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Technical Evaluation Report on the Third 10-year Interval Inservice Inspection Program Plan: Omaha Public Power District, Fort Calhoun Station, Unit 1, Docket Number 50-285

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ABSTRACT

This report presents the results of the evaluation of the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3, submitted August 28, 1995, 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 *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3 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.

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SUMMARY

The licensee, Omaha Public Power District, has prepared the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, through Revision 3, to meet the requirements of the 1989 Edition of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, except that the extent of examination for Code Class 1 piping welds has been determined by the 1974 Edition through Summer 1975 Addenda (74S75) as permitted by 10 CFR 50.55a(b). The third 10-year interval began September 26, 1993, and ends September 25, 2003.

The information in the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted November 13, 1992, 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 to complete the review. The licensee provided the requested information in several submittals, including the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted September 30, 1994. As a result of the review of Revision 1, another RAI was prepared; the licensee provided the requested information in a submittal dated March 9, 1995. On May 10, 1995, a meeting between the NRC and the licensee was held to discuss the ISI program. Based on this discussion, the licensee submitted Revision 3[1] to the ISI program by letter dated August 28, 1995.

Based on the review of the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, through Revision 3, the licensee's responses to the Nuclear Regulatory Commission's RAIs, 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 *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3.

[1]Revision 2 was not submitted for evaluation.

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TECHNICAL EVALUATION REPORT ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN: OMAHA PUBLIC POWER DISTRICT, FORT CALHOUN STATION, UNIT 1, DOCKET NUMBER 50-285

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, Omaha Public Power District, has prepared the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3, (Reference 3) to meet the requirements of the 1989 Edition of the ASME Code, Section XI, except that the extent of examination for Class 1 piping welds has been determined by the 1974 Edition through Summer 1975 Addenda as permitted by 10 CFR 50.55a(b). The third 10-year interval began September 26, 1993, and ends September 25, 2003.

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 *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, through Revision 3, 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 November 13, 1992, the licensee submitted the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0 (Reference 5). By letter dated July 12, 1993 (Reference 6), the NRC requested additional information in order to complete the review of the ISI Program Plan. The requested information was provided by the licensee in letters dated September 10, 1993, March 23, 1994, and September 30, 1994, (References 7, 8, and 9). In these responses, the licensee, Omaha Public Power District, addressed the original RAI questions and submitted Revision 1 to the Program Plan (Reference 10). As a result of the review of Revision 1, the NRC requested additional information by letter dated January 31, 1995 (Reference 11). The licensee responded to this request by letter dated

March 9, 1995 (Reference 12). On May 10, 1995, a meeting between the NRC and the licensee was held. As a result of this meeting, the licensee submitted additional clarification on the ISI program as well as Revision 3 to the ISI program by letter dated August 28, 1995 (Reference 13).

The *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, through Revision 3, 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 the 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 and valves are being evaluated in other reports.

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) *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted November 13, 1992;

(b) *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted September 30, 1994;

(c) *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3, submitted August 28, 1995;

(d) Response to Request for Additional Information, submitted September 10, 1993;

(e) Response to Request for Additional Information, submitted March 23, 1994;

(f) Response to Request for Additional Information, submitted September 30, 1994; and

(g) Response to Request for Additional Information, submitted March 9, 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 September 26, 1993, the Code applicable to the third interval ISI program is the 1989 Edition. As stated in Section 1 of this report, the licensee has prepared the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, through Revision 3, to meet the requirements of 1989 Edition of the Code, except that the extent of examination for Class 1, Examination Category B-J welds has been determined by the requirements of the 1974 Edition through Summer 1975 Addenda as permitted by 10 CFR 50.55a(b).

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 in 10 CFR 50.55a(b). The 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 automated reactor pressure vessel examinations in accordance with Regulatory Guide 1.150, Rev. 1 (Reference 14).

Effective September 8, 1992, 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel", imposed new regulations regarding augmented examination of reactor vessels. As a result of these regulations, all licensees were required to augment their reactor vessel examinations by implementing once, as part of the inservice inspection interval in effect on September 8, 1992, the examination requirements for reactor vessel shell welds specified in Item B1.10 of Examination Category B-A of the 1989 Code. In addition, all previously granted relief for Item B1.10, Examination Category B-A, for the interval in effect on September 8, 1992 was revoked by the new regulation. For licensees with fewer than 40 months remaining in the interval on the effective date, deferral of the augmented examination is permissible with the conditions stated in the regulations. This report evaluates the licensee's submittal on the augmented reactor pressure vessel examinations performed in conjunction with Section XI reactor pressure vessel examinations during the 1992 refueling outage. The licensee submitted Request for Relief 1 for reactor pressure vessel weld examinations where essentially 100 percent coverage was not obtained.

2.3 Conclusion

Based on the review of the documents listed above, no deviations from regulatory requirements or commitments were identified in the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3.

It should be noted, however, that the Code does not require examinations for Code Class 1 integral attachment welds to piping, pumps, and valves during the third and fourth interval when implementing Inspection Program B. Examination of integral attachments in Code Class 2 and 3 systems is required in the third interval. The continued examination of integral attachments in Class 1 systems for the life of the plant is technically prudent. Therefore, it is recommended that an augmented 10% sample of Class 1 integral attachments be scheduled for examination.

3. EVALUATION OF RELIEF REQUESTS

The requests for relief from the ASME Code requirements that the licensee has determined to be impractical for the third 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 1 (Part 1), 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel"

Regulatory Requirement: 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel", requires the examination of essentially 100% of reactor vessel shell welds specified in Item B1.10 of Examination Category B-A of the 1989 Edition of Section XI.

Implementation: The licensee performed the augmented reactor pressure vessel weld examinations on the areas defined by Figures IWB-2500-1 and IWB-2500-2. Presented below are the volumetric coverages obtained on these welds using current examination technology.

WELD IDENTIFICATION	ITEM NUMBER	PERCENT COVERAGE
RPV-SC-B-11	B1.11	91
RPV-SC-C-11	B1.11	80
RPV-SL-A-1	B1.12	93
RPV-SL-B-1	B1.12	93
RPV-SL-C-1	B1.12	93
RPV-SL-A-2	B1.12	100
RPV-SL-B-2	B1.12	100
RPV-SL-C-2	B1.12	100

WELD IDENTIFICATION	ITEM NUMBER	PERCENT COVERAGE
RPV-SL-A-3	B1.12	86
RPV-SL-B-3	B1.12	86
RPV-SL-C-3	B1.12	86

Licensee's Discussion of the Augmented Examination (as stated):

"The subject shell weld, identified as RPV-SC-C-11 (Item B1.10), is located on the RPV (see Attachment 3A, Figure 1)[2]. This weld is examined from the inside of the RPV using an automated ultrasonic test (UT) device. The exams are limited due to the proximity of the permanently attached surveillance capsule holders on the inside of the RPV at 45°, 85°, 95°, 225°, 265°, and 275° positions (see Attachment 3A, Figure 1). The percentages of the Code required volumes obtainable with the automated UT device are shown in Attachment 3A.

"The subject shell welds identified as RPV-SL-A-3, RPV-SL-B-3 and RPV-SL-C-3 (Item B1.10) are also located on the RPV (Attachment 3A, Figure 1). These welds are also examined from the inside of the RPV using an automated UT device. The examinations are limited due to the proximity of the permanently attached flow skirt support lugs on the inside of the RPV at 20°, 60°, 100°, 140°, 180°, 220°, 260°, 300°, and 340° positions. The percentages of the Code required volumes obtainable with the automated UT device are shown in Attachment 3A.

"The examination percentages possible for all B1.10 welds and the B1.30 weld are shown in Attachment 3A. These percentages are shown for all scans performed by the automated UT device. As indicated above, these limitations are due to permanent obstructions which partially shield the areas not completely examined. During previous inspections of the RPV, no recordable indications have been noted in the examinations of the subject welds.

"Examination of the remainder of the Code-required volume on the B1.10 and B1.30 welds would necessitate removal of insulation to gain access into the high radiation environment in order to examine the welds from the exterior of the RPV. OPPD estimates the radiation level would be in excess of 15 R/hr at the exterior examination areas, and that a cumulative exposure of 150 Person-Rem would be necessary to complete the Code-required volumetric examination of the shell welds, Item B1.10 and shell-to-flange weld, Item B1.30.

[2]Figures and attachments are not included with this evaluation.

"The beltline region of the RPV receives a larger radiation fluence and is expected to be more susceptible to radiation induced weld deterioration. Based on weld stress, the limiting welds in the beltline region of the RPV are the axial welds. Relief is being requested for three axial welds (A-3, B-3, and C-3) however, the inaccessible portions of these welds are outside of the beltline region of the RPV. Thus, the inaccessible portions of these welds would receive less radiation and would be less likely to experience the weld deterioration that the UT examinations are attempting to detect. There are inaccessible portions of the beltline circumferential weld (C-11), but this is not a limiting weld. In summary, the portions of the limiting welds that are within the beltline region of the RPV are accessible for UT examination."

"Examination of 100 percent of the RPV weld volumes noted above is not practical. The UT examinations of the accessible portions of the FCS RPV shell welds provide reasonable assurance that public safety is not impaired by the examination limitation described above. This is supported by the following circumstances:

1. All the subject RPV weld areas that are not completely examined are partially shielded by their limiting obstructions; therefore, the inaccessible weld volume should be somewhat less susceptible to deterioration.

2. None of the RPV shell welds that were examined have any recordable indications during their most recent examinations, so it is reasonable to expect that the inaccessible weld volumes are equally free of recordable indications.

3. The inaccessible portions of the axial welds (A-2, B-3 and C-3) are outside of the beltline region of the RPV. The inaccessible portions of these welds would receive less radiation and be less likely to experience the weld deterioration that the UT examinations are attempting to detect. The beltline circumferential weld (C-11) is not a limiting weld. In summary, the portions of the limiting welds that are within the beltline region of the RPV are accessible for UT examination.

4. Excessive radiation levels make examinations from the RPV exterior impractical.

In addition, the licensee provided the following supplemental information as Attachment 3A to the ISI Program Plan on limitations for the Fort Calhoun Reactor Pressure Vessel Welds:

"This attachment describes the ultrasonic examination limitations encountered during the 1992 inservice examination (ISI) of the

Fort Calhoun Station reactor pressure vessel (RPV). The examination was performed by Southwest Research Institute (SWRI) personnel using automated ultrasonic (AUT) scanning equipment, data recording and analysis systems in accordance with a Scan Plan and procedures. Omaha Public Power District approved the plan and procedures, which complied with requirements of the 1980 Edition of the American Society of Mechanical Engineers (ASME) Code, Section XI with Addenda through Winter 1980, and with NRC Regulatory Guide 1.150, Revision 1, Appendix A.

"The scope of the AUT examinations included 100 percent of the accessible weld lengths of the RPV shell welds (Item B1.10) and shell-to-flange weld (Item B1.30) as well as a manual examination of the shell-to-flange weld from the flange surface. The examination coverage obtained was compared to the weld and base metal volumes identified as the examination areas in Section XI, IWB-2500 figures. The ASME Code-specified techniques for RPV examination were augmented by special SWRI-qualified techniques to obtain complete and highly sensitive coverage of the underclad and near-surface material volumes.

"The surveillance tube holders, the flange taper, the flow skirt support lugs, and the core barrel support lugs limited scanning accessibility to the full length and/or width of some areas from the inside surface. The size and location of the flange surface limited the scanning area as well as the angles used in manual examination.

"The examination coverage table in this attachment quantifies the volume of material examined. This percentage is derived from adding together the percentages from the clockwise (CW), counter clockwise (CCW), up and down scans, and dividing by four.

"The maximum credited percentage for the 0°, 45°, or 60° automated examination is 75%, although sound passed through 100% of the thickness. The near 25% of the required volume for automated examinations is credited to the 50/70 scan as it is more sensitive to near surface anomalies."

Licensee's Proposed Alternative Examination:

The licensee has determined that examinations were performed to the extent practical, utilizing state of the art examination technology. The licensee believes that the reactor pressure vessel examination coverages obtained provide an acceptable alternative to essentially 100 percent coverage of each weld required to be examined in accordance with the augmented reactor pressure vessel examination regulation.

Evaluation: For compliance with the augmented reactor vessel examination requirements, licensee's must volumetrically examine essentially 100% (>90%) of each of the Item B1.10 shell welds. Based on the review of the sketch[3] depicting the scanning interferences caused by surveillance capsule holders and the flow skirt supports in the Ft. Calhoun reactor pressure vessel, and supporting information describing limitations to alternatives to increase coverage, it has been determined that essentially 100% coverage of all Item B1.10 reactor pressure vessel welds is not feasible. To achieve complete volumetric coverage, design modifications or replacement of the components with ones of a design providing for complete coverage would be required. Imposition of this requirement would cause a considerable burden on the licensee.

The volumetric examinations of the subject reactor pressure vessel shell welds were performed to the extent possible from the inside surface using mechanized inspection equipment. The licensee satisfied the 100% coverage (>90%) requirement for all of the reactor pressure vessel, Item B1.10, welds except for one circumferential weld and three longitudinal welds. The licensee obtained 80% coverage on the circumferential weld and 86% coverage on each of the longitudinal welds. Based on the significant percentages of coverage obtained, in combination with essentially 100% coverage of the remaining reactor pressure vessel welds subject to examination, the INEL staff believes that degradation, if present would have been detected.

Examination of welds from the external surface of the vessel is not feasible because of limited access between the vessel and the bioshield and the high radiation levels. Assuming access could be obtained, the increase in examination coverage would be insignificant compared with the portion already examined. Therefore, the INEL staff concludes that imposing additional

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[3]Sketch is not included with this evaluation.

examinations from the external surface would result in a considerable burden without an increase in quality and safety.

Based on review of the information submitted by the licensee, it is concluded that the licensee has maximized examination coverage. Therefore, it is recommended that the licensee's alternative to 100% coverage be authorized.

Conclusion: Based on the information submitted, the INEL staff believes that imposing the augmented reactor pressure vessel examinations to the extent required by 10 CFR 50.55a(g)(6)(ii)(A), will result in a burden without a compensating increase in the level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative to examine the Item B1.10 reactor pressure shell welds to the extent possible from the inside surface, be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

3.1.1.2 Request for Relief 1 (Part 2), Examination Category B-A, Items B1.11 and B1.12, Pressure-Retaining Welds in the Reactor Pressure Vessel

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Items B1.11 and B1.12 require that essentially 100% of all circumferential and longitudinal reactor pressure vessel shell welds be volumetrically examined as defined in Figures IWB-2500-1 and IWB-2500-2.

Licensee's Code Relief Request: The licensee requested relief from the Code-required reactor pressure vessel weld volumetric examination coverages defined by Figures IWB-2500-1 and IWB-2500-2 for cases where essentially 100% volumetric coverage was not obtained. Presented below are the volumetric coverages obtainable on the four reactor pressure vessel welds where essentially 100% examination can not be achieved using current examination technology.

WELD IDENTIFICATION	ITEM NUMBER	PERCENT COVERAGE
RPV-SC-C-11	B1.11	80
RPV-SL-A-3	B1.12	86
RPV-SL-B-3	B1.12	86
RPV-SL-C-3	B1.12	86

Licensee's Basis for Requesting Relief: See Request for Relief 1 (Part 1), for licensee's basis.

Licensee's Proposed Alternative Examination: See Request for Relief 1 (Part 1), for licensee's alternative.

Evaluation: The Code requires that all RPV shell welds receive essentially 100% volumetric examination. Based on the review of the sketch[4] depicting the scanning interferences caused by surveillance capsule holders and the flow skirt supports in the Ft. Calhoun reactor pressure vessel and of the supporting information, it has been determined that 100% coverage of the subject reactor pressure vessel welds is impractical. To achieve complete volumetric examination, modifications or replacement of the components with ones of a design providing for complete coverage would be required. Imposition of this requirement would cause a considerable burden on the licensee.

The licensee proposes to perform the Code-required volumetric examinations to the extent practical, achieving the Code-required coverage for all but four welds — one circumferential weld that will receive 80% coverage and three longitudinal welds that will receive 86% coverage. Based on the significant percent of coverage obtainable for the reactor pressure vessel welds, it is reasonable to conclude that degradation, if present, will be

[4]Sketch is not included with this evaluation.

detected. As a result, reasonable assurance of operational readiness will be confirmed.

Conclusion: Because of scanning limitations, complete volumetric examination of the subject reactor pressure vessel welds is impractical. The licensee will obtain sufficient coverage of the subject welds to provide reasonable assurance of operational readiness. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3.1.1.3 Request for Relief 1 (Part 3), Examination Category B-A, Item B1.30, Reactor Pressure Vessel Shell-to-Flange Weld

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.30 requires 100% volumetric examination of the reactor pressure vessel shell-to-flange weld as defined in Figure IWB-2500-4.

Licensee's Code Relief Request: The licensee requested relief from the Code-required 100% volumetric examination of upper shell-to-flange Weld RPV-A-11.

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

"The subject shell-to-flange weld identified as RPV-A-11 (Item B1.30) is located on the RPV. It is examined manually from the flange surface of the RPV and the remaining accessible Code-required volume is examined from the inside of the RPV using an automated UT device. The automated exam is limited due to the circumferential proximity of the flange taper and the permanently attached core barrel support ledge, as shown in Attachment 3A, Figure 1.[5]

Licensee's Proposed Alternative Examination:

"No alternate examinations are proposed at this time by OPPD. Technological improvements are continually evaluated for incorporation into the FCS ISI Program, as applicable."

[5]Attachments are not included with this evaluation.

Evaluation: The Code requires that the subject shell-to-flange weld be 100% volumetrically examined. However, due to scanning limitations and the geometry of the examination area, complete examination is impractical. To achieve complete volumetric coverage, modifications or component replacement with ones of a design providing for complete coverage would be required. Imposition of this requirement would cause a considerable burden on the licensee.

The licensee proposes to examine the shell-to-flange weld to the extent practical, obtaining an estimated 64% coverage. Based on the percent of the Code-required volumetric examination that can be performed, it is reasonable to conclude that degradation, if present, will be detected. As a result, reasonable assurance of operational readiness will be provided.

Conclusion: Complete Code-required volumetric examination of the reactor pressure vessel shell-to-flange weld is impractical due to scanning limitations and the examination area geometry. The licensee proposes to perform the examination to the extent practical, obtaining approximately 64% coverage. Based on the coverage obtainable, it can be concluded that reasonable assurance of operational readiness will be provided. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3.1.2 Pressurizer (No relief requests)

3.1.3 Heat Exchangers and Steam Generators (No relief requests)

3.1.4 Piping Pressure Boundary (No relief requests)

3.1.5 Pump Pressure Boundary (No relief requests)

3.1.6 Valve Pressure Boundary (No relief requests)

3.1.7 General (No relief requests)

3.2 Class 2 Components

3.2.1 Pressure Vessels (No relief requests)

3.2.2 Piping

3.2.2.1 Request for Relief (Appendix 1C), Examination Category C-F-2, Item C5.81, Circumferential Branch Connection Welds Equal To Or Greater Than 2 Inches Nominal Pipe Size

Code Requirement: Table IWC-2500-1, Examination Category C-F-2, Item C5.81 requires 100% surface examination of circumferential branch connection welds as defined in Figures IWC-2500-9 to -13 (as applicable).

Licensee's Code Relief Request: The licensee requested relief from the Code-required surface examinations for the listed branch connection welds:

28-MS-2001/12-BC-1	28-MS-2002/12-BC-1
28-MS-2001/12-BC-2	28-MS-2002/12-BC-2
28-MS-2001/15-BC-1	28-MS-2002/15-BC-1
28-MS-2001/15-BC-2	28-MS-2002/15-BC-3

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

"The Fort Calhoun Updated Safety Analysis Report (USAR), Appendix M, Section 3.5.8 states:

> 'A protective enclosure (has been) provided around the main steam and feedwater lines between the penetration sleeves and the first isolation valves, where a large rupture is postulated.
>
> 'This enclosure, although designed primarily to limit the effects of jet impingement, also serves to minimize the reaction effects of a longitudinal rupture by containing the jet and preventing the formation of an unbalanced external force.'

"In the past, the NRC has conducted a review of the piping exam areas (Docket 50-285, November 10, 1986) and determined that the required examinations were impractical to perform.

"Since one of the eight branch connection welds listed above is required by ASME Section XI, OPPD will substitute a similar branch connection weld on the non-class portion of the main steam line shown on isometric B-86.

"The Code required IWA-5000 system leakage test monitors all the cable wrapped welds."

Licensee's Alternative Examination (as stated):

"OPPD will substitute a similar branch connection weld on the non-class portion of the main steam line."

Evaluation: The Code requires that the subject branch connection welds receive a 100% surface examination. Based on the information provided, it appears that the branch connection welds are inaccessible for surface examination due to a permanent protective enclosure. Therefore, the Code-required surface examinations are impractical. To obtain access, modifications or replacement of these components of ones of a design that provides access would be required. Imposition of this requirement would cause a considerable burden on the licensee.

Conclusions: Because the examination areas are inaccessible, the Code-required surface examination is impractical for the subject branch connection welds. Therefore, it is recommended that relief be granted pursuant to 10 CFR 50.55a(g)(6)(i).

3.2.3 Pumps (No relief requests)

3.2.4 Valves (No relief requests)

3.2.5 General (No relief requests)

3.3 Class 3 Components (No relief requests)

3.4 Pressure Tests (No relief requests)

3.5 General

3.5.1 Ultrasonic Examination Techniques (No relief requests)

3.5.2 Exempted Components (No relief requests)

3.5.3 Other

3.5.3.1 Request for Relief (Appendix 1A), IWA-2600, Weld Reference System

Code Requirement: Section XI, IWA-2620 states that a reference system shall be established for all welds and areas subject to surface or volumetric examination. Each such weld and area shall be identified by a system of reference points. The system shall permit identification of each weld, location of the weld center line, and designation at regular intervals along the length of the weld.

Licensee's Code Relief Request: The licensee requested relief from establishing a weld reference system for all welds and areas subject to surface or volumetric examination.

Licensee's Basis for Requesting Relief: Weld identification at Fort Calhoun Station was not performed during preservice examinations.

Licensee's Proposed Alternative: Weld identifications will be marked at the time the welds are examined per Station Engineering Instruction SEI-27, Revision 3.

Evaluation: The Code requires that the licensee establish a reference system for all welds subject to surface or volumetric examination. The licensee stated that the Code in effect at the time of construction did not require the establishment of a reference system. The licensee is proposing to implement Station Engineering Instruction SEI-27, Revision 3, "Inservice Inspection and Test Program". This procedure specifies that piping welds shall be marked with the system number and weld number as identified on the ISI isometric drawing when the weld is examined. The marking will be performed near the weld by the ISI NDE Technician using low stress stamps or a vibrating etching tool. The ISI Administrator or designee shall independently verify that the weld identification agrees with the isometric drawing and correctly identifies the weld.

The INEL staff believes that this procedure will provide an acceptable level of quality and safety in that scheduled examination areas will be verified and identified correctly at the next scheduled examination.

Conclusion: The licensee's alternative, to use Station Engineering Instruction SEI-27, Revision 3, "Inservice Inspection and Test Program", will provide assurance that welds scheduled for examination will be verified and identified for future reference. As a result, the alternative will provide 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).

3.5.3.2 Request for Relief, Pressure Test Requirements for Repair, Replacements, and Modifications

Code Requirement: IWA-4700(a), Pressure Test, states:

"After repairs by welding on the pressure retaining boundary, a system hydrostatic test shall be performed in accordance with IWA-5000".

IWA-5214, Repairs and Replacements, states:

"A component repair or replacement shall be pressure tested prior to resumption of service if required by IWA-4400 and IWA-4600.

"The test pressure and temperature for a system hydrostatic test subsequent to the component repair or replacement shall comply with the system test pressure and temperature specified in IWB-5222, IWC-5222, and IWD-5223, as applicable to the system which contains the repaired or replaced component."

Licensee's Code Relief Request: The licensee has listed Code Case N-416-1 in its Program Plan. This Code Case contains alternatives to the Code-required hydrostatic test following repairs, replacements, and modifications for Code Class 1, Class 2, and Class 3 systems.

Licensee's Basis for Requesting Relief:

Code Case N-416-1 has been incorporated into the Fort Calhoun Station ISI Program by reference in Part 4, Number 9.

Licensee's Proposed Alternative Examination:

The licensee proposes to implement alternatives to Code hydrostatic tests contained in Code Case N-416-1 following repairs and replacements.

Evaluation: 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 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 for Class 3 components 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 during repair or replacement activities.

Conclusion: Compliance with the Code's 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 implementation of alternatives contained in 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.

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 Requests for Relief 1, Parts 2 and 3, and Appendix 1C, 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), it is concluded that for the Request for Relief in Appendix 1A, the licensee's proposed alternative will (i) provide an acceptable level of quality and safety, or (ii) Code compliance will result in hardship or unusual difficulty without a compensating increase in safety. In these cases, it is recommended that the proposed alternative be authorized.

In addition, pursuant to 10 CFR 50.55a(a)(3), it is recommended that the licensee's proposed alternative be authorized for the Request for Relief from Pressure Test Requirements for Repair, Replacements and Modifications provided that the licensee satisfies the conditions stated in the request for relief evaluation.

In Request for Relief 1, Part 1, the licensee has proposed, as an alternative to regulatory requirement 10 CFR 50.55a(g)(6)(ii)(A), "Augmented Examination of Reactor Vessel", to examine the Item B1.10 reactor pressure vessel shell welds to the extent possible. Based on the information provided and coverages obtained, the INEL believes that imposing requirements to increase coverage would result in a hardship without a compensating increase in the level of quality and safety. Therefore, it is recommended that the licensee's proposed alternative be authorized pursuant to 10 CFR 50.55a(a)(3)(ii).

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 Fort Calhoun Station, Unit 1. 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 in the ISI program plan examination requirements.

Based on the review of the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3, the licensee's responses to the NRC's requests 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.

5. REFERENCES

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

2. American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Division 1:

   1989 Edition
   1974 Edition through Summer 1975 Addenda

3. *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 3, submitted August 28, 1995.

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. *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted November 13, 1992.

6. Letter dated July 12, 1993, S. D. Bloom (NRC) to T. L. Patterson (OPPD) containing NRC Request for Additional Information (RAI).

7. Letter dated September 10, 1993, W. G. Gates (OPPD) to Document Control Desk (NRC) containing the response to the July 12, 1993, Request for Additional Information.

8. Letter dated March 23, 1994, W. G. Gates (OPPD) to Document Control Desk (NRC) containing the response to the July 12, 1993, Request for Additional Information.

9. Letter dated September 30, 1994, W. G. Gates (OPPD) to Document Control Desk (NRC) containing the response to the July 12, 1993, Request for Additional Information.

10. *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 1, submitted September 30, 1994.

11. Letter dated January 31, 1995, S. D. Bloom (NRC) to T. L. Patterson (OPPD) containing NRC Request for Additional Information.

12. Letter dated March 9, 1995, T. L. Patterson (OPPD) to Document Control Desk (NRC) containing the response to the January 31, 1995, Request for Additional Information.

13. Letter dated August 28, 1995, T. L. Patterson (OPPD) to Document Control Desk (NRC) containing clarification to items discussed during the May 10, 1992, meeting with the NRC as well as Revision 3 to the program.

14. NRC Regulatory Guide 1.150, *Reactor Pressure Vessel Beltline Weld Examinations*, Rev. 1, February 1983.

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

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11. ABSTRACT (200 words or less)

This report presents the results of the evaluation of the *Fort Calhoun Station, Unit 1, Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, through Revision 3, submitted August 28, 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 *Fort Calhoun Station, Unit 1, Third 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 reviews. The requests for relief are evaluated in Section 3 of this report.

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