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TECHNICAL EVALUATION REPORT ON THE THIRD 10-YEAR INTERVAL INSERVICE INSPECTION PROGRAM PLAN: CAROLINA POWER AND LIGHT COMPANY, H. B. ROBINSON NUCLEAR PROJECT, UNIT 2, DOCKET NUMBER 50-261

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ABSTRACT

This report presents the results of the evaluation of the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, submitted August 1, 1991, 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 *H. B. Robinson Nuclear Project, Unit 2, 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 reviews by the Nuclear Regulatory Commission (NRC). The requests for relief are evaluated in Section 3 of this report.

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Review of ISI for ASME Code Class 1, 2, and 3 Components

SUMMARY

The Licensee, Carolina Power and Light Company, has prepared the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, to meet the requirements of the 1986 Edition of the ASME 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 February 19, 1992 and ends February 19, 2002.

The information in the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted in August 1991, 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. 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 submittal dated February 13, 1992.

Based on the review of the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the Licensee's response to the Nuclear Regulatory Commission's RAI, 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, it is concluded that the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

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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. The Licensee, Carolina Power and Light Company, has prepared the *H. B. Robinson Nuclear Project, Unit 2 (HBR2), Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0 (Reference 3), to meet the requirements of the 1986 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 February 19, 1992 and ends February 19, 2002.

As required by 10 CFR 50.55a(g)(5), if a licensee determines that certain Code examination requirements are impractical and requests relief from them, the licensee shall submit information and justifications to the Nuclear Regulatory Commission (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 *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval ISI Program Plan*, Revision 0, submitted in August 1991, 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 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 January 13, 1992 (Reference 5), the NRC requested additional information that was required in order to complete the review of the ISI Program Plan. Following conference calls dated January 3, 1992, and January 23, 1992, the requested information was provided by the Licensee in a submittal dated February 13, 1992 (Reference 6). In this submittal, the Licensee, Carolina Power and Light Company, withdrew 9 relief requests and revised Relief Request RR-01.

As a result of another telephone conversation with the Licensee on May 19, 1992, Relief Request No. 07 was withdrawn and Relief Request No. 18 was submitted in a letter dated June 18, 1992 (Reference 7).

The *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval ISI Program Plan* is evaluated in Section 2 of this report. The ISI Program Plan is evaluated for (a) compliance with the appropriate edition/addenda of Section XI, (b) acceptability of examination sample, (c) correctness of the application of system or component examination exclusion criteria, and (d) compliance with ISI-related commitments identified during 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, 1986 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 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) *H. B. Robinson Nuclear Project, Unit 2, Third Ten-Year Inservice Inspection Program Plan*, Revision 0, submitted August 1991 (Reference 3); and

(b) Letter, dated February 13, 1992 (Reference 6), response to the NRC request for additional information dated January 13, 1992.

(c) Letter, dated June 18, 1992 (Reference 7), additional information regarding the inservice inspection program.

2.2 Compliance with Code Requirements

2.2.1 Compliance with Applicable Code Editions

The Inservice Inspection Program Plan shall be based on the Code editions defined in 10 CFR 50.55a(g)(4) and 10 CFR 50.55a(b). Based on the starting date of February 19, 1992, the Code applicable to the third interval ISI program is the 1986 Edition. As stated in Section 1 of this report, the Licensee has prepared the *H. B. Robinson Nuclear Project, Unit 2, Third Ten-Year ISI Program Plan* to meet the requirements of the 1986 Edition of the Code, except that the extent of examination for Class 1, Examination Category B-J welds has been determined by the 1974 Edition through Summer 1975 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 10 CFR 50.55a(b). In the response to the NRC request for additional information, the Licensee committed to perform an augmented volumetric examination on a minimum 7.5% sampling of the piping welds in the containment spray system. With this commitment, 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 exclude 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 examinations specified in Section XI of the ASME Code, the Licensee has committed to perform the following augmented examinations:

(a) Containment spray system (CS) piping welds will receive volumetric examinations on a minimum 7.5% sampling;

(b) The reactor pressure vessel will be examined to the requirements of Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds During Preservice and Inservice Examination*, Revision 1 (Reference 8);

(c) Reactor coolant pump flywheels will be examined in accordance with the HBR2 Technical Specifications. In addition, surface examinations are performed on bores and keyways each time a flywheel is removed from the motors;

(d) Feedwater nozzle welds will be examined on a more frequent basis than is required by Section XI;

(e) 100% of the reactor vessel shell welds (Examination Category B-A, Item B1.10) are scheduled for volumetric examination by the end of the interval;

(f) Accumulator instrumentation nozzle welds will be examined; and

(g) Bolting examinations will be performed per the requirements of IE Bulletin 82-02 (Reference 9).

2.3 Conclusions

Based on the review of the documents listed above, it is concluded that the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval ISI Program Plan*, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

3. EVALUATION OF RELIEF REQUESTS

The requests for relief from the ASME Code Section XI 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 No. 4, Examination Category B-F, Items B5.10 and B5.130, Reactor Vessel Nozzle-to-Safe End Welds and Dissimilar Metal Welds

NOTE: Relief Request No. 4 was withdrawn by the Licensee in the response to the NRC's request for additional information dated February 13, 1992. As a result of a telephone conference on May 19, 1992, Relief Request No. 18 (paragraph 3.1.1.3) was submitted to supersede RR-4.

3.1.1.2 Request for Relief No. 5, Examination Category B-A, Item B1.21, Reactor Vessel Circumferential Head Weld

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-A, Item B1.21 requires a 100% volumetric examination of the accessible length of one circumferential head weld in the successive 2nd, 3rd, and 4th inspection intervals as defined by Figure IWB-2500-3.

Licensee's Code Relief Request: Relief is requested from performing the Code-required volumetric examination of the closure head peel segment-to-disk Weld No. 1.

Licensee's Basis for Requesting Relief: Accessibility for examination of this weld was not provided in the original plant design, which occurred prior to the issuance of Section XI inservice inspection requirements. This weld is considered inaccessible for volumetric examination due to physical space constraints. The closure head peel segment-to-disc weld is completely enclosed within the pattern of control rod drive mechanism (CRDM) penetrations inside the shroud such that no portion of the weld is accessible to either surface or volumetric examination.

Licensee's Proposed Alternative Examination: None. Visual examination for leakage will be performed during leak testing, after each refueling outage, and during the hydrostatic test to be performed near the end of the 120-month interval.

Evaluation: Sketch CPL-101 has been reviewed. Due to the CRD shroud and the CRDM penetrations in the closure head, access to closure head peel segment-to-disk circumferential Weld No. 1 for the Code-required volumetric examination is impractical without significant design changes. The visual examination for evidence of leakage, performed during system pressure tests, will provide reasonable assurance of the continued inservice structural integrity.

Conclusions: It is concluded that the Code-required volumetric examination of the subject closure head peel segment-to-disk weld is impractical to perform. Compliance with this specific requirement of Section XI would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.1.3 <u>Request for Relief No. 18, Examination Category B-F, Items B5.10 and B5.130, Reactor Vessel Nozzle-to-Safe End Welds and Dissimilar Metal Welds</u>

<u>Code Requirement</u>: Section XI, Table IWB-2500-1, Examination Category B-F, Items B5.10 and B5.130 require 100% volumetric and surface examination of each safe end and dissimilar metal weld in each loop of the reactor coolant system as defined by Figure IWB-2500-8.

<u>Licensee's Code Relief Request</u>: Relief is requested from performing 100% of the Code-required surface examination of the primary nozzle safe end and dissimilar metal welds.

<u>Licensee's Basis for Requesting Relief</u>: The sandplug access provided from the floor of the refueling cavity to the outside of the primary nozzle safe ends is insufficient to permit surface examination to be performed on the safe ends and associated dissimilar metal welds.

<u>Licensee's Proposed Alternative Examination</u>: The Licensee will perform an alternative volumetric examination in accordance with IWA-2240 for the full weld volume and heat-affected zone instead of only the inner on-third of the welds. The ultrasonic testing instrumentation and procedure will be demonstrated to be capable of detecting OD surface-connected defects in the circumferential orientation, in a laboratory test block. The demonstration sample defects will be cracks and not machined notches.

<u>Evaluation</u>: As the Licensee has stated, the required surface examinations of the subject welds cannot be performed due to inaccessibility of the welds. Therefore, a volumetric examination from the ID surface was proposed by the Licensee in lieu of the Code-required surface examination. The Licensee has included the following provisions regarding the proposed alternative volumetric examination:

(1) The remote volumetric examination will include the entire weld volume and heat-affected zone instead of only the inner one-third of the weld, as required by the Code.

(2) The ultrasonic testing instrumentation and procedure will be demonstrated to be capable of detecting OD surface-connected defects, in the circumferential orientation, in a laboratory test block. The defects in the test block will be cracks and not machined notches.

The proposed alternative examination will provide adequate assurance that unallowable flaws have not developed in the subject welds, or that they will be detected and repaired prior to continued operation of the system.

Conclusions: It is concluded that the Code-required surface examination of the subject welds is impractical to perform at HBR-2, and that the public health and safety will not be endangered by allowing the proposed alternative examination in lieu of the Code requirement. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted.

3.1.2 Pressurizer

3.1.2.1 Request for Relief No. 1 (revised in letter dated 02/13/92), Examination Category B-D, Item B3.120 Pressurizer Surge Nozzle Inside Radius Section

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.120 requires a 100% volumetric examination of the nozzle inside radius section as defined by Figure IWB-2500-7 during each inspection interval.

Licensee's Code Relief Request: Relief is requested from performing the Code-required volumetric examination of the pressurizer surge nozzle inner radius section.

Licensee's Basis for Requesting Relief: The surge nozzle inner radius is not accessible for volumetric examination due to the heaters connected to the bottom head around the nozzle and, furthermore, is restricted inside by the retaining basket.

Licensee's Proposed Alternative Examination: None. The surge nozzle will receive a VT-2 visual examination during the Code-required system pressure tests.

Evaluation: The Code requires that all pressurizer nozzle inside radius sections receive a 100% volumetric examination each inspection interval. The surge line nozzle inside radius section is inaccessible from the outside diameter due to the heater penetrations around the nozzle and from the inside diameter due to the retaining basket on the inside of the pressurizer. Performing the Code-required examination would require design modifications and/or replacement of the pressurizer. Imposition of the Code requirement on CP&L would cause a burden that would not be compensated by an increase in safety above that provided by the VT-2 visual examination performed during system pressure tests.

Conclusions: The Code-required volumetric examination of the pressurizer surge nozzle inside radius section is impractical to perform at HBR2. Pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.3 Heat Exchangers and Steam Generators

3.1.3.1 Request for Relief No. 2 (part 1 of 2), Examination Categories B-B and B-D, Items B2.51, B2.60, B3.150, and B3.160, Welds and Nozzle Inside Radii in Regenerative Heat Exchangers

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-B, Item B2.51 requires a 100% volumetric examination of one circumferential head weld as defined by Figures IWB-2500-1 and -3. Item B2.60 requires a 100% volumetric examination of the tubesheet-to-head welds as defined by Figure IWB-2500-6.

Examination Category B-D, Items B3.150 and B3.160 require a 100% volumetric examination of the nozzle-to-vessel welds and the nozzle inside radius sections as defined by Figure IWB-2500-7.

Licensee's Code Relief Request: Relief is requested from performing the Code-required volumetric examinations of the regenerative heat exchanger circumferential head welds, tubesheet-to-head weld, nozzle-to-vessel welds, and the nozzle inside radius sections.

Licensee's Basis for Requesting Relief: Based on radiation exposures in the area, i.e., general area of 3 to 4 R/hr and 9 to 10 R/hr on contact, with hot spots of 12 to 15 R/hr, and in view of the fact that previous examinations revealed no indications, relief is requested from the volumetric examinations of the regenerative heat exchanger shell and nozzles.

Licensee's Proposed Alternative Examination: None. A VT-2 visual examination will be performed during the hydrostatic test at the end of the interval.

Evaluation: The Code-required volumetric examination of the subject regenerative heat exchanger welds and inner radius sections would result in personnel receiving excessive radiation exposure. Based on the ALARA concerns surrounding the performance of these examinations, the Code requirement is impractical to perform. The VT-2 visual examination for evidence of leakage, performed during the system hydrostatic test will provide reasonable assurance of the continued inservice structural integrity of the regenerative heat exchanger welds and inner radii.

Conclusions: Imposition of this Code requirement on HBR2 would cause an undue burden without a compensating increase in safety.

Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.3.2 Request for Relief No. 11, Examination Category B-D, Item B3.150, Regenerative Heat Exchanger Nozzle-to-Vessel Welds

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 11 due to a conflict with Relief Request No. 2 (paragraph 3.1.3.1).

3.1.3.3 Request for Relief No. 12, Examination Category B-D, Item B3.140, Steam Generator Nozzle Inner Radius Sections

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-D, Item B3.140 requires a 100% volumetric examination of the nozzle inner radius section as defined by Figure IWB-2500-7.

Licensee's Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination of the steam generator nozzle inner radii.

Licensee's Basis for Requesting Relief: The nozzles in the H. B. Robinson Unit 2 steam generators are integrally cast with the vessel heads. The inner radius area is covered by weld-deposited stainless steel cladding that is in an as-welded condition. In addition, radiation levels inside the primary channel head are in the range of 10 R/hr. In view of the cast nozzle design, rough clad surfaces, and radiation levels, volumetric examinations in this area will not be attempted.

Licensee's Proposed Alternative Examination: All steam generator nozzle inner radii will be visually examined once during the examination interval.

Evaluation: The steam generator nozzle sections at HBR2 were not designed for external examination of the inside radius using ultrasonic methods. The component geometry and the as-cast surface of the steam generator heads, along with the excessively long test metal distance that results in high ultrasonic attenuation, preclude the volumetric examination of the nozzle inside radius section from the external surface. The steam generator nozzle design, therefore, makes the Code-required examination impractical to perform. In order to examine the nozzle inside radius sections in accordance with the requirements, the steam generator nozzles, and thus the steam generators, would have to be redesigned, fabricated, and installed. The increase in plant safety would not compensate for the burden placed on the Licensee that would result from imposition of the requirement. Surface examination is not practical to perform because of the rough surface of the as-welded cladding and because inspection personnel would receive excessive radiation exposure.

CP&L's proposed alternative is to perform a visual examination of the inside surface of each steam generator primary side nozzle inner radius section once during the interval. The Licensee should consider color capabilities for any remote visual equipment being used. There have been instances where rust in cladding cracks was not detected by ordinary black and white monitors.

Conclusions: The volumetric examination required by Section XI of the ASME Code for the nozzle inside radius sections in the steam generators is impractical to perform at HBR2. It is concluded that the public health and safety will not be endangered by allowing the alternative examination to be performed in lieu of the Code requirement. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.4 Piping Pressure Boundary

3.1.4.1 Request for Relief No. 6, Examination Category B-J, Items B9.10 and B9.31, Volumetric in Lieu of Surface Examination

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 6.

3.1.4.2 Request for Relief No. 7, Examination Category B-J, Items B9.11 and B9.12, RCS Crossover Leg Elbows

In a letter dated June 18, 1992, the Licensee withdrew Relief Request No. 7 as a result of the May 19, 1992 conference call.

3.1.4.3 Request for Relief No. 8, Examination Category B-J, Item B9.11, RPV Cold Leg Circumferential Butt Weld

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-J, Item B9.11 requires a volumetric and surface examination of the circumferential welds as defined by Figure IWB-2500-8.

Licensee's Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric and surface examinations of the pressure-retaining circumferential butt welds attaching the pipe to the 15° elbow in each reactor coolant cold leg. These welds are shown on sketches CPL-107, 107A, and 107B as Weld 13 for each cold leg.

Licensee's Basis for Requesting Relief: The circumferential butt weld attaching the pipe to the 15° elbow in each cold leg of the reactor coolant system is completely enclosed within the biological shield and is not accessible for examination by either volumetric or surface techniques.

Licensee's Proposed Alternative Examination: None.

Evaluation: Review of the referenced sketches in the Licensee's submittal shows the 15° elbow adjacent to the RPV nozzle on the cold leg side. The pipe-to-elbow weld is not accessible from the inside diameter during the automated reactor vessel inspection. The pipe-to-elbow weld and the nozzle safe end weld do not share the same axis, therefore, the remote inspection tool could not be used to scan this weld. Access to the outside diameter is restricted by the biological shield.

The design of the reactor pressure vessel and biological shield makes the Code-required volumetric and surface examinations impractical to perform at HBR2. The Code-required system pressure tests will provide reasonable assurance of the continued inservice structural integrity of the reactor coolant system.

Conclusions: Based on the above evaluation, it is concluded that complying with this Code requirement is impractical in this circumstance. Imposition of the Code-required volumetric and surface examinations would necessitate redesign or replacement of the piping at the 15° elbow and create a burden on the Licensee that would not be compensated with an increase in safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.1.4.4 Request for Relief No. 9, Examination Category B-K-1, Item B10.10, Piping Integrally Welded Attachments

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 9.

3.1.5 Pump Pressure Boundary

3.1.5.1 Request for Relief No. 10, Examination Category B-K-1, Item B10.20, Integral Attachments for Pumps

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 10.

3.1.5.2 Request for Relief No. 13, Examination Category B-L-1 and B-L-2, Items B12.10 and B12.20, Pump Casing Welds and Internal Surfaces

Code Requirement: Section XI, Table IWB-2500-1, Examination Category B-L-1, Item B12.10 requires a 100% volumetric examination of the Class 1 pump casing welds as defined by Figure IWB-2500-16. Examination Category B-L-2, Item B12.20 requires a VT-3 visual examination of the Class 1 pump casing internal surfaces. These examinations can be deferred until the end of the interval.

Licensee's Code Relief Request: Relief is requested from performing 100% of the Code-required volumetric examination of the reactor coolant pump casing welds and from the VT-3 visual examination of the pump casing internal surfaces unless the pump is disassembled for maintenance during the interval.

Licensee's Basis for Requesting Relief: The Licensee states that visual and volumetric examinations were performed on Loop B Pump during the 1982 refueling outage using visual and radiographic techniques. These examinations revealed no indications. The 41 man-rem radiation exposure associated with this examination far exceeded the normally expected exposures for an ISI outage program.

In addition, the casings consist of four cast stainless steel rings joined by three circumferential welds. The pump internals are removed and transported to the reactor vessel cavity for storage during radiographic and visual examinations and then brought back to the pump. This creates the possibility for significant damage during disassembly and/or transport of the internals.

Carolina Power and Light Company feels that the increased radiation exposures and the excessive costs associated with performance of these examinations far exceeds their possible benefits. CP&L feels this is particularly true since the 1982 examinations revealed no reportable indications and the proposed alternatives would satisfy any safety concerns.

Licensee's Proposed Alternative Examination: As an alternative to the volumetric examination of the casing welds, the exterior of the pump casing will be visually examined during the hydrostatic pressure test required by IWB-5000. An outside surface examination will be performed on the weld to the extent practicable and as access permits.

If maintenance or operational problems are encountered that require disassembly of the pump, the pump interior surface will be VT-3 visually examined. The need for performance of a volumetric examination will be re-evaluated at that time.

Evaluation: The visual examination requirements for internal surfaces of pumps necessitate complete disassembly of the pump. Disassembly of the reactor coolant pumps for the sole purpose of visual examination of the casing internal surfaces and volumetric examination of the pump casing weld is a major effort and requires many manhours from skilled maintenance and inspection personnel. In addition to the possibility of damage to the pump, personnel would receive excessive radiation exposure. Therefore, the Code requirement is impractical.

The VT-3 visual examination is performed to determine if unanticipated degradation of the casing is occurring due to phenomena such as erosion, corrosion, or cracking. However, previous experience during examination of similar pumps at other plants has not shown any significant degradation of pump casings. Imposition of the requirements on CP&L would cause a burden that would not be compensated significantly by an increase in safety above that provided by the proposed examination.

Later editions and addenda of the ASME Code (1988 Addenda and later) have eliminated disassembly of pumps for the sole purpose of performing examinations of the internal surfaces and state that the internal surface visual examination requirement is only applicable to pumps that are disassembled for reasons such as maintenance, repair, or volumetric examination. Since no major problems have been reported in the industry with regard to pump casings, the Licensee's proposal will provide adequate assurance of the continued inservice structural integrity.

CP&L's proposed alternative is to perform a surface examination of the selected pump casing weld outside surface to the extent practical and as access permits. However, the Licensee should perform the Code-required VT-3 visual examination of the internal surfaces of one pump and perform volumetric examination of the pump casing weld if the internal surfaces are made accessible due to disassembly for maintenance or repair.

<u>Conclusions</u>: It is concluded that the disassembly of a pump for the sole purpose of performing the inspections required by Section XI of the ASME Code is impractical. Public health and safety will not be endangered by allowing the proposed examination to be performed in lieu of the Code requirement if one of the pumps is not disassembled for maintenance or repair. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted provided that (1) the Code-required volumetric and VT-3 visual examinations be performed if a pump is disassembled for repair or maintenance, (2) the Licensee's proposed VT-2 visual and surface examinations are performed if one of the pumps is not disassembled, and (3) if a pump has not

been disassembled, this fact should be reported by the Licensee in the ISI Summary Report at the end of the interval.

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

3.2.1.1 Request for Relief No. 2 (part 2 of 2), Examination Category C-A, Items C1.10 and C1.30, Regenerative Heat Exchanger Pressure Retaining Welds

Code Requirement: Section XI, Table IWC-2500-1, Examination Category C-A, Item C1.10 requires 100% volumetric examination of the shell circumferential welds at gross structural discontinuities as defined by Figure IWC-2500-1. Item C1.30 requires 100% volumetric examination of tubesheet-to-shell welds as defined by Figure IWC-2500-2.

Licensee's Code Relief Request: Relief is requested from performing the Code-required volumetric examination of the regenerative heat exchanger shell circumferential welds and tubesheet-to-shell welds.

Licensee's Basis for Requesting Relief: Based on radiation exposures in the area, i.e., general area of 3 to 4 R/hr and 9 to 10 R/hr on contact, with hot spots of 12 to 15 R/hr, and in view of the fact that previous examinations revealed no indications, relief is requested from volumetric and surface examinations of the regenerative heat exchanger circumferential shell and tubesheet-to-shell welds.

Licensee's Proposed Alternative Examination: None. VT-2 visual examination during the hydrostatic test at the end of the interval.

Evaluation: The Code-required volumetric examination of the subject regenerative heat exchanger welds would result in personnel receiving excessive radiation exposure. Based on the ALARA, the Code requirement is impractical. The VT-2 visual examination for evidence of leakage, performed during the system hydrostatic test, will provide reasonable assurance of the continued inservice structural integrity of the regenerative heat exchanger welds.

Conclusions: Imposition of this Code requirement on HBR2 would cause an undue burden without a compensating increase in safety. Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), it is recommended that relief be granted as requested.

3.2.1.2 Request for Relief No. 14, Examination Category C-A, Item C1.30, Regenerative Heat Exchanger Tubesheet-To-Shell Weld

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 14 due to a conflict with Relief Request No. 2 (paragraph 3.1.3.1).

3.2.2 Piping

3.2.2.1 Request for Relief No. 15, Examination Category C-F-1, Item C5.11 and C5.21, Volumetric Examination of Class 2 Piping in Lieu of Surface Examination

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 15.

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 No. 3, Paragraph IWA-2232, Materials for Fabrication of Calibration Blocks

Code Requirement: Section XI, paragraph IWA-2232 requires that the material from which the basic calibration blocks are fabricated be:

(a) a nozzle dropout from the component;
(b) a component prolongation;
(c) material of the same material specification, product form and heat treatment as one of the materials being joined;
(d) clad by the same method as was used on the component (i.e., automatic, manual, etc.).

Licensee's Code Relief Request: Relief is requested to use SA-533 Grade B material in lieu of SA-302 Grade B and SA-508 material in lieu of SA-336 for the reactor vessel calibration blocks and SA-533 Grade B in lieu of SA-302 Grade B for the pressurizer calibration blocks. Also, the existing manually-clad calibration blocks will be used for reactor vessel examinations in lieu of required automatic-clad blocks (method used on the reactor vessel).

Licensee's Basis for Requesting Relief: The required materials, SA-302 Grade B and SA-336, are not available. Based on chemical and physical properties, SA-533 Grade B and SA-302 Grade B are considered essentially equivalent. This parity is also evident in the properties of SA-336 and SA-508 material. These materials are considered to be acoustically equivalent, thereby meeting the intent of the Code.

The use of existing manually-clad vessel calibration blocks would facilitate comparison of data for the third ISI interval with data obtained from previous examinations.

Licensee's Proposed Alternative Examination: Not applicable.

Evaluation: All of the subject calibration blocks have been used for previous intervals, therefore, their continued use would provide consistent results. At this time the procurement of calibration blocks of the exact materials would be difficult, if not impossible, therefore, the Code requirement is impractical. Because the chemical and physical properties of the subject materials are equivalent, the increase in plant safety would not compensate for the burden placed on the Licensee that would result from requiring the fabrication of new calibration blocks to meet the current Code. The Licensee has demonstrated that the use of the alternative calibration block material provides an acceptable level of quality and safety and that compliance with the specific Code requirement would result in hardship or unusual difficulties without a compensating increase in the level of quality and safety.

Conclusions: Based on the above, it is concluded that the Code requirement is impractical for HBR2 and that public health and safety will not be endangered by allowing the continued use of the alternative calibration blocks. Therefore, pursuant to

10 CFR 50.55a(a)(6)(i), it is recommended that relief be granted as requested.

3.5.3.2 Request for Relief No. 16, Wall Thickness Variation Between Calibration Block and Reactor Coolant Piping

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 16 based on implementation of Code Case N-461, "Alternative Rules for Piping Calibration Block Thickness."

3.5.3.3 Request for Relief No. 17, Use of Code Case N-461

In the February 13, 1992 response to the NRC's request for additional information, the Licensee withdrew Relief Request No. 17 based on the NRC approval of Code Case N-461 in Regulatory Guide 1.147 (Reference 10).

4. CONCLUSION

Pursuant to 10 CFR 50.55a(g)(6), it has been determined that certain inservice examinations cannot be performed to the extent required by Section XI of the ASME Code. As a result of the NRC's request for additional information and multiple conference calls, Requests for Relief Nos. 4, 6, 7, 9, 10, 11, 14, 15, 16, and 17 were withdrawn, Request for Relief No. 18 was submitted, and Request for Relief No. 1 was revised by the Licensee. In all the remaining cases for which relief is requested, the Licensee has demonstrated that specific Section XI requirements are impractical.

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 H. B. Robinson Nuclear Project, Unit 2, facility. Compliance with all the exact Section XI required inspections would necessitate redesign of a significant number of plant systems, sufficient replacement components to be obtained, installation of the new components, and a baseline examination of 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. Pursuant to 10 CFR 50.55a(g)(6), relief is allowed from the requirements that are impractical to implement. Relief may be granted only if the 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.

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 *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, the Licensee's response to the NRC's request for additional information, and the recommendations for granting relief from the ISI examination requirements that have been determined to be impractical, it is concluded that the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, is acceptable and in compliance with 10 CFR 50.55a(g)(4).

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:

   1974 Edition through Summer 1975 Addenda
   1986 Edition

3. *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection Program Plan*, Revision 0, submitted August 1991.

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

5. Letter, dated January 13, 1992, D. H. Dorman (NRC) to G. E. Vaughn (CP&L), containing request for additional information on the Third 10-Year Interval ISI Program Plan.

6. Letter, dated February 13, 1992, G. E. Vaughn (CP&L) to Document Control Desk (NRC), containing the response to NRC request for additional information.

7. Letter, dated June 18, 1992, R. B. Starkey, Jr. (CP&L) to Document Control Desk (NRC), containing additional information regarding the ISI Program.

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

9. IE Bulletin 82-02, *Degradation of Threaded Fasteners in the Reactor Coolant Pressure Boundary of PWR Plants*, June 2, 1982.

10. NRC Regulatory Guide 1.147, *Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1*, Revision 8, November 1990.

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11. ABSTRACT (200 words or less)

This report presents the results of the evaluation of the *H. B. Robinson Nuclear Project, Unit 2, Third 10-Year Interval Inservice Inspection (ISI) Program Plan*, Revision 0, submitted August 1, 1991, 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 *H. B. Robinson Nuclear Project, Unit 2, 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 reviews by the Nuclear Regulatory Commission (NRC). The requests for relief are evaluated in Section 3 of this report.

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