UNITED STATES
NUCLEAR REGULATORY COMMISSION
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

OF THE SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN

AND ASSOCIATED REQUESTS FOR RELIEF

CALLAWAY PLANT, UNIT NO. 1

DOCKET NO. 50-483

1.0 INTRODUCTION

The Technical Specifications for Callaway Plant, Unit No. 1, state that the inservice inspection of the American Society of Mechanical Engineers (ASME) Code Class 1, 2, and 3 components shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). Section 50.55a(a)(3) states that alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Case 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 for 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 edition of Section XI of the ASME Code for the Callaway Plant, Unit No. 1 second 10-year inservice inspection (ISI) interval is the 1989 Edition. The components (including supports) may meet the requirements set forth in subsequent editions and addenda of the ASME Code incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein and subject to Commission approval.

Pursuant to 10 CFR 50.55a(g)(5), if the licensee determines that conformance with an examination requirement of Section XI of the ASME Code is not practical for its facility, information shall be submitted to the Commission in support of that determination and request made for relief from the ASME Code requirement. After evaluation of the determination, pursuant to 10 CFR 50.55a(g)(6)(i), the Commission 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. By letters dated October 12, 1994, and August 18, 1995, Union Electric Company submitted Revisions 0 and 1, respectively, of its second 10-year interval inservice inspection program plan and associated requests for relief for Callaway Plant, Unit No. 1.

2.0 EVALUATION

The staff, with technical assistance from its contractor, the Idaho National Engineering Laboratory (INEL), has evaluated the information provided by the licensee in support of its second 10-year interval inservice inspection program plan and associated requests for relief for Callaway Nuclear Power Plant, Unit 1. This review is based on the information contained in Revision 1 of the Callaway Plant, Unit No. 1 Second 10-Year Interval Inservice Inspection Program Plan.

Based on the information provided by the licensee, the staff adopts the contractor's conclusions and recommendations presented in the attached Technical Evaluation Report. The staff has concluded that no deviations from regulatory requirements or commitments were identified in Revision 1 of the Callaway Plant, Unit No. 1 Second 10-Year Interval Inservice Inspection Program Plan.

In addition, the staff has determined that for request for relief ISI-03, the examinations required by the Code are impractical and that the licensee's proposed alternatives to Code requirements provide reasonable assurance of operational readiness. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), relief is granted as requested for request for relief ISI-03.

The staff has also concluded that the alternatives contained in requests for relief ISI-04, ISI-06, ISI-07, ISI-08, and ISI-09 provide an acceptable level of quality and are authorized pursuant to 10 CFR 50.55a(a)(3)(i) with the conditions listed below:

(1) The alternative contained in request for relief ISI-04 is authorized provided that the volumetric examinations of the adjacent circumferential welds provide scanning for reflectors transverse to the weld.

(2) The alternative contained in request for relief ISI-06 is authorized provided that a minimum 10 percent sample of all nonexempt Code Class 1, 2, and 3 integral attachments is examined.

(3) The alternative contained in request for relief ISI-07 is authorized provided that the licensee's alternative is supplemented by the following:

(a) At each refueling outage the licensee shall remove all existing removable insulation at bolted connections in systems borated for the purpose of controlling reactivity and perform a VT-2 visual examination for evidence of leakage.

(b) Where nonremovable insulation exists at bolted connections, the licensee may visually examine the joint without removing the insulation provided that a 4-hour hold time is satisfied prior to the VT-2 visual examination.

(4) The alternative contained in request for relief ISI-08 is authorized provided that if the bolting is susceptible to corrosion or the initial evaluation indicates the need for a more in-depth evaluation, the bolt closest to the source of leakage will be removed, VT-3 examined, and evaluated in accordance with IWA-3100(a).

(5) The alternative contained in request for relief ISI-09 is authorized provided that the tests are conducted at peak calculated containment pressure and that the test procedures include methods for detecting and locating through-wall leakage in containment isolation valves (CIVs) and pipe segments between the CIVs.

Furthermore, the staff has concluded that for request for relief ISI-05, compliance with the Code requirements would result in a hardship without a compensating increase in safety and that the proposed testing provides reasonable assurance of operational readiness. Therefore, the alternative contained in request for relief ISI-05 is authorized pursuant to 10 CFR 50.55a(a)(3)(ii) provided that all of the examinations are completed within the same period in which the examinations were performed in the preceding interval, or earlier, so that examinations are no more than 10 years apart, except where the length of a 10-year interval is adjusted in accordance with IWA-2430.

With regard to request for relief ISI-02, the staff has also concluded that the licensee's commitment to perform the augmented examinations of NRC Standard Review Plan, Section 3.6.1, volumetric examinations on circumferential welds in Class 2 high-energy fluid system piping located outside the containment and exceeding 1-inch nominal pipe size, is impractical and that the licensee's alternative will insure the structural integrity of the piping. Therefore, the licensee's proposed alternative to the NRC Standard Review Plan, Section 3.6.1 examinations is acceptable.

Request for relief ISI-01 is denied because the licensee did not demonstrate the impracticality or the hardship of implementing the Code requirement of Section XI, paragraph IWA-2311(b), that the training, qualification and certification of ultrasonic examination personnel comply with the requirements specified in Appendix VII.

3.0 CONCLUSION

The staff has determined that, with respect to requests for relief ISI-04, ISI-06, ISI-07, ISI-08, and ISI-09, the proposed alternatives, with the conditions listed in the staff's safety evaluation, are acceptable pursuant to 10 CFR 50.55a(a)(3)(i). The staff has determined that with respect to request for relief ISI-05, compliance by Union Electric Company would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and therefore the proposed alternative is authorized pursuant to 10 CFR 50.55a(a)(3)(ii). With respect to request for relief ISI-03, the staff has determined that the testing requirements for the subject component are impractical and pursuant to 10 CFR 50.55a(g)(6)(i), the granting of relief is authorized by law, will not endanger life, property or the common defense and security, and is otherwise in the public interest. With respect to request for relief ISI-02, the staff has determined that the augmented examinations of NRC Standard Review Plan, Section 3.6.1, volumetric examinations on circumferential welds in Class 2 high-energy fluid system piping located outside the containment and exceeding 1-inch nominal pipe size are impractical, and that the licensee's alternative will ensure the structural integrity of the piping. Request for relief ISI-01 is denied because Union Electric Company did not demonstrate the impracticality or the hardship of implementing the Code requirement of Section XI, paragraph IWA-2311(b).

Attachments: 1. Summary of Relief Requests
2. Technical Evaluation Report

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CALLAWAY NUCLEAR POWER PLANT, UNIT 1
Second 10-Year ISI Interval

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	System or Component	Exam Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Status
ISI-01		IWA-2311(b)			Volumetric examination in accordance with Appendix VII	Deferral of implementation until Appendix VIII is required	Denied
ISI-02	Reactor Coolant Pump Seal Water	Standard Review Plan	3.6.1	Circumferential Welds	Volumetric examination	Surface examination	Acceptable
ISI-03	Reactor Vessel	F-A	F1.40	Supports	VT-3 visual examination	VT-3 Visual examination to extent practical	Granted
ISI-04	Class 1 an 2 Piping	B-J C-F-1 C-F-2		Longitudinal Welds	Surface and/or Volumetric examination	Code Case N-524	Authorized Conditionally
ISI-05	Reactor Vessel	B-D B-F	B3.90 B3.100 B5.10	Nozzle-to-Vessel Welds Inside Radius Sections Nozzle-to-Safe End Welds	Volumetric examination of 25% during first period	Deferral of examination to third period	Authorized Conditionally
ISI-06	Integral Attachments	B-H, B-K, C-C, D-A, D-C		Alternative rules of selection of integrally welded attachments	Volumetric or surface examination	Code Case N-509	Authorized Conditionally
ISI-07	Class 1 and 2 Borated	IWA-5242(a)		Bolted connections	Insulation removed for direct VT-2 visual examination	Perform evaluation	Authorized Conditionally

TABLE 1
SUMMARY OF RELIEF REQUESTS

Relief Request Number	System or Component	Exam Category	Item No.	Volume or Area to be Examined	Required Method	Licensee Proposed Alternative	Relief Request Status
ISI-08	Class 1, 2, and 3	IWA-5250(a)(2)		Leaking bolted connections	VT-3 visual examination of all bolting at connection	Perform evaluation	Authorized Conditionally
ISI-09	Class 2	C-H	C7.30 C7.40 C7.70 C7.80	Components at Containment Penetrations	System leakage test	10 CFR 50, Appendix J testing	Authorized Conditionally

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Technical Evaluation Report on the Second 10-year Interval Inservice Inspection Program Plan: Union Electric Company, Callaway Nuclear Power Plant, Unit 1, Docket Number 50-483

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ABSTRACT

This report presents the results of the evaluation of the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted August 18, 1995, including the requests for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, requirements that the licensee has determined to be impractical. The *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1 is evaluated in Section 2 of this report. The Inservice Inspection (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 reviews. The requests for relief are evaluated in Section 3 of this report.

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SUMMARY

The licensee, Union Electric Company, has prepared the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, 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 August 1, 1995 and ends December 18, 2004.

The information in the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted October 10, 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 request for additional information (RAI) was prepared describing the information and/or clarification required from the licensee in order to complete the review. The licensee provided the requested information in the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted August 18, 1995.

A conference call was held September 13, 1995, to clarify that the augmented requirement of 10 CFR 50.55a(g)(6)(ii)(A) was satisfied during the previous interval.

Based on the review of the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, 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 *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, except as noted in the evaluation of Request for Relief ISI-01.

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TECHNICAL EVALUATION REPORT ON THE
SECOND 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN:
UNION ELECTRIC COMPANY,
CALLAWAY NUCLEAR POWER PLANT, UNIT 1,
DOCKET NUMBER 50-483

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 the 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, Union Electric Company, prepared the *Callaway Nuclear Power Plant, Unit 1, 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 August 1, 1995, and ends December 18, 2004.

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 *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted October 10, 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."

In a letter dated May 23, 1995 (Reference 5), the NRC requested additional information that was required to complete the review of the ISI Program Plan. The requested information was provided by the licensee, Union Electric Company, in the *Callaway Nuclear Power Plant Unit 1, Second Interval Inservice Inspection Program Plan*, Revision 1, dated August 18, 1995 (Reference 6).

A conference call was held September 13, 1995, to clarify that the augmented requirement of 10 CFR 50.55a(g)(6)(ii)(A) was satisfied during the previous interval.

The *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, 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 programs for pumps and valves are being evaluated in other reports.

2. EVALUATION OF INSERVICE INSPECTION PROGRAM PLAN

This evaluation consisted of a review of the applicable program documents to determine whether 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) *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, (Reference 3); and

(b) *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, (Reference 6).

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 August 18, 1995, the Code applicable to the second 10-year interval ISI program is the 1989 Edition. As stated in Section 1 of this report, the licensee has prepared the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan* to meet the requirements of 1989 Edition of the Code.

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

(a) Reactor vessel examinations in accordance with the requirements of NRC Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examinations*, Revision 1, (Reference 7);

(b) Volumetric examination of the reactor coolant pump flywheel high stress areas every 3 years, as well as volumetric and surface examinations with the flywheel removed at 10-year intervals, satisfying NRC Regulatory Guide 1.14, *Reactor Coolant Pump Flywheel Integrity*, (Reference 8);

(c) Examination of the portions of high energy lines specified in Standard Review Plan 3.6.1;

(d) Examinations of the portions of sensitized stainless steel specified in Section B of Technical Specification Table 4.2-1;

(e) Ultrasonic examination of Steam Generator Feedwater Nozzles per IE Bulletin 79-13, *Cracking in Feedwater System Piping* (Reference 9);

(f) Volumetric and surface examination of all low-pressure turbine blades and a volumetric examination of the low-pressure turbine disc bore and keyway every five years as specified in Technical Specification 4.2-1; and

(g) Eddy current examination (100%), each refueling outage, of all reactor vessel in-core detector thimble tubes that are in service per IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors* (Reference 10).

2.3 Conclusions

Based on the review of the documents listed above, no deviations from regulatory requirements or commitments were identified in the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, except as noted in the evaluation of Request for Relief ISI-01.

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 ISI-05, Examination Categories B-D and B-F, Items B3.90, B3.100, and B5.10, Deferral of Inspections of Reactor Vessel Nozzle-to-Vessel Welds, Inside Radius Sections and Nozzle-to-Safe End Welds

Code Requirement: Examination Category B-D, Items B3.90 and B3.100, Note 2 requires that at least 25% but not more than 50% (credited) of the nozzles shall be examined by the end of the first inspection period and the remainder by the end of the inspection interval.

Examination Category B-F, Item B5.10, reactor vessel nozzle safe ends, states that the Code-required examinations may be performed coincident with the vessel nozzle examinations required by Examination Category B-D.

Paragraph IWB-2420(a) requires that the sequence of component examinations established during the first inspection interval be repeated during each successive inspection.

Licensee's Code Relief Request: The licensee requested relief from the scheduling requirements of IWB-2420(a) for the reactor pressure vessel nozzle-to-vessel welds, inside radius sections, and nozzle-to-safe end welds.

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

"Relief is requested to defer 100 percent of the reactor vessel nozzle-to-vessel weld examinations, the nozzle inside radius section examinations, and the nozzle-to-safe end weld examination to the end of Callaway's second ten-year inspection interval.

"Union Electric believes that performing 25 percent to 50 percent of the reactor vessel nozzle examination in the first period of the second inspection interval is impractical for the following reasons.

"1) The vendor cost alone (not including site training, plant support, or potential critical path time) to perform these examinations with automated tooling in the first inspection period is currently estimated at $250,000. The cost to perform these same examinations at the end of the second inspection interval concurrent with the reactor vessel ten-year examination is estimated at only $25,000. The major expense associated with the first inspection period examinations is the added equipment and personnel mobilization cost and equipment assembly and disassembly costs.

"2) Approximately three to four man-rem exposure is currently expended for automated equipment assembly and disassembly in the reactor cavity area. In addition to exposure, there are approximately two to three cubic feet of solid radwaste generated during performance of automated examinations in the reactor vessel. Under current Code rules, this personnel exposure and radwaste generation would be incurred twice, once for the nozzle first inspection period examinations and again for the reactor vessel examinations at the end of the inspection interval. Performing the nozzle examinations concurrent with the reactor vessel ten-year examinations will save approximately three to four man-rem exposure and two to three cubic feet of solid radwaste.

"For reasons listed below, Union Electric believes that deferral of 100% of the reactor vessel nozzle examinations to the end of the second inspection interval will provide an acceptable level of safety and quality.

"1) All four of Callaway's Reactor Vessel hot leg nozzle-to-vessel welds, and hot leg nozzle-to-safe end welds were examined in 1987 during the first period of the first ten-year inspection interval. No indications or relevant conditions were discovered that required successive inspections in accordance with Paragraph IWB-2420(b). Furthermore, no inservice repairs or replacements by welding have ever been performed on any of the nozzle-to-vessel welds, nozzle inside radius sections, or nozzle-to-safe end welds at Callaway.

"2) From an industry perspective, there are two reasons why deferral of Callaway's nozzle examinations to the end of the second inspection interval will not decrease the level of quality and safety. First, PWR reactor vessels similar to Callaway's have been operating for over 20 years with no recorded inservice induced flaws or potential degradation mechanisms. Since each PWR reactor vessel in operation is representative of the operating conditions throughout the industry, continued inspection of these vessels ensures that any potential degradation mechanism will be detected. Second, given the present large population of PWR reactor vessels in operation, the examination of nozzles within the industry during any ten-year interval is evenly distributed. This distribution is essentially equivalent, regardless of whether or not a percentage of the nozzle examinations are performed in the first inspection period or performed concurrent with the reactor vessel ten-year examinations at the end of the inspection interval.

"3) The pressurizer and primary steam generator nozzle-to-vessel welds, inside radius sections, and nozzle-to-safe end welds are similar in configuration, material properties, weld process parameters, and operate in the same reactor coolant system environment as the reactor vessel nozzles. Due to this similarity, distribution of the pressurizer and steam generator nozzle examinations in accordance with Examination Category B-D and Examination Category B-F will further substantiate the integrity of the reactor vessel nozzles until they are examined at or near the end of the second inservice inspection interval.

"4) Performing all the automated reactor vessel examinations during a single refueling outage improves consistency of the examinations by utilizing the same equipment, personnel, and procedures. Moreover, this improves the reliability and reproducibility of the examinations."

In response to a request for additional information the licensee submitted the following:

"Callaway Relief Request ISI-05 requests deferral to the end of the second ten-year interval examination of the Reactor Vessel nozzle-to-vessel welds, nozzle inside radius sections, and nozzle-to-safe end welds. These examination will not be deferred over a time period exceeding the ten years, plus one year, allowed by ASME Section XI. One hundred percent of these nozzle examinations were performed during the third period of the first ten-year interval and are scheduled again for the third period of the second ten-year interval."

Licensee's Proposed Alternative Examination (as stated):

"Union Electric shall complete the required nozzle-to-vessel weld examinations, the nozzle inside radius section examinations, and the nozzle-to-safe end weld examinations concurrent with the reactor vessel ten-year examinations at or near the end of the second ten-year inservice inspection interval. In addition, the reactor vessel hot leg nozzle inside surfaces, including the inside radius sections and nozzle-to-safe end weld areas, that are made accessible with the upper internals removed and lower internals (core barrel) installed, shall be visually VT-3 examined once each inspection period of the second ten-year inservice inspection interval."

Evaluation: The licensee stated that the scheduling requirements of Examination Categories B-A and B-D, Items B1.30, B3.90 and B3.100 result in a hardship. The INEL staff believes that, for these items, deferral of the first period examinations is acceptable provided that they are completed within the same period in which the examinations were performed in the preceding interval, or earlier, so that there is no more than 10 years between examinations.

The licensee has established an acceptable level of quality and safety for the subject welds by examinations performed during the third period of the previous 10-year interval. The imposition of examinations of the subject welds in the first period of this interval is regarded as a burden. The proposed alternative, performance of all Item B1.30, B3.90, and B3.100, examinations in the third period of this interval, will provide reasonable assurance of operational readiness if all of the examinations are completed within the same period in which the preceding interval examinations were performed, or earlier, so that there is no more than 10 years between examinations, except where the length of a 10-year interval is adjusted in accordance with IWA-2430.

Conclusion: The INEL staff has reviewed the licensee's request for relief from the scheduling requirements for the subject examinations. It is concluded that, for the subject welds, performing the required examinations in the first period of the second interval would result in an unnecessary burden without a compensating increase in the level of quality and safety. Furthermore, the proposed alternative, performance of all Item B1.30, B3.90, and B3.100, examinations in the third period of the second 10-year interval, will provide reasonable assurance of operational readiness. Therefore, it is recommended that the alternative scheduling be authorized pursuant to 10 CFR 50.55a(a)(3)(ii) provided that all of the examinations are completed within the same period in which the examinations were performed in the preceding interval, or earlier, so that there are not more than 10 years between examinations, except where the length of a 10-year interval is adjusted in accordance with IWA-2430.

3.1.2 Pressurizer (No requests for relief)

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

3.1.4 Piping Pressure Boundary (No requests for relief)

3.1.5 Pump Pressure Boundary (No requests for relief)

3.1.6 Valve Pressure Boundary (No requests for relief)

3.1.7 General

3.1.7.1 Request for Relief ISI-03, Examination Category F-A, Item F1.40, Examination of Reactor Vessel Supports

Code Requirement: Code Case N-491, Table-2500-1, Examination Category F-A, Item F1.40 requires 100% VT-3 visual examination of all Class 1, 2, and 3 supports other than piping supports.

Licensee's Code Relief Request: The licensee requested relief from performing 100% of the Code-required VT-3 visual examinations for the reactor vessel supports.

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

"The Callaway Reactor Vessel is supported by two cold leg nozzles and two hot leg nozzles. There is a support assembly at each of these nozzles that consists of a nozzle weld build up, shoe plate, air cooled box, and steel support structure embedded in the primary shield wall. Figure ISI-03* depicts these support assemblies. As shown in the figure, only the nozzle weld build up and shoe plate are completely accessible for a visual VT-3 examination. The majority of the air cooled box and the entire steel support structure are located beneath a steel walk plate and only the top of the air cooled box is directly accessible. An additional 20 to 30 percent of the air cooled box and a very small percentage of the steel support structure would be made accessible if the steel walk plate and insulation were removed.

"The Reactor Vessel supports are located in a confined space below the refueling pool permanent seal ring. The area can only be accessed through four seal ring hatches. In addition to difficult access, the radiation in the area is between 1.5 to 2.0 man-rem per hour. It is estimated that the removal and re-installation of the walk plate and insulation in this confined space, combined with the visual VT-3 examination, would result in an exposure of approximately 36 man-rem. Removal of the walk plate and insulation under these conditions to increase the examination of the air cooled box by approximately 20 to 30 percent and a very small percentage of the steel support structure is considered impractical without a commensurate increase in quality or safety. Based on this, relief is requested from the visual VT-3 examination of the air cooled box and steel support structure that is obstructed by the walk plate and insulation."

In response to a request for additional information the licensee submitted the following:

"With the walk plate and insulation installed only the topmost portion, or approximately 30 percent, of the air cooled box is accessible for visual inspection. As stated in ISI-03, removal of the insulation and walk plate would increase the inspection coverage by 20 to 30 percent or result in visual inspection of approximately 36 to 39 percent of the air cooled box.

"The only additional parts of the Reactor Vessel support accessible to remote visual inspection are the end sections of two I-beam end stiffener plates, and the vessel side of the air cooled box. These accessible parts are loaded in compression with no bending or tension loads. Visual inspection of these parts would provide little, if any, indication of support structural integrity."

*Figure ISI-03 is not included as part of this evaluation.

Licensee's Proposed Alternative Examination (as stated):

"A limited visual VT-3 examination, with the walk plate and insulation installed, shall be performed on the accessible NF portions of the Reactor Vessel support assemblies to satisfy the requirements of Code Case N-491, Table-2500-1, Item No. F1.40. If conditions are discovered during this limited VT-3 examination that do not meet the acceptance standards of N-491, -3400, the walk plate or insulation will, if necessary, be removed to meet the evaluation requirements of N-491, -3112.2 or -3112.3."

Evaluation: Code Case N-491 requires a VT-3 visual examination of the reactor vessel supports. However, due to access restrictions of the support design and high local radiation levels, the licensee proposed to perform a limited visual examination.

The support assembly at each of the reactor vessel nozzles consists of nozzle weld build up, shoe plate, air-cooled box, and steel support structure. The steel support structure is embedded in the primary shield wall, making the VT-3 visual examination of the reactor vessel support impractical to perform to the extent required by the Code. Additionally, to require the licensee to remove the walk plate and insulation and incur approximately 36 man-rem additional exposure to increase coverage of one component of the support by 6-9% would be a burden not commensurate with the increase in safety.

The licensee's proposed alternative, to perform a VT-3 visual examination of the entire nozzle weld build up and shoe plate and approximately 30% of the air-cooled box, will provide reasonable assurance of structural integrity.

Conclusion: Based on the design of the reactor vessel supports, it is impractical to perform the VT-3 visual examination to the extent required by the Code. The licensee's alternative will provide reasonable assurance of operational readiness, therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

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 ISI-02, Augmented Examination of Class 2 Piping per NRC Standard Review Plan, Section 3.6.1

Augmented Requirement: *Standard Review Plan*, Section 3.6.1 requires that circumferential welds in Class 2 high-energy fluid system piping located outside containment and exceeding 1 inch NPS be subject to volumetric examination.

Licensee's Relief Request: The licensee requested relief from the volumetric examination of the following welds, specified by *Standard Review Plan*, Section 3.6.1.

RCP "A" Seal Water Injection Line Welds

2-BG-09-FW387	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW386	1 1/2" Pipe to Valve
2-BG-09-FW385	1 1/2" Pipe to Valve
2-BG-09-FW384	2" x 1 1/2" Reducer to 1 1/2" Pipe

RCP "B" Seal Water Injection Line Welds

2-BG-09-FW432	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW431	1 1/2" Pipe to Valve
2-BG-09-FW430	1 1/2" Pipe to Valve
2-BG-09-FW429	2" x 1 1/2" Reducer to 1 1/2" Pipe

RCP "C" Seal Water Injection Line Welds

2-BG-09-FW417	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW416	1 1/2" Pipe to Valve
2-BG-09-FW415	1 1/2" Pipe to Valve
2-BG-09-FW414	2" x 1 1/2" Reducer to 1 1/2" Pipe

RCP "D" Seal Water Injection Line Welds

2-BG-09-FW402	2" x 1 1/2" Reducer to 1 1/2" Pipe
2-BG-09-FW401	1 1/2" Pipe to Valve
2-BG-09-FW400	1 1/2" Pipe to Valve
2-BG-09-FW399	2" x 1 1/2" Reducer to 1 1/2" Pipe

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

"The sixteen welds listed above are all within portions of the Reactor Coolant Pump Seal Water Injection Lines which are schedule 160, 1 1/2" NPS. A combination of the small pipe diameter and pipe thickness cause the volumetric examinations to yield meaningless results.

"These are stainless steel welds joining 1 1/2 NPS schedule 160 pipe. The wall thickness of the pipe is 0.281 inch and the outside diameter is 1.9 inches. The combination of thin wall thickness and small outside diameter make the results obtained with volumetric nondestructive examination methods (i.e., ultrasonic and radiographic) unreliable, and at best, questionable.

"A surface examination technique, such as a liquid penetrant, is considered by Union Electric to be more adequate to ensure the structural integrity of the RCP seal injection welds than either a radiographic or ultrasonic examination. This position is based on the following:

"1. A volumetric examination on the 1 1/2 NPS seal injection welds goes beyond the current requirements of ASME Section XI Examination Categories B-J, C-F-1, or C-F-2 for Code Class 1 and 2 pipe welds. Code Class 2 pipe welds less than 2 NPS are exempt entirely from nondestructive examination. Only a surface (i.e., liquid penetrant or magnetic particle) examination is required on Code Class 1 pipe welds between 4 NPS and 1 NPS. Code Class 1 Reactor Coolant System (RCS) components operate at nearly the same pressure and at higher temperatures than the RCP Seal Water Injection System at the high energy break locations. It is reasonable to assume that the liquid penetrant examination required to ensure the integrity of small bore RCS pipe welds is equally adequate to ensure the structural integrity of the high energy RCP seal injection welds.

"2. Ultrasonic examination of the RCP seal injection welds is questionable due to transducer near field effects, beam redirection, and excessive sound attenuation in stainless steel welds. The 1/2 vee ultrasonic examination technique is questionable due to transducer near field effects. When performing calibrations on 1 1/2 NPS schedule 160 pipe the 1/2t calibration hole cannot be resolved. Extended beam path techniques are also questionable due to beam redirection and excessive sound attenuation in the weld volume. Recent Appendix VIII performance demonstrations at the EPRI NDE Center reveal that extended beam path techniques performed from one side of the weld are unreliable for detection of the Appendix VIII implanted mechanical and thermal fatigue cracks in stainless steel welds.

"3. Radiography is ideal for detection of construction type weld flaws, however, tight service induced (i.e., crack like) flaws are not typically detected. Radiographic examination for crack like indications is almost totally ineffective unless the indication is aligned exactly perpendicular to the radiation source. In addition, with small diameter pipe, a double wall shot technique is required, thus further reducing the probability of detecting a crack like indication.

"4. There is no history of inside diameter service induced degradation mechanisms, such as Intergranular Stress Corrosion Cracking (IGSCC), on small bore RCP seal injection lines in Pressurized Water Reactor plants. A search on the Nuclear Plant Reliability Data System (NPRDS) revealed only one failure of a pressure boundary in an RCP seal injection line. In this instance the crack initiated from a construction flaw at the weld root due to mechanical vibration. A similar failure is unlikely at the Callaway Plant since the subject RCP seal injection welds are high quality full penetration butt welds installed and tested in accordance with ASME Section III.

"5. The primary flaw initiator for small bore pipe is typically vibration induced fatigue. In the absence of an internal flaw where a crack may propagate, fatigue cracks typically initiate from the pipe outside diameter since the maximum stress due to cyclic bending is located on the outside surface membrane. Furthermore, fatigue failures normally occur at gross structural discontinuities such as socket welded joints. The subject RCP seal injection welds are all full penetration single vee butt joints."

Licensee's Proposed Alternative Examination (as stated):

"As an alternative to the requirement of NRC Standard Review Plan, Section 3.6.1, liquid penetrant examinations shall be performed on all sixteen of the subject welds. In addition, a visual VT-2 examination shall be performed on these welds as specified in ASME Section XI."

Evaluation: *Standard Review Plan*, Section 3.6.1 requires volumetric examination of the subject RCP seal water injection piping welds. The licensee proposed to perform a surface examination of these welds.

The subject welds join 1 1/2 inch NPS, Schedule 160, stainless steel pipe. The wall thickness of the pipe is 0.281 inch and the outside diameter is 1.9 inches. The thin wall and small outside diameter cause transducer near field effects, beam redirection, and excessive sound attenuation in stainless steel welds, making the augmented volumetric examinations impractical to perform.

The augmented requirements are more stringent than ASME Section XI, Examination Categories B-J, C-F-1, or C-F-2 requirements for Code Class 1 and 2 pipe welds. Code Class 2 pipe welds less than 2 inch NPS are exempt entirely from nondestructive examination, and only a surface examination is required on Code Class 1 pipe welds between 4 inch NPS and 1 inch NPS. Code Class 1 Reactor Coolant System (RCS) components operate at nearly the same pressure and at higher temperatures than the RCP Seal Water Injection System at the high energy break locations. It is, therefore, reasonable to assume that the liquid penetrant examination required to ensure the integrity of small bore RCS pipe welds is equally adequate to ensure the structural integrity of the high energy RCP seal injection welds.

Conclusion: The augmented requirements are impractical and the licensee's alternative will ensure piping integrity. It is, therefore, recommended that the licensee's alternative be found acceptable.

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 (No requests for relief)

3.4 Pressure Tests

3.4.1 Class 1 System Pressure Tests (No requests for relief)

3.4.2 Class 2 System Pressure Tests

3.4.2.1 Request for Relief ISI-09, Examination Category C-H, Items C7.30, C7.40, C7.70, and C7.80, Pressure Testing Class 2 Components at Containment Penetrations

Code Requirement: Examination Category C-H, Items C7.30, C7.40, C7.70, and C7.80, in conjunction with Code Case N-498-1, require system leakage testing of Class 2 piping and valves once each inspection period.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required system leakage test for the following Code Class 2 piping and valves at containment penetrations where the balance of the system is outside the scope of Section XI.

LINE NUMBER	PENETRATION	DESCRIPTION
BB-103-HCB-1"	P-62	Pressurizer relief tank gas line
BL-028-HCB-3"	P-25	Reactor water storage tank to RCP standpipes
BM-053-HBB-3"	P-78	Steam Generator drain
EC-067-HCB-6"	P-53	Fuel pool cooling return
EC-072-HCB-6"	P-54	Refueling pool to fuel pool cooling pump suction
EC-081-HCB-3"	P-55	Refueling pool to fuel pool skimmer pump
EM-071-BCB-3/4"	P-92	SIS pump test line return to Reactor water storage tank
GP-003-HBB-1"	P-51	ILRT test connection lines
GP-005-HBB-1"	P-51	ILRT test connection lines
GS-025-HBB-6"	P-65	Hydrogen purge subsystem to ESF filters
GT-007-HBB-36"	V-160	Containment shutdown purge
GT-004-HBB-36"	V-161	Containment shutdown purge
GT-029-HBB-18"	V-161	Containment shutdown purge
GT-034-HBB-18"	V-160	Containment shutdown purge

LINE NUMBER	PENETRATION	DESCRIPTION
GT-033-HBB-18"	V-160	Containment shutdown purge
GT-030-HBB-18"	V-161	Containment shutdown purge
HB-015-HCB-3"	P-26	From reactor coolant drain tank heat exchanger
HB-025-HBB-3/4"	P-44	Reactor coolant drain tank to waste gas compressor
HD-015-HBB-2"	P-43	Auxiliary steam for reactor vessel head decontamination
KA-244-HCB-1 1/2"	P-30	Compressed air
KA-259-HCB-1 1/2"	P-30	Compressed air
KA-051-HBB-5"	P-63	Service air
KA-261-HBB-1"	P-63	Reactor building service air
KA-732-HBB-1"	N/A	Personnel hatch penetration test lines
KA-733-HBB-1"	N/A	Personnel hatch penetration test lines
KB-001-HCB-2"	P-98	Breathing air
KC-560-HBB-4"	P-67	Fire protection
LF-842-HCB-6"	P-32	Containment building floor drain header
SJ-002-BCB-1"	P-69	Nuclear sampling from pressurizer vapor space
SJ-003-ECB-1"	P-95	Nuclear sampling from accumulator tanks
SJ-001-BCB-1"	P-93	Loop 1 hot leg liquid sample to PASS
SJ-029-BCB-1"	P-93	Loop 1 hot leg liquid sample to PASS
SJ-021-BCB-1"	P-64	Loop 3 hot leg & pressurizer liquid sample to PASS
SJ-024-BCB-1"	P-57	PASS to reactor drain tank
SJ-024-BCB-1"	P-58	PASS to reactor drain tank

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

"Based on the discussion below, these pressure tests are considered redundant and without a compensating increase in the level of quality and safety.

"The lines listed above are portions of non-safety related piping systems that penetrate the primary reactor containment. At each containment penetration, the process pipe is classified Code Class 2 and provided with isolation valves that are either locked shut during normal operation, capable of automatic closure, or capable of remote closure to support the containment safety function. The piping and valves are considered part of the primary reactor containment and upgraded to Code Class 2 at the penetration only to support the primary reactor containment safety function. Except for this, the lines listed above provide no safety function.

"The primary reactor containment integrity, including all containment penetrations, is periodically verified by performing leakage tests in accordance with a 10 CFR 50, Appendix J. Each of the Code Class 2 lines listed above and their associated isolation valves are tested during an Appendix J, Type A, B, or C leakage test at a pressure not less than 48.2 psig. The Type A leakage test is performed three times in a ten year interval and the Type B and C leakage tests are performed at intervals not greater than 24 months. Performance of these Appendix J leak tests will verify the integrity of the subject Code Class 2 lines at each respective penetration. The performance of ASME Section XI, Examination Category C-H pressure tests on these same lines will provide little, if any, additional verification of primary reactor containment integrity. Based on this, the performance of Examination Category C-H pressure tests on these lines is considered by Union Electric to be unnecessary and provides a negligible increase in the level of quality or safety."

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

"Union Electric shall perform 10 CFR 50, Appendix J leakage tests on the primary reactor containment penetration lines listed above, and on their associated valves, in accordance with Callaway Technical Specification 3/4.6."

<u>Evaluation</u>: The system leakage test required by Examination Category C-H provides periodic verification of the leak-tight integrity of Class 2 piping systems or segments once every 40 months. Pipe segments from non-code class systems that penetrate containment are designed and examined as Class 2 pipe to protect the integrity of containment. The Appendix J pressure testing provides periodic verification of the leak-tight integrity of the primary reactor containment, and of systems and components that penetrate containment. The Appendix J test frequency provides assurance that the containment pressure

boundary is being maintained at an acceptable level while monitoring for deterioration of seals, valves, and piping. Appendix J requires that three Type A tests be performed at approximately equal intervals during the 10-year ISI interval, with the third test done while shutdown for the 10-year plant ISI. Appendix J also requires Type B and C test be performed during each refueling outage, but in no case at intervals greater than 2 years.

The Class 2 containment isolation valves (CIVs) and connecting pipe segments must withstand the peak calculated containment internal pressure related to the maximum design containment pressure. The containment penetration piping is classified as Class 2 because it is part of the containment pressure boundary, and because containment integrity is the only safety-related function performed by this piping. Therefore, it is logical to test the penetration piping portion of the associated system to the Appendix J criteria. The INEL staff finds that the pressure-retaining integrity of the CIVs and connecting piping and their associated safety functions may be verified with an Appendix J, Type C test if it is conducted at the peak calculated containment pressure.

IWC-5210(b) requires that where air or gas is used as a testing medium, the test procedure shall include methods for detection and location of through-wall leaks in system components. Because an Appendix J, Type C test most likely uses air as a testing medium, the licensee's test procedure should meet the above requirement for the CIVs and pipe segments between the CIVs.

<u>Conclusion</u>: The INEL staff concludes that compliance with Appendix J would provide an acceptable level of quality and safety for the subject Class 2 piping that penetrates containment, where the balance of the piping system is non-code class. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i), provided that the tests are conducted at peak calculated

containment pressure and that the test procedures include methods for detection and location of through-wall leakage in CIVs and pipe segments between the CIVs.

3.4.3 Class 3 System Pressure Tests (No requests for relief)

3.4.4 General

3.4.4.1 Request for Relief ISI-08, IWA-5250(a)(2), System Pressure Test Corrective Measures

Code Requirement: IWA-5250(a)(2) states that if leakage occurs at a bolted connection during a system pressure test, then all bolting must be removed and a VT-3 visual examination performed to detect corrosion.

Licensee's Code Relief Request: The licensee requested relief from performing the Code-required removal and VT-3 visual examination of bolting if leakage occurs during a system pressure test of Class 1, 2, and 3 systems.

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

"A leaking environment at a bolted connection may be a significant variable in the degradation mechanism of bolted connections. However, leakage is not the only variable, and in some cases may not be the degradation mechanism. Other variables to be considered are: bolting materials, leaking medium, duration of the leak, and orientation of the leak (not all the bolts may be wetted). These variables are important to consider before disassembling a bolted connection for a visual VT-3 examination. Removal of bolting at a mechanical connection may not be the most prudent decision and may cause undue hardship without a compensating increase in the level of quality or safety. Union Electric proposes an alternative to the requirements of IWA-5250(a)(2) that will provide an equivalent level of quality and safety at Class 1, 2, and 3 bolted connections."

Licensee's Proposed Alternative Examination (as stated):

"Leakage discovered at a bolted connection by visual VT-2 examination during system pressure test will be evaluated to determine the susceptibility of the bolting to corrosion and potential future failure. The evaluation will, as a minimum, consider the following variables:

"1) Location of leakage
"2) History of leakage
"3) Bolted connection materials
"4) Visual evidence of corrosion with the connection assembled
"5) Corrosiveness of the process fluid
"6) History and studies of similar bolted material in a similar environment
"7) Other components in the vicinity that may be degraded due to the leakage

"When evaluation of the variables above indicates the need for further evaluation, the bolt closest to the source of leakage will be removed, receive a visual VT-3 examination, and be evaluated in accordance with IWA-3100(a). If the leakage was identified with the bolted connection in service and evaluation supports continued service, this VT-3 examination may be deferred to the next outage of sufficient duration. When the removed bolt has evidence of rejectable degradation, all remaining bolts shall be removed and subsequently receive a visual VT-3 examination and evaluated in accordance with IWA-3100(a)."

Evaluation: In accordance with the 1989 Edition of the Code, when leakage occurs at bolted connections, all bolting is to be removed for VT-3 visual examination. In lieu of the Code-required removal of bolting to perform a VT-3 visual examination, the licensee has proposed to evaluate the bolting to determine the susceptibility of the bolting to corrosion. If the bolting is susceptible to corrosion or the initial evaluation indicates the need for a more in-depth evaluation, the bolt closest to the source of leakage will be removed, VT-3 examined, and evaluated in accordance with IWA-3100(a).

The INEL staff believes that the licensee's proposed alternative to bolting removal is based on sound engineering judgement. As a result, it is believed that the licensee's proposed alternative to the Code-required removal of bolting at a joint when leakage occurs will provide an acceptable level of quality and safety.

Conclusion: It is reasonable to conclude that the licensee's proposed alternative, to evaluate the bolting at a leaking connection, will detect degradation of bolting, if present. Therefore, it is recommended that the proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i), provided if the bolting is susceptible to corrosion or the initial evaluation indicates the need for a more in-depth evaluation, the bolt closest to the source of leakage will be removed, VT-3 examined, and evaluated in accordance with IWA-3100(a).

3.4.4.2 Request for Relief ISI-07, IWA-5242(a), System Pressure Tests for Insulated Bolted Connections

Code Requirement: IWA-5242(a) states that for systems borated for the purpose of controlling reactivity, insulation shall be removed from pressure-retaining bolted connections for a direct VT-2 visual examination.

Licensee's Code Relief Request: The licensee requested relief from the Code-required removal of insulation for VT-2 visual examinations of bolted connections in borated systems.

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

"Union Electric believes that removal of insulation at bolted connections in borated systems solely for a visual VT-2 examination is impractical for the reasons listed below:

"1) Code Class 1 and 2 systems borated for the purpose of controlling reactivity are extensive and large systems covering many areas and elevations. Scaffolding will be required to access many of the bolted connections. In addition, many of the bolted connections are located in difficult to access areas and in medium to high radiation areas. Insulation removal combined with scaffolding requirements will increase the financial cost, personnel exposure, and generation of radwaste associated with performance of visual VT-2 examinations.

"2) The visual VT-2 examination of Class 1 systems, primarily the Reactor Coolant System (RCS) piping and components, is performed between plant mode 3 and 2 ascending. As required by IWB-5221, the RCS is at a normal operating pressure of

2235 psig. Between modes 3 and 2 ascending, the temperature is approximately 557°F. Performance of a visual VT-2 examination, installation of insulation, and disassembly of scaffolding at bolted connections under these operating conditions is a personnel safety hazard. The visual VT-2 examination is a critical path activity and normally has a duration of six to eight hours. Since the majority of Class 1 piping is inside the containment building bio-shield wall, insulation installation and disassembly of scaffolding will add to the outage duration. Critical path cost is currently estimated at $207,000 per day."

Licensee's Proposed Alternative Examination (as stated):

"Union Electric believes that the established Callaway programs described below in addition to the alternative examination proposed below, provide an acceptable level of safety and quality for bolted connections in systems borated for the purpose of controlling reactivity.

"1) In response to NRC Generic Letter 88-05, Union Electric has established a program for Engineering to inspect all boric acid leaks discovered in the containment building and to evaluate the impact of those leaks on carbon steel or low alloy steel components. All evidence of leaks, including boric acid crystals or residue, is inspected and evaluated regardless of whether the leak was discovered at power or during an outage. Issues such as the following are considered in the inspection and evaluation: 1) evidence of corrosion or metal degradation, 2) effect the leak may have on the pressure boundary, 3) possibility of boric acid traveling along the inside of insulation on piping, and 4) possibility of dripping or spraying on other components. Based on this evaluation, Engineering initiates appropriate corrective actions to prevent reoccurrence of the leak and to repair, if necessary, any degraded materials or components.

"2) In addition to the nondestructive examinations required by ASME Section XI, Union Electric has committed to the bolting examination requirements of NRC Bulletin 82-02. In accordance with this Bulletin, at least two nondestructive examination techniques (e.g., ultrasonic, liquid penetrant, magnetic particle, or visual VT-1) are performed on bolted connections of the following components: Steam Generator primary manways, Pressurizer primary manway, Pressurizer safety valves, and a total of 22 Reactor Coolant System isolation valves that are greater than 6" NPS. As a minimum, two nondestructive examination techniques are used whenever the bolted connection of one of the subject components is dissembled for maintenance or other inspection. These additional examinations ensure that degradation mechanisms such as Stress Corrosion Cracking or corrosion do not go undetected in bolted connections critical to reactor safety.

"3) The only carbon steel components at the Callaway Plant that are in systems borated for the purpose of controlling reactivity are clad with stainless steel. Specifically, these clad components are the Reactor Vessel, Steam Generators (primary side), and Pressurizer. All other piping and components in borated systems that are within inservice inspection boundaries are fabricated of stainless steel. There is substantial information, such as EPRI NP-5679, attesting to the resistance of stainless steels to boric acid corrosion. To ensure that degradation mechanisms in stainless steels are mitigated, Union Electric maintains a program at the Callaway Plant that controls materials (insulation, thread lubricant, boron, etc.) that may come in contact with safety related components, including bolting. This program ensure that impurities are not present in concentrations that would promote development of Stress Corrosion Cracking in stainless steel bolted connections.

"Bolted connections in systems borated for the purpose of controlling reactivity shall receive a visual VT-2 examination during the system pressure tests of IWB-5000 and IWC-5000 with the insulation installed. If evidence of leakage is detected, either by discovery of active leakage or evidence of boric acid crystals, the insulation shall be removed and the bolted connection shall be re-examined and, if necessary, evaluated in accordance with the corrective measures of Subarticle IWA-5250.

"If insulation is removed for planned maintenance, repair, or other inspection at a bolted connection in a system borated for the purpose of controlling reactivity, a visual VT-2 examination shall be performed on the bolted connection prior to disassembly and, if evidence of leakage is discovered, evaluation in accordance with the corrective measures of Subarticle IWA-5250."

Evaluation: Paragraph IWA-5242(a) 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. Based on the review of the licensee's basis for relief and proposed alternative, it has been determined that the licensee's approach to the Code-required insulation removal is acceptable provided that the licensee's alternative is supplemented by the following:

1) The licensee shall remove all existing removable insulation each refueling outage at bolted connections in systems borated

for the purpose of controlling reactivity and perform a VT-2 visual examination for evidence of leakage; and

2) Where non-removable insulation exists at bolted connections, the licensee may visually examine the joint without removing the insulation provided that a 4-hour hold time is satisfied prior to the VT-2 visual examination.

Conclusion: The INEL staff believes that the proposed alternative provides an acceptable level of quality and safety with the conditions stated above. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(i), provided that the licensee satisfy the stated conditions.

3.5 General

3.5.1 Ultrasonic Examination Techniques

3.5.1.1 Request for Relief ISI-01, Qualification of Nondestructive Examination Personnel for Ultrasonic Examination

Code Requirement: IWA-2311(b) requires that the training, qualification, and certification of ultrasonic examination personnel comply with the requirements specified in Appendix VII.

Licensee's Code Relief Request: Relief is requested from implementation of Appendix VII until the performance demonstration requirements of Appendix VIII are fully implemented.

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

"Implementation of Appendix VII prior to full implementation of Appendix VIII is considered impractical and without a compensating increase in quality and safety.

"Appendix VII was first introduced in the 1988 Addenda to Section XI. This Appendix represents a dramatic change from previous Code editions and current industry practices in the requirements for qualification of ultrasonic examination personnel. New training programs must be developed and taught by trained instructors, employer's written practices must be completely rewritten, examination question banks must be developed, flaw specimens containing actual or simulated flaws must be acquired, and performance demonstrations (practical examinations) must be completed.

"Implementation of Appendix VII will require a substantial industry effort. Although work is progressing towards compliance with Appendix VII, full implementation has not yet been achieved. Since Appendix VII provides for use of specimens prepared for ultrasonic performance demonstrations per Appendix VIII, many NDE vendors are developing these two programs concurrently in order to avoid duplicated effort. Though currently not required, the nuclear industry anticipates that the Appendix VIII performance demonstration requirements will be mandated by a backfit ruling in the Federal Register. In anticipation of this ruling, the Performance Demonstration Initiative (PDI) Committee is currently leading an industry wide effort to implement Appendix VIII. The tentative completion dates for pipe weld performance demonstrations and reactor vessel performance demonstrations are January of 1996, and January of 1997, respectively.

"The Union Electric Company intends to fully implement Appendix VII when the performance demonstrations of Appendix VIII are mandated by a back-fit ruling in the Federal Register."

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

"The Callaway Plant shall utilize ultrasonic examination personnel qualified in accordance with the requirements of IWA-2300, except for IWA-2311(b). The additional Appendix VII training, qualification, and certification requirements referenced in IWA-2311(b) shall be fully implemented when the performance demonstrations of Appendix VIII are mandated by a ruling in the Federal Register."

<u>Evaluation</u>: Appendix VII was incorporated into the Code in 1988 to enhance the abilities of ultrasonic examiners. This appendix places controls on a wide variety of classroom and laboratory training. Although Appendices VII and VIII are both designed to improve flaw detection via ultrasonic examinations, their concurrent implementation is not necessary. Certain requirements of Appendix VIII may strengthen the efforts of Appendix VII, but they are not necessary for its implementation.

The INEL staff believes that the licensee has had sufficient time to develop an Appendix VII program. The licensee states that "Implementation of Appendix VII will require a substantial industry effort"; however, other utilities have already committed to following Appendix VII. Although Appendix VIII will further improve flaw detection using ultrasonics, it's implementation is not required in conjunction with Appendix VII. An Appendix VII program will increase quality and safety and is not considered impractical.

Conclusion: Sufficient technical justification has not been provided and, therefore, it is recommended that the proposed alternative be denied.

3.5.2 Exempted Components (No requests for relief)

3.5.3 Other

3.5.3.1 Request for Relief ISI-04, Request for Authorization to Use ASME Code Case N-524

Code Requirement: Code Cases are periodically published by ASME to either clarify the intent of the Code rules or to provide rules and regulations for circumstances that are not covered by existing Code rules and need to be addressed in a timely manner. Use of non-mandatory Code Cases is allowed after general acceptance by the NRC staff and incorporation into Regulatory Guide 1.147. Pursuant to 10 CFR 50.55a, Code Cases not incorporated into Regulatory Guide 1.147 may be used provided specific authorization is granted.

Licensee's Code Relief Request: This relief request is for authorization to use ASME Code Case N-524, *Alternative Examination Requirements for Longitudinal Welds in Class 1 and 2 Piping, Section XI, Division 1* in the Callaway Nuclear Power Plant, Unit 1, ISI Program.

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

"Based on the reasons stated below, the performance of surface and volumetric examination on longitudinal piping welds has a negligible compensating effect on the quality or safety of Class 1 and 2 piping. In addition, there is little if any, technical benefit associated with the performance of these examinations, but they result in a substantial man-rem exposure and cost.

"1) Throughout the nuclear industry, there has been no evidence of rejectable service induced flaws being attributed to longitudinal piping welds.

"2) During the first inservice inspection interval at the Callaway Plant, no inservice flaws have been detected in longitudinal piping welds.

"3) There are distinct differences between the processes used in the manufacturing of longitudinal and circumferential welds which enhance the integrity of longitudinal welds. First, longitudinal welds are typically manufactured under controlled shop conditions whereas circumferential welds are produced in the field under less than ideal conditions. Secondly, longitudinal welds usually undergo heat treatment in the shop which improves their material properties and relieves the residual stresses created by welding. Finally, shop manufacturing inspections can be performed under more favorable conditions which further increase the confidence level of the longitudinal weld quality.

"4) During field installation of piping, the ends of the longitudinal welds may be affected during welding of the intersecting circumferential field welds. This small area falls within the circumferential weld inspection boundaries. Therefore, the ends of the longitudinal welds will still be subject to examination.

"5) From industry-wide standpoint, there has been no evidence of longitudinal weld defects compromising safety at nuclear generating facilities.

"6) No significant loading conditions or known material degradation mechanisms have become evident to date which specifically relate to longitudinal seam welds in nuclear plant piping.

"7) There is a significant accumulation of man-rem exposure and cost associated with the inspection of Class 1 and 2 longitudinal piping welds.

"8) The alternative examinations proposed below provide an acceptable level of quality and safety without causing undue hardship or difficulties."

Licensee's Proposed Alternative Examination (as stated):

"Surface and volumetric examinations shall be performed, as applicable, on the length of the longitudinal weld that is normally examined during inspection for the intersecting circumferential weld(s). The volumetric examination at the intersection of circumferential and longitudinal welds will include both transverse and parallel scans within the length of the longitudinal welds that falls within the circumferential weld examination boundary."

Evaluation: The licensee requested relief from performing the surface and volumetric examinations, as applicable, of the longitudinal welds in Class 1 and Class 2 piping to the extent required by the Code.

The licensee's proposed alternative is to follow ASME Code Case N-524, which requires a surface examination of the longitudinal weld in conjunction with the examination of the circumferential weld and volumetric examination of circumferential welds, including both transverse and parallel scans. As a result, the portion of longitudinal weld that falls within the circumferential weld examination area will be examined. The examination records for the circumferential weld document the extent of longitudinal weld examined.

Code Case N-524 is based on the position that longitudinal welds are unlikely to fail, as the result of fabrication controls and lack of susceptibility to the conditions that lead to failure. The potentially critical portions of the longitudinal welds are the portions that intersect circumferential welds; these regions will be examined in conjunction with the circumferential welds. However, a possible error in the use of this Code Case could occur if it were applied to ferritic welds where there is not normally a Code-required scan for reflectors located transverse to the circumferential welds. The use of this Code Case is contingent on the volumetric examinations of the adjacent circumferential welds providing scanning for reflectors transverse to the weld. Based on the quality of longitudinal welds and the extent of examinations performed, this provides an acceptable level of quality and safety.

Conclusion: Examination of Class 1 and Class 2 longitudinal piping welds in accordance with Code Case N-524 results in 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), provided that the volumetric examinations of the adjacent circumferential welds provides scanning for reflectors transverse to the weld.

3.5.3.2 Request for Relief ISI-06, Request Authorization to Use ASME Code Case N-509

Code Requirement: Code Cases are periodically published by ASME to either clarify the intent of the Code rules or to provide rules and regulations for circumstances that are not covered by existing Code rules and need to be addressed in a timely manner. Use of non-mandatory Code Cases is allowed after general acceptance by the NRC staff and incorporation into Regulatory Guide 1.147. Pursuant to 10 CFR 50.55a, Code Cases not incorporated into Regulatory Guide 1.147 may be used provided specific authorization is granted.

Licensee's Code Relief Request: The licensee requested authorization to use Code Case N-509, "Alternate Rules for the Selection and Examination of Class 1, 2, and 3 Integrally Welded Attachments, Section XI, Division 1".

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

"1) During the first inservice inspection interval at the Callaway Plant, no inservice flaws were detected in integrally welded attachments which would affect safety or compromise the integrity of the plant.

"2) Within the nuclear industry, failures in integral attachments have been very rare and have not affected plant safety. When failures or inservice defects are found in integral attachments, they are usually associated with a support which has been damaged during operation. Therefore, flawed or broken integral attachments are typically detected during the investigation of damaged supports rather than during Code scheduled inservice inspections. One purpose of the alternative examination proposed below is to focus the inspection of integral attachments on those instance where the associated supports show signs of damage. This will increase the likelihood of locating damaged integral attachments.

"3) There is a significant amount of man-rem exposure and cost associated with the scheduled inspection of Class 1, 2, and 3 integral attachments.

"4) Unlike ASME Section XI, the alternate examinations proposed below do not impose a minimum thickness requirement for the inspection of an integral attachment. Therefore, a greater population of integral attachments will be available for inspection because inspections will not be limited to thick attachments. This provision improves the quality and safety level established by these examinations.

"5) The alternate examinations proposed below provide an acceptable level of quality and safety without causing undue hardship or difficulties."

Licensee's Proposed Alternative Examination:

The licensee has proposed to follow Code Case N-509.

Evaluation: The licensee proposed to apply the requirements of Code Case N-509 to the selection and examination of integral attachments on Code Class 1, 2, and 3 piping and components. This is in lieu of the existing Code requirement to examine 100% of the non-exempt Class 1, 2 and 3 integrally welded attachments.

It has been determined that the notes of the Code Case N-509 examination tables could be misinterpreted, allowing selection of component supports for examination, per IWF of the 1989 Edition with the 1990 Addenda, that do not contain any welded attachments. Thus, no welded attachments would be required to be inspected. The INEL staff believes Code Case N-509 should be enhanced to ensure this does not occur. Therefore, to use Code Case N-509, the licensee should schedule a minimum of 10% of all integral attachments in non-exempt Code Class 1, 2, and 3 systems.

Conclusion: The licensee's proposed alternative, to implement Code Case N-509 for the examination of integral attachments, should provide an acceptable level of quality and safety provided that a minimum 10% sample of all non-exempt Code Class 1, 2, and 3 integral attachments is examined. Therefore, it is recommended that the licensee's proposed alternative be authorized, pursuant to 10 CFR 50.55a(a)(3)(i), with the provision that a minimum 10% sample of all non-exempt Code Class 1, 2, and 3 integral attachments be examined.

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 case of Request for Relief ISI-03, the licensee has demonstrated that specific Section XI requirements are impractical; it is 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), the proposed alternatives for Requests for Relief ISI-04, ISI-06, ISI-07, ISI-08 and ISI-09 will provide an acceptable level of quality and safety in lieu of the Code-required examinations and are recommended to be authorized only if the licensee satisfies the conditions stated in the request for relief evaluations above.

Pursuant to 10 CFR 50.55a(a)(3)(ii), it is concluded that for Request for Relief ISI-05 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 if the licensee satisfies the conditions stated in the request for relief evaluation above.

For Request for Relief ISI-02, the proposed alternative will ensure piping integrity and it is recommended that it be found acceptable.

For Request for Relief ISI-01, it is concluded that the licensee has not provided sufficient justification to support the determination that the Code requirement is impractical or that compliance with the Code requirement would result in hardship. Therefore, in this case it is recommended that relief be denied.

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 Callaway Nuclear Power Plant, Unit 1, 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 examination requirements.

Based on the review of the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, 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 ISI-01.

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. *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted October 10, 1994.

4. NUREG-0800, *Standard Review Plans*, 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 23, 1995, L. R. Wharton (NRC) to D. F. Schnell (Union Electric Company) containing request for additional information on the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*.

6. Letter, dated August 18, 1995, D. F. Schnell (Union Electric Company) to Document Control Desk (NRC) containing the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1.

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

8. NRC Regulatory Guide 1.14, *Reactor Coolant Pump Flywheel Integrity*, Revision 1, August 1975.

9. IE Bulletin 79-13, *Cracking in Feedwater System Piping*, dated August 30, 1979.

10. IE Bulletin 88-09, *Thimble Tube Thinning in Westinghouse Reactors*, dated July 26, 1988.

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11. ABSTRACT (200 words or less)

This report presents the results of the evaluation of the *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted August 18, 1995 including the requests for relief from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI requirements that the licensee has determined to be impractical. The *Callaway Nuclear Power Plant, Unit 1, Second 10-Year Interval Inservice Inspection Program Plan*, Revision 1, is evaluated in Section 2 of this report. The Inservice Inspection (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 reviews. The requests for relief are evaluated in Section 3 of this report.

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