineering
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under DOE Idaho Operations Office
Contract DE-AC07-94ID13223
JCN No. J2229 (Task Order A07)

ABSTRACT

This report presents the results of the evaluation of the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted December 27, 1994, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements that the licensee has determined to be impractical. The *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during previous Nuclear Regulatory Commission (NRC) reviews. The requests for relief are evaluated in Section 3 of this report.

This work was funded under:

U.S. Nuclear Regulatory Commission
JCN No. J2229, Task Order A07
Technical Assistance in Support
of the NRC Inservice Inspection Program

SUMMARY

The licensee, Washington Public Power Supply System, has prepared the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, to meet the requirements of the 1989 Edition of the *American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code*, Section XI. The second 10-year interval began February 10, 1995 and ends February 9, 2005.

The information in the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted December 27, 1994, was reviewed. Included in the review were the requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical. As a result of this review, a conference call was conducted to discuss the information and/or clarification required from the licensee in order to complete the review. The licensee provided the requested information in the submittal dated May 12, 1995.

Based on the review of the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the licensee's response to the Nuclear Regulatory Commission's request for additional information, and the recommendations for granting relief from the ISI examinations that cannot be performed to the extent required by Section XI of the ASME Code, no deviations from regulatory requirements or commitments were identified in the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, with the exception of Request for Relief 2ISI-01 (Part 1).

CONTENTS

TECHNICAL EVALUATION REPORT ON THE
SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN:
WASHINGTON PUBLIC POWER SUPPLY SYSTEM,
WPPSS NUCLEAR PROJECT, UNIT 2,
DOCKET NUMBER 50-397

1. INTRODUCTION

Throughout the service life of a water-cooled nuclear power facility, 10 CFR 50.55a(g)(4) (Reference 1) requires that components (including supports) that are classified as American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Class 1, Class 2, and Class 3 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 of Nuclear Power Plant Components* (Reference 2), to the extent practical within the limitations of design, geometry, and materials of construction of the components. This section of the regulations also requires that inservice examinations of components and system pressure tests conducted during successive 120-month inspection intervals comply with the requirements in the latest edition and addenda of the Code incorporated by reference in 10 CFR 50.55a(b) on the date 12 months prior to the start of the 120-month inspection interval, subject to the limitations and modifications listed therein. The components (including supports) may meet requirements set forth in subsequent editions and addenda of this Code that are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein, and subject to Nuclear Regulatory Commission (NRC) approval. The licensee, Washington Public Power Supply System, has prepared the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, (Reference 3), to meet the requirements of the 1989 Edition of the ASME Code, Section XI. The second 10-year interval began February 10, 1995 and ends February 9, 2005.

As required by 10 CFR 50.55a(g)(5), if the licensee determines that certain Code examination requirements are impractical and requests relief from them, the licensee shall submit information and justification to the NRC to support that determination.

Pursuant to 10 CFR 50.55a(g)(6), the NRC will evaluate the licensee's determination that Code requirements are impractical to implement. The NRC may grant relief and may impose alternative requirements that are determined to be authorized by law, will not endanger life, property, or the common defense and security, and are 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.

Alternatively, pursuant to 10 CFR 50.55a(a)(3), the NRC will evaluate the licensee's determination that either (i) the proposed alternatives provide an acceptable level of quality and safety, or (ii) Code compliance would result in hardship or unusual difficulty without a compensating increase in safety. Proposed alternatives may be used when authorized by the NRC.

The information in the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted December 27, 1994, was reviewed, including the requests for relief from the ASME Code Section XI requirements that the licensee has determined to be impractical. The review of the Inservice Inspection (ISI) Program Plan was performed using the Standard Review Plans of NUREG-0800 (Reference 4), Section 5.2.4, "*Reactor Coolant Boundary Inservice Inspections and Testing*," and Section 6.6, "*Inservice Inspection of Class 2 and 3 Components*."

During a conference call that was held May 11, 1995, the NRC requested additional information that was required in order to complete the review of the ISI Program Plan. The requested information was provided by the licensee in a submittal dated May 12, 1995 (Reference 5).

The *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during the NRC's previous reviews.

The requests for relief are evaluated in Section 3 of this report. Unless otherwise stated, references to the Code refer to the ASME Code, Section XI, 1989 Edition. Specific inservice test (IST) programs for pumps, valves, and snubbers are being evaluated in other reports. Request for Relief 2ISI-11 and Section 6.2.2 of *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan* concern Component Supports Operability Assurance of Snubbers. Therefore, they are will not be evaluated as part of this report.

2. EVALUATION OF INSERVICE INSPECTION PROGRAM PLAN

This evaluation consists of a review of the applicable program documents to determine whether or not they are in compliance with the Code requirements and any previous license conditions pertinent to ISI activities. This section describes the submittals reviewed and the results of the review.

2.1 Documents Evaluated

Review has been completed on the following information from the licensee:

(a) *WNP-2, Second 10-Year Inservice Inspection Program Plan*, Revision 0, dated December 27, 1994;

(b) *WNP-2, Operating License NPF-21 Second 10-Year Inservice Inspection and Testing (ISI/IST) Program Implementation*, dated February 15, 1995; and

(c) *WNP-2, Operating License NPF-21 Second 10-Year Inservice Inspection Program Plan*, dated May 11, 1995.

2.2 Compliance with Code Requirements

2.2.1 Compliance with Applicable Code Editions

The Inservice Inspection Program Plan shall be based on the Code editions defined in 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Based on the starting date of February 10, 1995, the Code applicable to the second interval ISI program is the 1989 Edition. As stated in Section 1 of this report, the licensee has prepared the *WPPSS Nuclear Project, Unit 2 Second 10-Year Inservice Inspection Program Plan* to meet the requirements of 1989 Edition.

2.2.2 Acceptability of the Examination Sample

Inservice volumetric, surface, and visual examinations shall be performed on ASME Code Class 1, 2, and 3 components and their supports using sampling schedules described in Section XI of the ASME Code and 10 CFR 50.55a(b). Sample size and weld selection have been implemented in accordance with the Code and 10 CFR 50.55a(b) and appear to be correct.

2.2.3 Exemption Criteria

The criteria used to exempt components from examination shall be consistent with Paragraphs IWB-1220, IWC-1220, IWC-1230, IWD-1220, and 10 CFR 50.55a(b). The exemption criteria have been applied by the licensee in accordance with the Code, as discussed in the ISI Program Plan, and appear to be correct.

2.2.4 Augmented Examination Commitments

In addition to the requirements specified in Section XI of the ASME Code, the licensee has committed to perform the following augmented examinations:

(a) NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking* (Reference 6);

(b) NRC Generic Letter 88-01, *Intergranular Stress Corrosion Cracking* (Reference 7);

(c) NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations* (Reference 8);

(d) IE Bulletin 80-13, *Cracking in Core Spray Spargers* (Reference 9);

(e) NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors* (Reference 10);

(f) Supply System Letter G02-90-024, *Request for Amendment to Technical Specifications, Final Feedwater Temperature Reduction (FFTR)* (Reference 11); and

(g) FSAR Section 3.6.2.1.2.1(a), *Augmented High Energy Piping Examination* (Reference 12).

2.3 Conclusion

Based on the review of the documents listed above, no deviations from regulatory requirements or commitments were identified in the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, with the exception of Request for Relief 2ISI-01 (Part 1).

3. EVALUATION OF RELIEF REQUESTS

The requests for relief from the ASME Code requirements that the licensee has determined to be impractical for the second 10-year inspection interval are evaluated in the following sections.

3.1 Class 1 Components

3.1.1 Reactor Pressure Vessel

3.1.1.1 Request for Relief 2ISI-01 (PART 1), Examination Category B-A, Item B1.11, Reactor Pressure Vessel Shell Welds

Code Requirement: Examination Category B-A, Item B1.11, requires a volumetric examination of 100% of reactor pressure vessel (RPV) circumferential welds and adjacent base metal as defined in Figure IWB-2500-1.

Licensee's Code Relief Request: The licensee requested relief from performing the complete Code-required volumetric examinations for the following reactor pressure vessel circumferential welds:

TABLE 5.1.2.1*
Examination Coverage

Weld #	Description	Drawing #	% Examinable**	Notes
AB	#1-#2 SC CRC WD	RPV-101	79%	1
AD	#3-#4 SC CRC WD	RPV-101	83%	2

Notes:
1. Examination coverage limited by weld repair.
2. Examination coverage limited by RPV stabilizer lugs.

* Table from licensee's submittal
** Examination coverage obtained during first inspection interval

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

"Relief is requested from ASME Section XI examination requirements on the basis of partial inaccessibility of the welds due to plant design. Design of the Reactor Pressure Vessel and biological shield wall were completed prior to promulgation of amendments to 10 CFR 50.55a. The design limits access to less than 100% of these welds. Table 5.1.2.1 illustrates the coverage obtained during the first inspection interval and reasons full Code coverage was not obtained.

"There will be no adverse impact on plant quality and safety by doing only a partial Code examination of these welds.

"1. The Class 1 RPV welds have passed radiographic, magnetic particle and ultrasonic examinations in accordance with ASME Section III requirements.

"2. Examinations of the RPV welds during the first inspection interval did not reveal any unacceptable indications.

"3. The identified welds will be subject to a system pressure test in accordance with ASME Section XI Code Case N-498 requirements.

"4. Leak detection systems identify significant leakage in the areas of the subject welds. Appropriate operator action would occur due to leak detection system alarms.

"5. Other similar welds in the vessel will receive full Code examinations. During the first inspection interval 84% of the total weld volume in the RPV received a complete Code volume examination. No unacceptable indications were found. Table 5.1.2.2[a] presents the coverage obtained during the first inspection interval.

"6. The welds examined represent a large and representative sample of the reactor pressure vessel. The integrity of the RPV can be verified by examination of this sample."

Licensee's Proposed Alternative Examination (as stated):

"Each weld will be examined per Section XI requirements to the extent defined in Table 5.1.2.1 above."

Evaluation: This issue was also evaluated for the first ten year inservice inspection interval. By letter dated June 24, 1993 the licensee submitted revised Request for Relief 2-ISI-001, which

---

a. This table was provided in the licensee's submittal but is not included in this Technical Evaluation Report.

addressed the augmented reactor vessel shell welds examinations pursuant to 10 CFR 50.55a(6)(ii)(A). In a meeting at NRC Headquarters on November 2, 1994, the licensee indicated that in the near future it will submit an alternative to the augmented requirements. The licensee also indicated that this issue would be resolved prior to the end of the first period of the second 10-year inservice inspection interval.

Performing the examinations to the extent defined in Table 5.1.2.1 above is the same alternative that was denied in an NRC safety evaluation report dated December 9, 1994. Therefore, this issue has not been resolved.

<u>Conclusion</u>: Until the commitment to submit an alternative to the reactor vessel shell welds augmented examinations pursuant to 10 CFR 50.55a(6)(ii)(A) has been met, this request for relief cannot be evaluated. Therefore, it is recommended that relief be denied.

3.1.1.2 <u>Request for Relief 2ISI-01 (PART 2), Examination Category B-A, Items B1.21, B1.22, and B1.30, Reactor Vessel Head and Shell-to-Flange Welds</u>

<u>Code Requirement</u>: Examination Category B-A, Items B1.21, B1.22, and B1.30 require a volumetric examination of 100% of the Reactor Vessel Head and Shell-to-Flange Welds and adjacent base metal as defined in Figure IWB-2500-3 or -4.

<u>Licensee's Code Relief Request</u>: The licensee requested relief from performing the complete Code-required volumetric examinations for the following Reactor Vessel Head and Shell-to-Flange Welds:

TABLE 5.1.2.1*
Examination Coverage

Weld #	Description	Drawing #	% Examinable**	Notes
AE	#4 SC-FL CRC WD	RPV-101	49%	3
DA	BTM HD MRD at 272°	RPV-102	78%	4
DB	BTM HD MRD at 332°	RPV-102	78%	4
DC	BTM HD MRD at 32°	RPV-102	78%	4
DD	BTM HD MRD at 92°	RPV-102	78%	4
DE	BTM HD MRD at 152°	RPV-102	78%	4
DF	BTM HD MRD at 212°	RPV-102	78%	4
DG	BOT HD MRD at 270°	RPV-102	17%	5
DR	BOT HD MRD at 90°	RPV-102	17%	5

Notes:

3. Design of flange limits examination to one side.
4. Only 21"starting from the intersection of weld AA and 14" starting from the intersection of weld AJ can be examined due to the vessel skirt. (Approximately one foot is not being examined on each weld).
5. Only 12" to 23" on each end of the weld, starting from the intersection of weld AJ, can be examined due to CRD penetrations and housings.

* Table from licensee's submittal
** Examination coverage obtained during first inspection interval

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

"Relief is requested from ASME Section XI examination requirements on the basis of partial inaccessibility of the welds due to plant design. Design of the Reactor Pressure Vessel and biological shield wall were completed prior to promulgation of amendments to 10CFR50.55a. The design limits access to less than 100% of these welds. Table 5.1.2.1 illustrates the coverage obtained during the first inspection interval and reasons full Code coverage was not obtained.

"There will be no adverse impact on plant quality and safety by doing only a partial Code examination of these welds.

"1. The Class 1 RPV welds have passed radiographic, magnetic particle and ultrasonic examinations in accordance with ASME Section III requirements.

"2. Examinations of the RPV welds during the first inspection interval did not reveal any unacceptable indications.

"3. The identified welds will be subject to a system pressure test in accordance with ASME Section XI Code Case N-498 requirements.

"4. Leak detection systems identify significant leakage in the areas of the subject welds. Appropriate operator action would occur due to leak detection system alarms.

"5. Other similar welds in the vessel will receive full Code examinations. During the first inspection interval 84% of the total weld volume in the RPV received a complete Code volume examination. No unacceptable indications were found. Table 5.1.2.2[b] presents the coverage obtained during the first inspection interval.

"6. The welds examined represent a large and representative sample of the reactor pressure vessel. The integrity of the RPV can be verified by examination of this sample."

Licensee's Proposed Alternative Examination (as stated):

"Each weld will be examined per Section XI requirements to the extent defined in table 5.1.2.1 above."

Evaluation: The Code requires that 100% of the volume, as defined in Figure IWB-2500-3 or -4, of all Reactor Pressure Vessel Head and Shell-to-Flange Welds be examined during each interval. The licensee submitted drawings[c] that depict cross-sectional geometry and Table 5.1.2.1, which provides examination percentages that can be achieved for the subject welds. Based on restricted access to the area, the INEL staff has determined that it is impractical for the licensee to completely examine the required weld volumes. A considerable burden would result if the licensee was required to redesign and replace the vessel for the sole purpose of increasing volumetric examination coverage.

The licensee stated that at least 84% of the total reactor pressure vessel weld volume could be examined. While the limited volumetric examination does not meet Code requirements for each individual weld, a significant percentage of the vessel has been

b. This table was provided in the licensee's submittal but is not included in this Technical Evaluation Report.

c. Not included in this Technical Evaluation Report.

inspected, which provides reasonable assurance of detection of a pattern of degradation that, if present, might impact the overall structural integrity of the vessel.

Conclusion: Based on the restricted access to the examination area, which limits volumetric examination of the subject welds, and the significant examination coverage which can be achieved for the entire vessel, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3.1.1.3 Request for Relief 2ISI-02, Examination Category B-D, Item B3.90, Reactor Vessel Nozzle-to-Vessel Welds

Code Requirement: Examination Category B-D, Item B3.90, requires a volumetric examination of 100% of the Reactor Vessel Nozzle-to-Vessel Welds and adjacent base metal as defined in Figure IWB-2500-7(b).

Licensee's Code Relief Request: The licensee requested relief from performing the complete Code-required volumetric examinations for the following Reactor Vessel Nozzle-to-Vessel Welds):

TABLE 5.1.2.3*
Examination Coverage

Weld #	Description	Drawing #	% Examined** 45 Degree	% Examined** 60 Degree
N1-0	RRC NZ-V @ 0	RPV-101	75%	81%
N1-180	RRC NZ-V @ 180	RPV-101	75%	81%
N2-120	RRC NZ-V @ 120	RPV-101	75%	81%
N2-150	RRC NZ-V @ 150	RPV-101	75%	81%
N2-210	RRC NZ-V @ 210	RPV-101	75%	81%
N2-240	RRC NZ-V @ 240	RPV-101	75%	81%
N2-270	RRC NZ-V @ 270	RPV-101	75%	81%
N2-30	RRC NZ-V @ 30	RPV-101	75%	81%
N2-300	RRC NZ-V @ 300	RPV-101	75%	81%
N2-330	RRC NZ-V @ 330	RPV-101	75%	81%
N2-60	RRC NZ-V @ 60	RPV-101	75%	81%
N2-90	RRC NZ-V @ 90	RPV-101	75%	81%
N3-108	MS NZ-V @ 108	RPV-101	86%	90%
N3-252	MS NZ-V @ 252	RPV-101	86%	90%
N3-288	MS NZ-V @ 288	RPV-101	86%	90%
N3-72	MS NZ-V @ 72	RPV-101	86%	90%
N4-150	FW NZ-V @ 150	RPV-101	71%	79%

Weld #	Description	Drawing #	% Examined** 45 Degree	% Examined** 60 Degree
N4-210	FW NZ-V @ 210	RPV-101	71%	79%
N4-270	FW NZ-V @ 270	RPV-101	71%	79%
N4-30	FW NZ-V @ 30	RPV-101	71%	79%
N4-330	FW NZ-V @ 330	RPV-101	71%	79%
N4-90	FW NZ-V @ 90	RPV-101	71%	79%
N5-120	LPCS NZ-V @ 120	RPV-101	86%	90%
N6-135	LPCS NZ-V @ 135	RPV-101	72%	79%
N6-345	LPCS NZ-V @ 345	RPV-101	72%	79%
N6-45	LPCS NZ-V @ 45	RPV-101	72%	79%
N16-240	HPCS NZ-V @ 240	RPV-101	72%	80%

* Table from licensee's submittal
** Examination coverage obtained during first inspection interval

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

"Relief is requested from ASME Section XI examination requirements on the basis of partial inaccessibility of the welds due to configuration. The design of the vessel to nozzle weld prevents examination of 100% of the volume defined in Figure IWB-2500-7(b) with equipment available.

"There will be no adverse impact on plant quality and safety by doing only a partial Code examination of these welds.

"1. The Class 1 RPV welds have passed radiographic, magnetic particle and ultrasonic examinations in accordance with Section III.

"2. No unacceptable indications were found during the first inspection interval examinations.

"3. The identified welds will be subject to a system pressure test in accordance with ASME Section XI Code Case N-498 requirements.

"4. Leak detection systems identify significant leakage in the areas of the subject welds. Appropriate operator action would occur due to leak detection system alarms.

"5. No automatic inspection system in use today can effectively examine 100% of the required Code volume. Additional manual examinations will not significantly increase, if at all, the volume examined.

"6. The achievable coverage will detect flaws in the inner volume where they are most likely to occur.

"7. The percent of achievable examination volume is significant and representative of the item B3.90 welds. RPV nozzle weld integrity will be ensured by completing the percent of the welds defined in Table 5.1.2.3."

Licensee's Proposed Alternative Examination (as stated):

"Each weld will be examined per Section XI requirements to the extent defined in table 5.1.2.3."

Evaluation: The Code requires that 100% of the volume of all RPV nozzle-to-vessel welds be examined during each interval. The licensee submitted drawings[d] that depict cross-sectional geometry, and Table 5.1.2.3, which provides examination percentages that can be achieved, for the reactor pressure vessel nozzle-to-vessel welds at WNP-2. Based on the geometry of the welds, the INEL staff has determined that it is impractical for the licensee to completely examine the Code-required volumes. A considerable burden would result if the licensee was required to redesign and replace the nozzles for the sole purpose of increasing volumetric examination coverage.

The licensee has stated that a significant percentage of the Code-required examination volumes (see Table above) can be examined. While the limited volumetric examination does not meet Code requirements, the significant volume that can be examined provides reasonable assurance of detection of a pattern of degradation that, if present, might impact the overall structural integrity of the weld.

Conclusion: Based on the impracticality of 100% examination due to configuration of the nozzles, which limits volumetric coverage, and the significant coverage that can be achieved, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

---

d. Not included in this Technical Evaluation Report.

3.1.1.4 Request for Relief 2ISI-03, Examination Category B-G-1, Item B6.10, Reactor Vessel Closure Head Nuts

Code Requirement: Examination Category B-G-1, Item B6.10, requires a surface examination of the reactor vessel closure head nuts.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required surface examination of Reactor Vessel Closure Head Nuts 36-1-1A through 36-1-76A.

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

"A meaningful surface examination of the thread area cannot be achieved with the protective phosphate coating.

"This alternative examination of the RPV nuts provides for a more thorough examination. This examination technique was used during the first inspection interval. During the first inspection interval no unacceptable indications were found."

Licensee's Proposed Alternative Examination (as stated):

"A volumetric (ultrasonic) examination of the nut will be performed to augment the surface examination. The ultrasonic examination will consist of a L-wave from the end and shear wave in four directions (two parallel to axis and two perpendicular to axis). A spare RPV nut will be used for the calibration."

Evaluation: The Code requires a surface examination of the reactor vessel closure head nuts. However, a protective phosphate coating prevents a surface examination of the threaded area. To perform the Code-required surface examination, removal of the protective coating would be required.

The licensee's proposed comprehensive volumetric examination of the reactor vessel closure head nuts should provide a thorough examination of the area. Therefore, a pattern of degradation, if present, should be detected.

Conclusion: The licensee's proposed volumetric examination of the reactor vessel closure head nuts in lieu of the surface examination should provide an acceptable level of quality. Therefore, it is recommended that this alternative examination be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

3.1.2 Pressurizer (Does not apply to BWRs)

3.1.3 Heat Exchangers and Steam Generators (No requests for relief)

3.1.4 Piping Pressure Boundary

3.1.4.1 Request for Relief 2ISI-08, Examination Category B-K-1, Item B10.10, Integrally Welded Attachments to Class 1 Piping

Code Requirement: Examination Category B-K-1, Item B10.10, requires a surface examination of all integrally welded attachments associated with piping examined per Examination Category B-J as defined by Figures IWB-2500-13, -14, and -15.

Licensee's Code Relief Request: The licensee requested relief from performing 100% of the Code-required surface examination of the following integrally welded attachments:

Identification No.	Description	ISI Diagram No.
RRC-HA-1(W)	4 Welded Lugs	RRC-101-1
RRC-HB-1(W)	4 Welded Lugs	RRC-102-1

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

"Relief is requested from ASME Section XI examination requirements for the two item B10.10 welds on the basis of high dose required to prepare for and perform the examinations. The welds identified in this relief request require disassembly of a component support collar to gain access to perform a 100% Code examination. Radiation dose rates in the area of these welds are 400mR/hr. To access the remaining portion of these welds will require the disassembly of the collars preventing full code coverage. The component supports associated with these welds are twin variable springs supporting a 24 inch diameter pipe. The support has a vertical load of approximately 24,000 pounds. To

access the welds the piping being supported by these springs will need to be temporarily supported so the collars can be disassembled. After the examination the collars will be reassembled, temporary supports removed, and the variable springs rebalanced. It is anticipated that total exposure to examine the remaining portions of attachment welds RRC-HA-1(W) and RRC-HB-1(W) will exceed 4 person rem.

"There will be no adverse impact on plant quality and safety by doing only a partial Code examination of the item B10.10 welds.

"1. During the first inspection interval, weld RRC-HA-1(W) received a dye penetrant examination of 75% of the required examination surface and weld RRC-HB-1(W) received a dye penetrant examination of 50% of the required surface. No unacceptable indications were found.

"2. The percent of Examination Category B-K-1 welds examined exceeds 95% of total welds in this category. No unacceptable indications were found in category B-K-1 welds during the first inspection interval.

"3. Other Examination Category B-K-1 welds will receive full Code surface examination coverage. There is a total of forty-two (42) attachment welds in Examination Category B-K-1. The integrity of the Class 1 attachment welds is verified by performing examinations on this large representative sample."

Licensee's Proposed Alternative Examination (as stated):

"The accessible portion of the item B10.10 attachment welds RRC-HA-1(W) and RRC-HB-1(W), without removing the component support, will be examined per Section XI requirements."

Evaluation: The Code requires surface examination of 100% of the selected integrally welded attachments. The licensee requested relief to perform partial examinations on Attachment Welds RRC-HA-1(W) and RRC-HB-1(W) due to limited access.

Access is limited by a component support consisting of twin variable springs supporting a 24 inch diameter pipe. The support carries a vertical load of approximately 24,000 pounds. To gain access to perform 100% of the Code examination, disassembly of the component support collar would be required. Removal of the support collar is complicated by an anticipated high radiation field. To meet the Code requirements, the piping run would have

to be replaced or the welded attachment redesigned, causing an undue burden on the licensee without a compensating increase in safety.

Attachment Weld RRC-HA-1(W) will receive a dye penetrant examination of 75% of the required surface and Attachment Weld RRC-HB-1(W) will receive a dye penetrant examination of 50% of the required surface. In addition, 95% of the total attachment welds in this category will be examined this interval. Therefore, a pattern of degradation, if present, should be detected.

Conclusion: Requiring the licensee to perform 100% of the Code required examination would result in an undue hardship without compensating increase in safety. The licensee's proposed partial examination of the subject welded attachments and the examination of other welded attachments should provide reasonable assurance of operational readiness. Therefore, it is recommended that the alternative be authorized, pursuant to 10 CFR 50.55a(a)(3)(ii).

3.1.5 Pump Pressure Boundary (No requests for relief)

3.1.6 Valve Pressure Boundary (No requests for relief)

3.1.7 General (No requests for relief)

3.2 Class 2 Components

3.2.1 Pressure Vessels (No requests for relief)

3.2.2 Piping

3.2.2.1 Request for Relief 2ISI-09, Examination Category C-C, Item C3.20, Integrally Welded Attachments for Class 2 Piping

Code Requirement: Examination Category C-C, Item C3.20, requires a surface examination of the integrally welded attachments on piping examined per Examination Categories C-F and C-G, as defined in Figure IWC-2500-5.

Licensee's Code Relief Request: The licensee requested relief from performing 100% of the Code-required surface examination of the following integrally welded attachments:

Identification No.	Description	ISI Diagram No.
RHR-77(W)	Welded Attachment	RHR-205-1
RHR-410(W)	Welded Attachment	RHR-203-2

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

"Relief is requested from ASME Section XI examination requirements for the two item C3.20 welds on the basis of inaccessibility of the welds due to their location in separate pipe chases where access will place a hardship on the plant to gain access. Access is gained by removing the steel and lead brick shielding walls.

"There will be no adverse impact on plant quality and safety by not doing a Code examination of two (2) item C3.20 welds.

"1. The percent of Category C-C welds examined exceeds 96% of total welds in this category. No unacceptable indications were found in Category C-C during the first inspection interval.

"2. Other similar welds in this system and in the area of the subject welds have or will receive full Code surface examination coverage. The total number of item C3.20 welds is fifty-five (55). Performing surface examination of 53 of 55 welded attachments in Item C3.20 represents a significantly large and representative sample of this type weld. The integrity of the attachment welds is verified by sampling."

Licensee's Proposed Alternative Examination (as stated):

"No alternate examination is proposed for these item C3.20 welds."

Evaluation: The Code requires 100% of the Class 2 integrally welded attachments to be examined per their respective examination category each inspection interval.

The licensee requested relief from performing the Code-required examination on Welded Attachments RHR-77(W) and RHR-410(W) due to their inaccessibility, which is caused by their location in pipe chases. Meeting the Code requirements would require removing the steel and lead brick shielding walls, causing an undue burden on the licensee.

The licensee is examining 53 of 55 integrally welded attachments, which comprises 96% of this Class 2 category. Since a significant number of attachments will be examined, a pattern of degradation, if present, should be detected.

Conclusion: Requiring the licensee to perform 100% of the Code required examination would result in an undue hardship without compensating increase in safety. Reasonable assurance of operational readiness will be maintained by the examinations that are to be performed. Therefore, it is recommended that the alternative be authorized, pursuant to 10 CFR 50.55a(a)(3)(ii).

3.2.3 Pumps (No requests for relief)

3.2.4 Valves (No requests for relief)

3.2.5 General (No requests for relief)

3.3 Class 3 Components

3.3.1 Piping

3.3.1.1 Request for Relief 2ISI-10, Examination Category D-B, Item D2.20, Class 3 Integrally Welded Attachments

Code Requirement: Examination Category D-B, Item D2.20, requires a surface examination of the integrally welded attachments associated with pipe supports selected per IWF-2510(b), as defined in Figure IWF-1300-1.

Licensee's Code Relief Request: The licensee requested relief from performing 100% of the Code-required surface examination of the following integrally welded attachments:

Identification No.	Description	ISI Diagram No.
SW-90(W)	Welded Attachment	SW-307
SW-123(W)	Welded Attachment	SW-301
SW-439(W)	Welded Attachment	SW-303
SW-946N(W)	Welded Attachment	SW-314
SW-951N(W)	Welded Attachment	SW-315

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

"The welded attachments are completely or partially inaccessible to examination. The welded attachments are within fire barriers or enclosed in cubicles or pipe chases.

"There will be no adverse impact on plant quality and safety. During the first inspection period the following percent of items were examined:

Examination Category	Percent Examined	Number Examined
D-B	>96%	141

"No unacceptable indications were found during these examinations.

"The class 3 welded attachments in category D-B examined during the first inspection interval represent greater than 96% of the total welded attachments in this category.

"The sample size in this category is reasonably large and representative and assures continued plant quality and safety."

Licensee's Proposed Alternative Examination (as stated):

"The welded attachments are completely or partially inaccessible to all examination techniques. No alternate examinations are proposed."

Evaluation: The Code requires 100% of the Class 3 integrally welded attachments to be examined each inspection interval. The licensee requests relief from performing the Code-required examination on Welded Attachments SW-90(W), SW-123(W), SW-439(W), SW-946N(W), and SW-951N(W) due to their inaccessibility.

The subject welded attachments are completely or partially inaccessible to examination. These attachments are within or close to wall penetrations or enclosed in cubicles or pipe chases. Attachments within wall penetrations are embedded in the concrete, and the annulus between the pipe and concrete is foam-filled. Therefore, the subject component examinations are impractical to perform to the extent required by the Code. To meet the Code requirements, the piping systems would have to be redesigned; imposition of the requirements would cause a considerable burden on the licensee.

Greater than 96% of Code-required examination of Item D-B integrally welded attachments will be performed. Since a significant number of these examinations will occur, a pattern of degradation, if present, should be detected.

Conclusion: Reasonable assurance of operational readiness will be maintained by the examinations that will be performed and, considering the impracticality of meeting the Code requirements, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3.3.2 Pumps (No requests for relief)

3.3.3 Valves (No requests for relief)

3.3.4 General (No requests for relief)

3.4 Pressure Tests

3.4.1 Class 1 System Pressure Tests

3.4.1.1 Request for Relief 2ISI-06, IWA-5250(a)(2) Corrective Action for Leaking CRD Flanges

Code Requirement: IWA-5250(a)(2) requires bolting at leaking connections to be removed and a VT-3 visual examination performed for corrosion.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required removal and VT-3 visual examination of bolting at leaking CRD flanges.

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

"Relief is requested from the corrective action of [IWA-5250(a)(2)] based on the hardship to remove the cap screws from these flanges at the end of a refueling outage. The pressure test is performed just prior to startup when the equipment required for cap screw removal has been removed from under the RPV. To remove all the bolting from a leaking drive will require depressurization of the RPV and reassembly of the equipment necessary to remove the bolting. The dose rates under the RPV in the area where bolting removal will occur are estimated to be between 200 mRem/hr at the knees and 600 mRem/hr at the head.

"Approximately 30-40 CRD flanges are disassembled for CRD exchange every 2 years. During this activity all the CRD cap screws from the disassembled flanges are required to be VT-1 visually examined in accordance with Examination Category B-G-2, item number B7.80. The provisions of this examination are to determine if degradation of the bolting has occurred. All the CRD flanges are disassembled and the cap screws examined within an approximate 10 year cycle.

"During the first 10 year inservice inspection interval, 187 sets of CRD cap screws received a VT-1 visual examination after removal. All new cap screws receive a Preservice Inspection VT-1 examination prior to installation. Corrosion pitting has been observed on the cap screw's shank. Metallurgical evaluation of the worst case from two outages was performed. The analysis

concluded that the pitting was shallow and the ASME acceptance standards were not exceeded. No cap screws that exhibited pitting corrosion were reused. This demonstrates that the VT-1 examinations required by Examination Category B-G-2, item number B7.80 provide an effective method to detect the early signs of bolting degradation.

"WNP-2 is a Boiling Water Reactor (BWR) that does not utilize a process with a borated water chemistry, except the standby liquid control system, like a Pressurized Water Reactor (PWR). The corrosiveness of leaking borated water in a PWR is known but is not applicable to WNP-2. The alternatives described provide assurance that corrosive conditions are evaluated.

"There will be no adverse impact on plant quality and safety by implementing the alternative measures. The alternative examination to perform VT-1 examination on removed bolting provides a superior method of detecting bolting degradation than that required by IWA-5250(a)(2). The sample size (every cap screw within an approximate 10 year period) and frequency (approximately 16-20% of the cap screws examined every two years) provides assurance that bolting degradation will be detected."

Licensee's Proposed Alternative Examination (as stated):

"When control rod drives are exchanged, perform a VT-1 visual examination on the removed bolting. This VT-1 examination will be in accordance with Table-2500-1 Examination Category B-G-2, Item Number B7.80."

Evaluation: The Code requires that all bolting at leaking mechanical connections be removed and VT-3 visually examined. As an alternative, the licensee proposes to perform a VT-1 visual examination in accordance with Examination Category B-G-2, Item B7.80, on the removed bolting when control rod drives are exchanged. Since the pressure test is performed just prior to startup, the equipment required for cap screw removal has been removed from under the reactor pressure vessel. Removing all the bolting from a leaking joint will require depressurization of the RPV and reassembly of the equipment necessary to remove bolting. This will result in additional radiation exposure to personnel under the RPV.

The licensee's experience indicates that approximately 30-40 CRD flanges are disassembled for CRD exchange every 2 years. During this activity, all the CRD cap screws from the disassembled

flanges are required to be VT-1 visually examined in accordance with Examination Category B-G-2, Item B7.80 to determine if degradation of the bolting has occurred. All the CRD flanges are disassembled and the cap screws examined within an approximate 10-year cycle.

The WNP-2 data indicate that during the first 10-year inservice inspection interval, 187 sets of CRD screws received a VT-1 visual examination after removal. All new caps have received a VT-1 visual examination prior to installation.

Corrosion pitting has been observed on the cap screw's shank. Metallurgical evaluation of the worst cases from two outages was performed. The analysis concluded that the pitting was shallow and the ASME acceptance standards were not exceeded. No cap screws that exhibited pitting corrosion were reused. No failed bolting was found during the first 10-year inspection interval. This demonstrates that the VT-1 examinations required by Examination Category B-G-2, Item B7.80 provide an effective method to detect the early signs of bolting degradation.

Conclusion: The licensee's alternative proposal, to examine bolting when control rod drives are exchanged, will provide a reasonable assurance of operational readiness. Imposing the Code requirement to remove all the bolting to perform VT-3 visual examination will impose undue hardship on the licensee without a compensating additional level of safety. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

3.4.2 Class 2 System Pressure Tests (No requests for relief)

3.4.3 Class 3 System Pressure Tests

3.4.3.1 Request for Relief 2ISI-04, IWA-5244, Buried Components in the Service Water System

Code Requirement: IWA-5244 requires a test that determines the change in flow between the ends of the buried nonisolatable components in redundant systems.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required test to determine flow changes between the ends of buried service water (SW) system piping.

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

"The design of the piping in the service water pump houses prevents direct flow monitoring. Figures 5.1.2.5 and 5.1.2.6[e] show the dimensions between the pump, valves and elbow. The close proximity of these items does not allow sufficient stable flow required for meaningful flow measurement. The direct measurement of flow at this end of the buried piping is impossible.

"There will be no decrease in plant quality and safety by performing the alternate examinations. Per the Pump Inservice Test Program, the pumps in both SW Loops A and B are tested quarterly to verify that they are operating correctly and providing adequate flow. Per Section XI IWA-5244 verification of adequate flow is an acceptable test to perform as the VT-2 visual examination for buried piping. In addition to being recognized by the Code, the alternate examination would be performed more frequently, quarterly versus once per inspection period. Based on more frequent testing, Code acceptability and system function, the performance of the alternate examination will not decrease plant quality or safety."

Licensee's Proposed Alternative Examination (as stated):

"In place of the Code required flow test, WNP-2 will verify that the flow during operation is adequate to perform the systems required function. This will be accomplished by verifying the

e. Not included as part of this report.

flow and pump discharge pressure is within the acceptable range per the last pump and valve surveillance. In addition, each inspection period the area between the pump house and reactor building where the buried piping runs will be observed for anomalies or disturbances which may indicate a leak."

Evaluation: As an alternative to a VT-2 visual examination, the Code requires a flow test to determine if a difference in flow exists on different ends of buried non-isolatable components. Portions of the Class 3 piping in the service water system are buried. The design of the service water system does not provide access to the buried piping to perform the Code-required visual examination during hydrostatic testing. In addition, the system design did not allow space to install a flow detector, thereby making the Code-required examination impractical to perform. To perform the Code-required visual examination during the hydrostatic test/flow measurement, the service water system would require design modifications, causing a burden on the licensee.

Each period the licensee will verify that the system flow and pump discharge pressure are within the acceptable range. This, in conjunction with an inspection of the area between the pump house and reactor building where the piping is buried for anomalies or disturbances that may indicate a leak, should provide reasonable assurance of operational readiness.

Conclusion: The Code-required flow test is impractical to perform on the buried service water piping between the pump house and the reactor building at WNP-2. The licensee's alternative should provide reasonable assurance of operational readiness. Therefore, it is recommended that relief be granted, pursuant to 10 CFR 50.55a(g)(6)(i).

3.4.3.2 Request for Relief 2ISI-05, IWD-5223(f), Pressure Testing Piping Downstream of Main Steam Relief Valves

Code Requirement: For safety or relief valve piping that discharges into the containment pressure suppression pool, IWD-5223(f) requires a pneumatic test (at a pressure of 90% of the pipe submergence head of water) to demonstrate leakage integrity in lieu of a system hydrostatic test.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required pneumatic test on the following main steam discharge lines:

Line #	ISI Diagram #
10MS(18)-2-1	MS-301-1, -2, -3
10MS(18)-2-2	MS-302-1, -2, -3
10MS(18)-2-3	MS-303-1, -2, -3
10MS(18)-2-4	MS-304-1, -2, -3
10MS(18)-2-10	MS-305-1, -2, -3
10MS(18)-2-11	MS-306-1, -2, -3
10MS(18)-2-12	MS-307-1, -2, -3
10MS(18)-2-13	MS-308-1, -2, -3
10MS(18)-2-14	MS-309-1, -2, -3
10MS(18)-2-9	MS-310-1, -2, -3
10MS(18)-2-8	MS-311-1, -2, -3
10MS(18)-2-7	MS-312-1, -2, -3
10MS(18)-2-6	MS-313-1, -2, -3
10MS(18)-2-5	MS-314-1, -2, -3
10MS(18)-2-18	MS-315-1, -2, -3
10MS(18)-2-17	MS-316-1, -2, -3
10MS(18)-2-16	MS-317-1, -2, -3
10MS(18)-2-15	MS-318-1, -2, -3

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

"The pressure test at less than two (2) percent of the operating pressure of the system does not add to the public safety. The test adds unnecessarily to the plants radiological exposure burden (approximately 1.6 person rems per outage). This test has been eliminated in the 1992 Edition, 1992 Addenda to Section XI (see page 207 1992 Addenda to ASME Section XI).

"The pressure test required by the Code (6.6 psig) is significantly less than (less than 2%) the operating pressure of the discharge lines (388-465 psig). The test pressure does not assure integrity of the piping and adds unnecessarily to the plants radiological burden.

"Additionally, the Code required test has been deleted in the 1992 Edition, 1992 Addenda of ASME Section XI."

Licensee's Proposed Alternative Examination (as stated):

"No alternative to this test is proposed since this test has been eliminated from the later Section XI Code."

Evaluation: For safety or relief valve piping that discharges into the containment pressure suppression pool, the Code requires a pneumatic test (at 90% of the pipe submergence head of water) that demonstrates the leakage integrity to be performed in lieu of a system hydrostatic test.

The licensee requested relief from performing the Code-required pneumatic test on the subject main steam discharge lines, stating that the requirement has been removed from the Code by the 1992 Addendum.

During a conference call held on May 11, 1995, the licensee reported that the subject lines were designed without test taps for pressurization of the lines. To perform the Code-required test, valves must be disassembled to allow a path for pressurization. Considerable burden is associated with the disassembly and re-assembly of a valve in each of the subject discharge lines.

The 90% of submergence head, the pressure required for this test, is 6.6 psig. This is much less than the 388 to 465 psig operating pressure of the discharge lines and does not significantly challenge the piping integrity.

Conclusion: Based on the above evaluation, requiring the licensee to perform the pneumatic test on the main steam discharge lines will impose undue hardship on the licensee without a compensating additional level of safety. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

3.4.4 General

3.4.4.1 Request for Relief 2ISI-13, IWA-4400(a), Pressure Test Requirements for Welded Repairs or Replacements of Class 1, 2, and 3 Items

Code Requirement: IWA-4400(a) requires that a system hydrostatic test be performed in accordance with IWA-5000 after repairs by welding on the pressure-retaining boundary.

Licensee's Code Relief Request: The licensee requested relief from the ASME Section XI elevated hydrostatic testing requirements for Class 1, 2, and 3 welded repair/replacement.

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

"Relief is requested from ASME Section XI hydrostatic test requirements on the basis of ASME Section XI Code Case N-416-1, 'Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items by Welding, Class 1, 2, and 3'. This Code Case allows alternative examination and testing requirements in lieu of performing the hydrostatic test on the welded joints.

"There will be no adverse impact on plant quality and safety by implementing Code Case N-416-1 requirements on ASME Section III Code Class 1, 2, and 3 welded joints due to the use of NDE examination and NDE examination acceptance criteria of ASME Section III, 1992 Edition and the alternate pressure test requirements at nominal operating pressure.

"The ASME Section III related piping systems and components at WNP-2 were constructed (material, designed, fabricated, installed, examined and tested) in accordance with ASME Section III requirements.

"The welded repairs or installation of replacement by welding will be performed in accordance with ASME Section XI, 1989 Edition and ASME Section III, (Code Edition and Addenda applicable to the component) requirements except that the NDE and pressure test requirements will be in accordance with Code Case N-416-1.

"The final welds for welded joints in ASME Section III, Code Class 1 (NB) and 2 (NC) piping systems and components will be NDE examined in accordance with ASME Section III, NX-5000, 1992 Edition requirements. The NDE examination acceptance criteria

for Code Class 1 and 2 will also be in accordance with ASME Section III, NX-5000, 1992 Edition requirements as required by Code Case N-416-1. The final welds for welded joints over 2" nominal pipe size (NPS) in ASME Section III, ND-5000, 1992 Edition. The NDE examination acceptance criteria will also be in accordance with ASME Section III, ND-5000, 1992 Edition requirements as required by Code Case N-416-1. The final welds for welded joints less than 2" nominal pipe size (NPS) in ASME Section III, Code Class 3 piping systems and components will be NDE examined by either magnetic particle (MT) or liquid penetrant (PT) examination in lieu of visual examination as required by ASME Section III, ND-5000, 1992 Edition.

"The VT-2 visual examination will be performed in conjunction with a system leakage test in accordance with ASME Section XI, IWA-5000, 1992 Edition requirements as required by Code Case N-416-1.

"The purpose of the hydrostatic test on the welded joints is to verify the leak tightness of the joint. The ability to identify a potential leak in the welded joint would be the same during a system leakage test at nominal operating pressure per Code Case N-416-1 as it would be at an elevated pressure test of 1.1 X design pressure through 1.25 X design pressure for the hydrostatic test of ASME Section XI."

<u>Licensee's Proposed Alternative Examination</u> (as stated):

"The non-destructive examination (NDE) on the welded joints in ASME Section III related piping systems and components will be performed in accordance with Code Case N-416-1 and ASME Section III requirements. Code Case N-416-1 requires that the NDE examination and acceptance criteria shall be in accordance with ASME Section III, 1992 Edition requirements.

"The VT-2 visual examination during pressure test on the welded joints in ASME Section III related piping systems and components will be performed at nominal operating pressure in accordance with Code Case N-416-1 and ASME Section XI, IWA-5000 requirements. Code Case N-416-1 requires that the VT-2 visual examination shall be performed in conjunction with a system leakage test in accordance with ASME Section XI, 1992 Edition requirements. The nominal operating pressure will be within a value that is obtained when the system is in service and the nominal operating temperature will be equal to the fluid state conditions for that pressure. In addition, welded joints less than 2" nominal pipe size (NPS) in ASME Section III, Code Class 3 piping systems and components will be NDE examined by either magnetic particle (MT) or liquid penetrant (PT) examination in lieu of visual examination as required by ASME Section III, ND-5000, 1992 Edition."

<u>Evaluation</u>: The Code requires a system hydrostatic pressure test for Class 1, Class 2, and Class 3 pressure-retaining components following a repair and/or replacement. Code Case N-416-1, *Alternative Pressure Test Requirements for Welded Repairs or Installation of Replacement Items by Welding*, requires a visual examination (VT-2) to be performed in conjunction with a system leakage test using the 1992 Edition of Section XI, in accordance with Paragraph IWA-5000, at nominal operating pressure and temperature. This Code Case also specifies that NDE of the welds be performed in accordance with the applicable subsection of the 1992 Edition of Section III.

Considering the Code requirements for NDE of Class 1 and Class 2 systems, the INEL staff believes that the increased assurance of structural integrity of Class 1 and Class 2 welds provided by the hydrostatic test is not commensurate with the burden. However, for Code Class 3 components there are no ongoing NDE requirements, except for visual examination for leaks in conjunction with the 10-year hydrostatic tests and the periodic pressure tests. Therefore, eliminating the hydrostatic test and only performing the system pressure test should only be considered acceptable if additional surface examinations are performed on the root pass layer of butt and socket welds on the pressure-retaining boundary of Class 3 components during repair or replacement activities.

<u>Conclusion</u>: Compliance with the Code hydrostatic testing requirements for welded repairs and replacements of Code Class 1, Class 2, and Class 3 components would result in hardship without a compensating increase in the level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative, to use Code Case N-416-1 be authorized pursuant to 10 CFR 50.55a(a)(3)(ii), provided that additional surface examinations are performed on the root pass layer of butt and socket welds on the pressure-retaining boundary during repair and replacement of Class 3 components. The surface examination method shall be in accordance with Section III. Use of Code Case

N-416-1, with the provision noted above, should be authorized until such time as the Code Case is published in a future revision of Regulatory Guide 1.147. At that time, the licensee should follow any provisions established for its use in Regulatory Guide 1.147.

3.4.4.2 Request for Relief 2ISI-07, IWA-5250(a)(2), Corrective Action for Leaking Bolted Connections

Code Requirement: IWA-5250(a)(2) requires bolting at leaking connections to be removed and VT-3 visually examined for corrosion.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required removal and VT-3 visual examination of bolting at leaking bolted connections in non-borated systems except:

1) Control Rod Drives and;
2) Bolted connections that were assembled as part of a Section XI repair and replacement activity and have not been subject to inservice conditions.

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

"Removing all the bolting from a leaking joint and performing a VT-3 visual examination on the bolting places a hardship on the plant. Bolting degradation due to bolted connection leaks would not occur at bolting that was not exposed to the leaking fluid. The Code requirement, if met, will require all the bolting at a leaking joint to be removed even if it is not in contact with the leaking fluid. This will result in unnecessary work for, and radiation exposure to, plant personnel.

"For bolted connections classified as hard joints, the retorquing of the bolting to no greater value than that allowed for the specific joint will provide a more positive test that the bolting is not degraded than removing the bolting and performing a visual examination on it. The retorquing will demonstrate that the bolting has not failed. In addition to being a demonstration of the bolting integrity, the retorquing should stop the leakage condition, which can cause degradation of the bolting.

"Where removal of the bolting is the corrective action, the removal of only one bolt is supported by subparagraph

IWA-5250(a)(2) of ASME Section XI, 1992 Edition, 1993 Addenda. By removing the bolt closest to the leak, the worst case condition will be examined. Bolting that is not in contact with the leaking fluid will not be unnecessarily removed. This will avoid unnecessary radiation exposure to plant personnel.

"WNP-2 is a Boiling Water Reactor (BWR) that does not utilize a process system with a borated water chemistry, except the standby liquid control system, like a Pressurized Water Reactor (PWR). The corrosiveness of leaking borated water in a PWR is known but is not applicable to WNP-2. The alternatives described provide assurance that corrosive conditions are evaluated.

"There will be no adverse impact on plant quality and safety by implementing the alternate corrective measures. The alternate corrective measures will demonstrate bolting integrity and provide reasonable assurance that bolting degradation caused by leakage at the bolted connection will be detected.

"Specific radiation exposure savings from this alternative corrective action cannot be calculated since they are bolted connection dependant."

Licensee's Proposed Alternative Examination (as stated):

"Relief is requested to use the corrective measures described in the following paragraphs.

"For bolted connections that are shown to have metal-to-metal contact ("hard joints"), torque the bolted connection up to the maximum torque allowed for this joint while this joint is under pressure. If the leak stops no further action will take place. If the leak does not stop, the leakage will be evaluated for further corrective action.

"For bolted connections other than hard joints, remove one bolt closest to the leak, perform a VT-3 examination for corrosion, and evaluate in accordance with IWA-3100."

Evaluation: For bolted connections classified as hard joints, the licensee proposes to re-torque the bolting up to the maximum torque allowed for that specific joint. This action will provide a more positive test that the bolting is not degraded than removing the bolting and performing a visual examination. The re-torquing will demonstrate that the bolting has not failed. In addition to being a demonstration of the bolting integrity, the re-torquing should stop the leakage condition, which can cause degradation of the bolting.

For bolted joints other than hard joints, the bolt closest to the leak will be removed. This is supported by subparagraph IWA-5250(a)(2) of ASME Section XI, 1989 Edition, 1990 Addenda. By removing the bolt closest to the leak, the worst case condition will be examined. Bolting that is not in contact with the leaking fluid will not be unnecessarily removed. This will avoid unwarranted radiation exposure to plant personnel.

<u>Conclusion</u>: For bolted metal-to-metal joints, the licensee's proposed alternative is to retorque the joint while it is under pressure to the maximum allowed in order to stop the leak. If the leak is not stopped, the licensee will evaluate the leakage for further action. For bolted connections other than metal-to-metal joints, the licensee will remove the bolt closest to the leak and perform VT-3 visual examination for corrosion per IWA-3100. The licensee's proposed alternative provides an acceptable level of quality and safety. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

3.4.4.3 <u>Request for Relief 2ISI-14, IWA-5250(a)(2) Corrective Action for Leaking Bolted Connections as part of Repair and Replacement Activities</u>

<u>Code Requirement</u>: IWA-5250(a)(2) requires bolting at leaking connections to be removed and VT-3 visually examined for corrosion.

<u>Licensee's Code Relief Request</u>: The licensee requested relief from performing the Code-required removal and VT-3 visual examination of bolting at connections as part of repairs or replacements.

<u>Licensee's Basis for Requesting Relief</u> (as stated):

"Relief is requested from ASME Section XI, IWA-5250(a)(2), removal and VT-3 visual examination of bolting material for bolted mechanical joints when leakage is observed during VT-2 visual examination during the system pressure test following ASME

Section XI repair and replacement activities. The alternatives addressed below will provide an equivalent determination of whether the bolting material is degraded. These alternatives provide acceptable detection of bolting corrosion in a Boiling Water Reactor (BWR) service with process systems not using borated water.

"ASME Section XI, IWA-5250(a)(2), 1989 Edition would literally require complete removal and inspection of all the bolting of a bolted mechanical joint for evaluation if leakage occurs at the bolted mechanical joint during the system pressure test following ASME Section XI repair and replacement activities, even though the leakage could be very small and considered acceptable or known to stabilize or stop after a period of time. This could require an unnecessary cooldown or shutdown of the plant, including extra unnecessary radiation exposure of personnel, and a delay in plant startup following an outage. The alternatives presented provide assurance that the bolted mechanical joints are adequately evaluated for a Boiling Water Reactor (BWR) plant to provide an acceptable level of quality and safety.

"The purpose of ASME Section XI, IWA-5250(a)(2) is determine if inservice leakage has degraded the bolting material. By installing new replacement bolting material or performing a VT-3 examination on the existing bolting material prior to reinstalling, the integrity of the bolting material is assured prior to performing the pressure test for repair and replacement activities. ASME Interpretation No 30, XI-1-92-01 supports exclusion of ASME Section XI, IWA-5250(a)(2) for new or VT-3 examined bolting material.

"WNP-2 is a Boiling Water Reactor (BWR) that does not utilize a process system with a borated water chemistry, (except the Standby Liquid Control system) like a Pressurized Water Reactor (PWR). The corrosiveness of leaking borated water in a PWR is not applicable to WNP-2. The alternatives described provide assurance that corrosive conditions are evaluated.

"Bolted material that has been replaced (new) would not warrant a VT-3 visual examination of the bolting material if leakage was observed. The purpose of the VT-3 visual examination is to examine for localized general corrosion. The localized general corrosion is not expected to be observed on the new replacement bolting material since the bolting material has not been through the service conditions associated with the piping system. If leakage is observed during VT-2 visual examination during the system pressure test, the leakage from the bolted mechanical joint would be evaluated for acceptability to determine the corrosive effects on the bolting material or the corrective action.

"VT-3 visual examination performed on the existing bolting material for mechanical joints prior to reassembly establishes a baseline that bolting corrosion is not present prior to installation. Therefore leakage observed during the VT-2 visual

examination during the system pressure test, for the joint would be evaluated for acceptability to determine the corrosion effects on the bolting material or for corrective action. Bolting not removed and not part of the repair or replacement activity are included in the ASME Section XI system pressure test boundary (relief request 2ISI-07).

"The VT-1 visual examination performed on the reused existing and/or replacement (new) bolting material for the Control Rod Drives (CRD) bolted mechanical joints prior to reassembly establishes a baseline that the bolting corrosion is not present prior to installation. If leakage is observed during VT-2 visual examination during the system pressure test, the leakage from the bolted mechanical joint would be evaluated for acceptability to determine the corrosive effects on the bolting material or the corrective action."

Licensee's Proposed Alternative Examination (as stated):

"The following alternatives will be implemented to comply with the intent of ASME Section XI, IWA-5250(a)(2) requirements.

"Bolting material for mechanical joints for Control Rod Drives (CRD) will have a VT-1 visual examination performed on all the reused *existing and/or new replacement* bolting material prior to installation. If leakage is observed from the bolted mechanical joint during the VT-2 visual examination during system pressure test it will be evaluated for acceptability or corrective action."

Evaluation: If leakage occurs at a joint during the system pressure test following ASME Section XI repair and replacement activities, the Code requires removal and inspection of all bolting as part of the evaluation. This is required even if the leakage is very small and considered acceptable, or if it is known to stabilize or stop after a period of time. Removal of bolting could require a cooldown or shutdown of the plant, including extra unnecessary radiation exposure of personnel, and a delay in plant startup following an outage.

The purpose of IWA-5250(a)(2) is to determine if inservice leakage has degraded the bolting. By installing new bolting prior to performing the pressure test for repair and replacement activities, or performing a VT-3 examination on the existing bolting prior to reinstalling, the integrity of the bolting is assured.

WNP-2 is a boiling water reactor; its only process system with a borated water chemistry is the standby liquid control system. Thus, the corrosiveness of leaking borated water that is normally not of concern and thus is not applicable to WNP-2. The alternatives described provide assurance that corrosive conditions are evaluated.

VT-3 visual examination performed on the existing bolting for mechanical joints prior to reassembly establishes a baseline that bolting corrosion is not present prior to installation. Therefore, leakage observed during the VT-2 visual examination during the system pressure test would be evaluated for acceptability to determine the corrosive effects on the bolting and the need for corrective action.

The VT-1 visual examination performed on the reused existing and/or replacement (new) bolting material for the Control Rod Drive's bolted mechanical joints prior to reassembly establishes that corrosion is not present prior to installation. If leakage is observed in the VT-2 visual examination during the system pressure test, the leakage from the bolted mechanical joint would be evaluated for acceptability to determine the corrosive effects on the bolting and the need for corrective action.

<u>Conclusion</u>: The licensee's proposed alternative, to evaluate the integrity of bolted connections disassembled and reassembled during the course of Section XI repair and replacement activities, provides an acceptable level of quality and safety, provided the licensee evaluates the leakage in accordance with IWB-3142.4 for joint integrity. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i).

3.5 General

3.5.1 Ultrasonic Examination Techniques (No requests for relief)

3.5.2 Exempted Components (No requests for relief)

3.5.3 Other

3.5.3.1 Request for Relief 2ISI-12, IWA-4340, Examination of Affected Surfaces After Final Grinding During Weld Repair

Code Requirement: IWA-4340(a) requires that after the final grinding, the affected surfaces, including surfaces of the cavities prepared for welding, shall be examined by magnetic particle (MT) or liquid penetrant (PT) methods to ensure that the indication has been reduced to an acceptable level.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required MT or PT examination when defect elimination removes the full thickness of the weld and the back side of the weld joint is not accessible for application or removal of the MT or PT examination material.

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

"The basis for the relief is that when the full thickness of the joint is removed the defect is removed and the back side is not accessible, the NDE material cannot be properly applied or removed and a meaningful examination cannot be performed. In addition, ASME requirements have been clarified in the 1983 Edition of ASME Section III and 1992 Edition of ASME Section XI.

"There will be no adverse impact on plant quality and safety by implementing the above discussed alternative since ASME Section III requirements for defect removal were previously implemented in conjunction with the requirements of ASME Section III, Code Case N-275 during construction of the piping systems.

"ASME Section III, Code Case N-275 was previously approved for use in Regulatory Guide 1.84. This Code Case was annulled on December 31, 1983 and was incorporated in ASME Section III, NX-4453.1, 1983 Edition. The surface examination using magnetic

particle (MT) or liquid (PT) methods of a through wall excavation, without ID access, results in NDE material such as magnetic particle powder or liquid penetrant dye or developer becoming entrained in a plant piping system and creates a difficulty in obtaining a meaningful NDE examination. The ASME Section III related piping systems and components at WNP-2 were constructed (material, designed, fabricated, installed, examined and tested) in accordance with ASME Section III requirements. The ASME Section III requirements which were implemented during defect removal for ASME Section XI repairs in conjunction with the requirements of ASME Section III, Code Case N-275, 'Repair Of Welds'.

"Gouging through the wall in order to qualify for use of the ASME Section III, Code Case N-275 will not be used.

"ASME Section III and ASME Section XI, IWA-4331(a), Page 34 of 1992 Edition with 1993 Addenda clarifies that magnetic particle (MT) or liquid penetrant (PT) examination is not required when defect elimination removes the full thickness of the weld where the back side of the weld joint is not accessible for removal of the examination material."

Licensee's Proposed Alternative Examination (as stated):

"The unacceptable indications will be removed by mechanical means or by the thermal gouging method as specified by ASME Section III, Code Case N-275, 'Repair Of Welds'. The area prepared for repair will be examined by magnetic particle (MT) or liquid penetrant (PT) method in accordance with ASME Section III, NX-5300 as specified by ASME Section III Code Case N-275. The NDE examination is not required where the full thickness of the weld is removed and where the back side of the weld joint assembly is not accessible for removal of the NDE examination material as specified by ASME Section III Code Case N-275. The completed weld repair will be NDE examined in accordance with ASME Section III, NX-4453.4 as specified by ASME Section III, Code Case N-275. Gouging through the wall in order to qualify for use of the ASME Section III, Code Case N-275 will not be used."

Evaluation: The Code requires that after the final grinding to remove a defect, the affected surfaces, including surfaces of the cavities prepared for welding, shall be examined to ensure that the indication has been reduced to an acceptable level.

The licensee requested relief from performing the Code-required surface examination when defect elimination removes the full thickness of the weld and the back side of the weld joint is not

accessible for removal of the examination material. ASME Section XI, IWA-4331(a), 1992 Edition, clarifies that magnetic particle (MT) or liquid penetrant (PT) examination is not required when this condition exists.

During construction of the piping systems, the requirements for defect removal were contained in ASME Section III, Code Case N-275. This Code case did not require a surface examination under these circumstances. Where the examination material cannot be removed, this Code requirement is impractical.

Conclusion: Based on the above evaluation, the Code-required surface examination is impractical to perform. Therefore, it is recommended that relief be granted, pursuant to 10 CFR 50.55a(g)(6)(i).

3.5.3.2 Request for Relief 2ISI-11, IWF-5300(a) Snubber Examination

Note: This request for relief is considered part of the Inservice Testing Program (IST) and is, therefore, not included in this evaluation. The Snubber Testing Program will be evaluated by the Mechanical Engineering Branch of NRC.

4. CONCLUSION

Pursuant to 10 CFR 50.55a(g)(6)(i), it has been determined that certain inservice examinations cannot be performed to the extent required by Section XI of the ASME Code. In the cases of Requests for Relief 2ISI-01 (Part 2), 2ISI-02, 2ISI-04, 2ISI-10, and 2ISI-12, the licensee has demonstrated that specific Section XI requirements are impractical; it is therefore recommended that relief be granted as requested. The granting of relief will not endanger life, 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.

Pursuant to 10 CFR 50.55a(a)(3)(i), it is concluded that for Requests for Relief 2ISI-03, 2ISI-07, and 2ISI-14, the licensee's proposed alternatives will provide an acceptable level of quality and safety in lieu of the Code-required examinations. In these cases, it is recommended that the proposed alternative be authorized.

Pursuant to 10 CFR 50.55a(a)(3)(ii), it is concluded that for Requests for Relief 2ISI-05, 2ISI-06, 2ISI-08, and 2ISI-09, the licensee has demonstrated that specific Section XI requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. In this case, it is recommended that the proposed alternative be authorized. In the case of Request for Relief 2ISI-13 it is recommended that the proposed alternative be authorized only if the licensee satisfies the conditions stated in the request for relief evaluation above.

For Request for Relief 2ISI-01 (Part 1), it is concluded that the licensee has not provided sufficient justification to support the determination that the Code requirement is impractical, and that requiring the licensee to comply with the Code requirement would not result in hardship. Therefore, in this case it is recommended that relief be denied.

Request for Relief 2ISI-11 was not evaluated as part of this report.

This technical evaluation has not identified any practical method by which the licensee can meet all the specific inservice inspection requirements of Section XI of the ASME Code for the existing WPPSS Nuclear Project, Unit 2, facility. Compliance with all of the Section XI examination requirements would necessitate redesign of a significant number of plant systems, procurement of replacement components, installation of the new components, and performance of baseline examinations for these components. Even after the redesign efforts, complete compliance with the Section XI examination requirements probably could not be achieved. Therefore, it is concluded that the public interest is not served by imposing certain provisions of Section XI of the ASME Code that have been determined to be impractical.

The licensee should continue to monitor the development of new or improved examination techniques. As improvements in these areas are achieved, the licensee should incorporate these techniques into the ISI program plan.

Based on the review of the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the licensee's response to the NRC's request for additional information, and the recommendations for granting relief from the ISI examinations that cannot be performed to the extent required by Section XI of the ASME Code, no deviations from regulatory requirements or commitments were identified, except those noted in the evaluation of Request for Relief 2ISI-01 (Part 1).

5. REFERENCES

1. Code of Federal Regulations, Title 10, Part 50.

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components*, Division 1:

   1989 Edition

3. *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, dated December 27, 1994.

4. NUREG-0800, *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants*, Section 5.2.4, "Reactor Coolant Boundary Inservice Inspection and Testing," and Section 6.6, "Inservice Inspection of Class 2 and 3 Components," July 1981.

5. Letter, dated May 12, 1995, WPPSS (Washington Public Power Supply System) to Document Control Desk (NRC), containing the response to NRC request for additional information.

6. NUREG-0619, *BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking*, November 1980.

7. NRC Generic Letter 88-01, *NRC Position on Intergranular Stress Corrosion Cracking in BWR Austenitic Stainless Steel Piping*, January 25, 1988.

8. NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, Revision 1, February 1983.

9. IE Bulletin 80-13, *Cracking in Core Spray Spargers*, April 4, 1980.

10. NRC Generic Letter 94-03, *Intergranular Stress Corrosion Cracking of Core Shrouds in Boiling Water Reactors.*

11. Supply System Letter G02-90-024, *Request for Amendment to Technical Specifications, Final Feedwater Temperature Reduction (FFTR)*, February 14, 1994.

12. WNP-2 Final Safety Analysis Report (FSAR) Section 3.6.2.1.2.1(a), *Augmented High Energy Piping Examination.*

NRC FORM 335 (2-89) NRCM 1102, 3201, 3202

U.S. NUCLEAR REGULATORY COMMISSION

BIBLIOGRAPHIC DATA SHEET

(See instructions on the reverse)

1. REPORT NUMBER *(Assigned by NRC. Add Vol., Supp., Rev., and Addendum Numbers, if any.)*

INEL-95/0400

2. TITLE AND SUBTITLE

Technical Evaluation Report on the Second 10-Year Interval Inservice Inspection Program Plan: Washington Public Power Supply System, WPPSS Nuclear Project, Unit 2 Docket Number 50-397

3. DATE REPORT PUBLISHED

MONTH	YEAR
August	1995

4. FIN OR GRANT NUMBER

JCN-J2229 (TWA-A07)

5. AUTHOR(S)

M. T. Anderson, E. J. Feige, K. W. Hall

6. TYPE OF REPORT

Technical

7. PERIOD COVERED *(Inclusive Dates)*

8. PERFORMING ORGANIZATION — NAME AND ADDRESS *(If NRC, provide Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)*

LITCO
P.O. Box 1625
Idaho Falls, ID 83415-2209

9. SPONSORING ORGANIZATION — NAME AND ADDRESS *(If NRC, type "Same as above"; if contractor, provide NRC Division, Office or Region, U.S. Nuclear Regulatory Commission, and mailing address.)*

Civil Engineering and Geosciences Branch
Office of Nuclear Regulatory Commission
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

10. SUPPLEMENTARY NOTES

11. ABSTRACT *(200 words or less)*

This report presents the results of the evaluation of the *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval Inservice Inspection (ISI) Program Plan*, submitted December 27, 1994, including the requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Section XI requirements. that the licensee has determined to be impractical. The *WPPSS Nuclear Project, Unit 2, Second 10-Year Interval ISI Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during previous Nuclear Regulatory Commission (NRC) reviews. The requests for relief are evaluated in Section 3 of this report.

12. KEY WORDS/DESCRIPTORS *(List words or phrases that will assist researchers in locating the report.)*

13. AVAILABILITY STATEMENT

Unlimited

14. SECURITY CLASSIFICATION

(This Page) Unclassified

(This Report) Unclassified

15. NUMBER OF PAGES

16. PRICE