January 28, 2004

Mr. John L. Skolds, President
Exelon Nuclear
Exelon Generation Company, LLC
4300 Winfield Road
Warrenville, IL 60555

SUBJECT: QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2 - FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION RELIEF REQUESTS NOS. I4R-01 THROUGH I4R-09 (TAC NOS. MB7695 THROUGH MB7712)

Dear Mr. Skolds:

By letter dated January 17, 2003, and as supplemented by letters dated February 7, August 13, and October 10, 2003, Exelon Generation Company, LLC (the licensee) requested relief from various American Society of Mechanical Engineers Section XI Code requirements for Quad Cities Nuclear Power Station, Units 1 and 2. These relief requests were for the fourth 10-year inservice inspection (ISI) interval which commenced on March 10, 2003, and is scheduled to be completed by March 9, 2013.

Based on the information provided in your submittals for Relief Request I4R-01, the staff concludes that complying with the specified Code requirement, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed alternative provides reasonable assurance of continued structural integrity. On this basis, for I4R-01, the proposed alternative is authorized pursuant to Section 50.55a(a)(3)(ii) of Title 10 of the *Code of Federal Regulations* (10 CFR), for the fourth 10-year ISI interval.

Based on the information provided in your submittals for Relief Requests I4R-02, I4R-03, I4R-04, I4R-06, I4R-07, and I4R-09, the staff concludes that the proposed alternatives provide an acceptable level of quality and safety. Therefore, the proposed alternatives under Relief Requests I4R-02, I4R-03, I4R-04, I4R-06, I4R-07, and I4R-09, are authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval.

Based on the information provided in your submittals for Relief Request I4R-05, the staff concludes that compliance with the specified Code requirements is impractical. The proposed alternative provides reasonable assurance of continued structural integrity. Therefore, Relief Request I4R-05 is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the fourth 10-year ISI interval. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and 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.

Based on the information provided in your submittals for Relief Request I4R-08, the staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, the proposed alternative under Relief Request I4R-08 is authorized pursuant to 10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval or until such time as Code Case

N-513-1 is referenced in 10 CFR 50.55a. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-513-1 with limitations or conditions specified in 10 CFR 50.55a, if any.

Our safety evaluation is enclosed.

Sincerely,

/RA by DPickett for/

Anthony J. Mendiola, Chief, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos.: 50-254 and 50-265

Enclosure: Safety Evaluation

cc w/encl: See next page

10 CFR 50.55a(a)(3)(i) for the fourth 10-year ISI interval or until such time as Code Case N-513-1 is referenced in 10 CFR 50.55a. At that time, if the licensee intends to continue implementing this code case, it must follow all provisions of Code Case N-513-1 with limitations or conditions specified in 10 CFR 50.55a, if any.

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cc w/encl: See next page

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

FOURTH 10-YEAR INTERVAL INSERVICE INSPECTION RELIEF REQUESTS

EXELON GENERATION COMPANY, LLC

QUAD CITIES NUCLEAR POWER STATION, UNITS 1 AND 2

DOCKET NOS. 50-254 AND 50-265

1.0 INTRODUCTION

The Inservice Inspection (ISI) of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) Class 1, Class 2, and Class 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by 10 CFR 50.55a(g), except where specific relief has been granted by the Commission pursuant to 10 CFR 50.55a(g)(6)(i). 10 CFR 50.55a(a)(3) states in part, that alternatives to the requirements of paragraph (g) may be used, when authorized by the Nuclear Requlatory Commission (NRC), if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) will meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, “Rules for ISI of Nuclear Power Plant Components,” to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The ISI code of record for Quad Cities Nuclear Power Station, Units 1 and 2, fourth 10-year ISI interval is the 1995 Edition of the ASME Boiler and Pressure Vessel Code through the 1996 Addenda.

By letter dated January 17, 2003, and as supplemented by letters dated February 7, August 13, and October 10, 2003, Exelon Generation Company, LLC, (the licensee) requested relief from various ASME Section XI Code requirements pertaining to the fourth 10-year ISI interval. Each relief request item will be described and addressed by section number throughout this safety evaluation.

2.0 REGULATORY EVALUATION

2.1 Relief Request No. IR4-01, Relief from Table IWC-2500-1, Examination Category B-D, Item B3,100, Standby Liquid Control Nozzle Inner Radius

Code Requirements for which Relief is Requested

The licensee stated that the 1995 Edition through the 1996 Addenda of ASME Section XI, IWB-2500, requires that components shall be examined and tested as specified in Table IWB-2500-1. Table IWB-2500-1 requires a volumetric examination be performed on the inner radius section of all reactor pressure vessel (RPV) nozzles each interval.

Licensee's Proposed Alternative to Code

The licensee proposed the performance of a visual inspection, VT-2 of the subject nozzles each refueling outage in conjunction with a Class 1 System Leakage Test and monitoring the reactor coolant system for leakage every 12 hours during operation.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(ii), relief is requested on the basis that compliance with the specified Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The standby liquid control (SBLC) nozzle is designed with an integral socket to which the boron injection piping is fillet welded. The nozzle is located in the upper segment of the lower head of the RPV in an area that is inaccessible for examinations from the inside of the reactor vessel which would require that the examination be performed from the outside surface of the lower RPV head. The licensee stated that due to the small diameter (2 inches) of the SBLC nozzle and the thickness of the lower RPV head (6.125 inches), the ratio of the nozzle diameter to the head thickness makes it difficult to perform a meaningful examination from the nozzle to head outer blend radius. The inside surface inspection angle, utilizing specialized contoured wedges or shoes, needs to be mid-range tangential angle to adequately detect flaws at the inside radius area of interest while scanning from the outer blend. Given the small diameter nozzle, the angle of incidence of the ultrasonic beam will be 0 degrees to 15 degrees which equates to nearly a longitudinal examination. This low angle will not provide sufficient reflectivity for detection of inservice induced defects.

The licensee went on to state that an examination from the head surface near the nozzle is difficult because the forged nozzle construction has stainless steel cladding welded to the inside surface with increased weldment applied at the inner radius area which makes it an engineered socket configuration. After final machining, this engineered socket receives an austenitic internal pipe, which is integrally welded to form a complex cladding/socket configuration. The complex geometric configuration and change in grain structures at the dissimilar material interface prohibit completing the examination in a timely fashion. The change in grain structures result in mode conversion and angle changes, consequently significantly longer exam duration would be required to resolve the multiple signals. The licensee estimated that the examination of this unique design would result in four man-hours in a high dose field. The total dose is estimated to exceed one Rem based on a radiation field in that area of 400 mrem/hour.

The licensee stated that a review of current ultrasonic techniques was conducted including discussions with the Electric Power Research Institute. The conclusion was that the long metal path and potential for multiple geometric and dissimilar material prevent a meaningful examination from being performed on the inner radius of the SBLC nozzle. At present, the industry has a qualified technique for dissimilar metal piping welds, but has not addressed this unique design.

The licensee indicated that the inner radius socket attaches to piping that delivers the boron solution far away from the nozzle inner radius. Therefore, the SBLC nozzle inner radius section is not subjected to turbulent mixing conditions that are a concern with other penetrations. In addition, a VT-2 examination of the SBLC nozzle is a part of the Class 1 system leakage test scheduled during each refueling outage. Finally, during reactor operation, technical specifications require that reactor coolant system leakage is monitored every 12 hours and specify that prudent actions including shutdown be taken should predetermined limits be exceeded.

Staff Evaluation

By letter dated January 17, 2003, the licensee submitted request for relief I4R-01 pursuant to 10 CFR 50.55a(g)(5)(iii) on the basis that conformance with the Code requirements is impractical. The licensee sought relief from the requirements of ASME Code, Section XI to perform volumetric examination on the inner radius of the SBLC nozzle. IWB-2500 of ASME Section XI requires that a volumetric examination be performed on the inner radius of all reactor pressure vessel nozzles each inspection interval. As an alternative examination, the licensee proposed to perform a VT-2 visual examination of the inner radius of the SBLC nozzle each refueling outage in conjunction with the Class 1 System Leakage Test plus monitoring the containment for leakage per its technical specification requirements. In its supplemental letter dated October 10, 2003, the licensee submitted a revised version of I4R-01 pursuant to 10 CFR 50.55a(a)(3)(ii) on the basis that conformance with the Code requirements would result in a hardship without a compensating increase in the level of quality and safety. The revised relief request provided a more detailed description of the construction, geometry, and radiation field to substantiate its earlier stated difficulties involved with performing a “meaningful examination.”

The staff concurs that the performance of the ultrasonic testing (UT) inspection of the nozzle inner radius would be time consuming and difficult to interpret. The intent of the examination is for the transmitted ultrasonic beam to strike perpendicular to any defects that may be emanating from the radius, forming a corner trap for the signal, which causes a strong, reflective response for interpretation. The configuration for this engineering nozzle with the small inner radius where the transducers would be placed would result in a shear wave examination that approaches a zero degree longitudinal examination. This would result in the ultrasonic beam striking parallel to any defects at the area of interest with no corner trap for reflection from the defect, which would result in a meaningless examination because any defects present would not be detectable. In addition, the staff concludes that the licensee has sufficiently investigated any new technology that may exist to aid in the performance of the examination by conducting discussions with the Electric Power Research Institute (EPRI).

The licensee's explanation with respect to extended inspection times in a high dose area provide a reasonable basis for hardship. The UT operator will have to discriminate from many signals caused by the geometry and the differences in metallurgical structure due to the cladding. Secondly, any UT responses that would be provided would be of marginal value in finding any defects as discussed previously which would result in a minimal increase in safety if at all. Finally, since the inner radius socket attaches to piping which injects boron at locations far removed from the nozzle and thus, the SBLC nozzle inner radius is not subjected to turbulent mixing conditions which would degrade the nozzle, the staff concludes that the licensee's alternative to perform periodic leakage tests and monitoring for leakage per the technical specifications provide reasonable assurance of the continued structural integrity of the nozzle.

Conclusion

The staff concludes that for Relief Request I4R-01, imposition of the Code requirements on the licensee would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety, and that the licensee's proposed alternative provides reasonable assurance of structural integrity of the RPV SBLC nozzle. Therefore, the licensee's proposed alternatives are authorized pursuant to 10 CFR 50.55a(a)(3)(ii) for Quad Cities Nuclear Power Station, Units 1 and 2 for the fourth 10-year ISI interval. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.2 RELIEF REQUEST NO. I4R-02, ALTERNATE RISK INFORMED SELECTION AND EXAMINATION CRITERIA FOR CODE CATEGORY B-F, B-J, C-F-1, AND C-F-2 PRESSURE RETAINING WELDS

Code Requirements for which Relief is Requested

The licensee stated that Table IWB-2500-1, Examination Category B-F, requires volumetric and/or surface examinations on all welds for Items B5.10 and B5.20. Examination Category B-J, requires volumetric and/or surface examinations on a sample of welds for Items B9.11, B9.21, B9.31, B9.32, and B9.40. The weld population selected for inspection includes the following:

1. All terminal ends in each pipe or branch run connected to vessels.

2. All terminal ends and joints in each pipe or branch run connected to other components where the stress levels exceed either of the following limits under loads associated with specific seismic events and operational conditions:

    a. primary plus secondary stress intensity range of $2.4S_m$ for ferritic steel and austenitic steel.

    b. cumulative usage factor U of 0.4.

3. All dissimilar metal welds not covered under Category B-F.

4. Additional piping welds so that the total number of circumferential butt welds, branch connections, or socket welds selected for examination equals 25 per cent of the circumferential butt welds, branch connection, or socket welds in the reactor coolant piping system. This total does not include welds excluded by IWB-1220.

The licensee stated IWC-2500-1, Examination Categories C-F-1 and C-F-2 require volumetric and/or surface examinations on a sample of welds for Items C5.11, C5.51, C5.70, and C5.81. The weld population selected for inspection includes the following:

1. Welds selected for examination shall include 7.5 percent, but not less then 28 welds, of all dissimilar metal, austenitic stainless steel and high alloy welds (Category C-F-1) or of all carbon and low alloy steel welds, (Category C-F-2) not exempted by IWC-1220. (Some welds not exempted by IWC-1220 are not required to be nondestructively examined per Examination Categories C-F-1 and C-F-2. These welds, however, shall be included in the total weld count to which the 7.5 percent sampling rate is applied). The examinations shall be distributed as follows:

    a. the examinations shall be distributed among the Class 2 systems prorated, to the degree practicable, on the number of nonexempt dissimilar metal, austenitic stainless steel and low alloy welds (Category C-F-1) or carbon and low alloy welds (Category C-F-2) in each system;

    b. within a system, the examinations shall be distributed among terminal ends, dissimilar metal welds, and structural discontinuities prorated, to the degree practicable, on the number of nonexempt terminal ends, dissimilar metal welds, and structural discontinuities in the system; and

    c. with each system, examinations shall be distributed between line sizes prorated to the degree practicable.

<u>Licensee's Proposed Alternative to Code</u>

The licensee plans to use the Risk Informed ISI methodology approved for use by the staff for the Quad Cities Nuclear Power Station Units 1 and 2 third 10-year ISI interval as the alternative for the fourth 10-year ISI interval. The alternative will continue to use the same two enhancements proposed and approved under the third 10-year interval.

The licensee stated that in lieu of the evaluation and sample expansion requirements in Section 3.6.6.2, "RI-ISI Selected Examinations" of EPRI Topical Report TR-112657, the requirements of Subarticle-2430, "Additional Examinations contained in Code Case N-578-1" will be used as the first enhancement.

The second enhancement proposed by the licensee is to use Table 1, Examination Category R-A, “Risk-Informed Piping Examinations” contained in Code Case N-578-1 as an alternative to the requirements listed in Table 4-1, “Summary of Degradation-Specific Inspection Requirements and Examination Methods” of EPRI TR-112657.

Licensee’s Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee stated that the use of the EPRI Topical Report (TR) 112657 Rev. B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure,” with two enhancements, which was approved by the staff February 5, 2002, will be used as the methodology for the fourth ISI interval. The initial Risk Informed ISI (RI-ISI) Program was submitted during the third period of the third interval for both units 1 and 2. This initial RI-ISI Program was developed in accordance with TR-112657, Rev. B-A, as supplemented by Code Case N-578-1.

The licensee stated the transition from the 1989 Edition of the ASME Code to the 1995 Edition with the 1996 Addenda of ASME Section XI for the fourth ISI interval does not impact the currently approved RI-ISI evaluation process used in the third ISI interval, and the requirements of the new Code Edition/Addenda will be implemented as detailed in the ISI Program Plan.

The Risk Impact Assessment completed as part of the original baseline RI-ISI Program was an implementation/transition check on the initial impact of converting from a traditional ASME Section XI program to the new RI-ISI methodology. For the fourth interval ISI update, there is no transition occurring between two different methodologies, but rather, the currently approved RI-ISI methodology and evaluation will be maintained for the new interval. As such, the licensee stated the initial screening of the risk impact assessment is not a part of the living program process and is not required to be continually updated.

Staff Evaluation

By letter dated January 17, 2003, the licensee submitted a request for relief pursuant to 10 CFR 50.55a(a)(3(i). The licensee sought relief from the requirements of ASME Code, Section XI to utilize a risk-informed ISI program plan at Quad Cities Nuclear Power Station to perform ISI during the fourth interval. Additional information was provided by the licensee by letter dated August 13, 2003. A risk-informed ISI program was reviewed and approved by the NRC for use during the third interval at Quad Cities Nuclear Power Station. In the January 17, 2003, letter, the license stated that the ranking and evaluation procedure of the RI-ISI program for the fourth interval remain unchanged. In the August 13, 2003, letter, the licensee clarified that the current welds selected are the same as those selected during the third interval. Furthermore, in its clarification, the licensee indicated that the timeline for expansion of the sample due to service related issues will be in the same outage as the issue was identified, which is acceptable to the staff.

The licensee is requesting relief to use the proposed RI-ISI program plan instead of the ASME Section XI program for piping in the fourth interval. An acceptable RI-ISI program plan is expected to meet the five key principles discussed in Regulatory Guide 1.178, Standard Review Plan 3.9.8, and the EPRI Topical report. The first principle is met in this relief request because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(3)(i) and all regulations are met. Therefore, an exemption request is not required. The second and third principles require assurance that the alternative program is consistent with the defense in depth philosophy and that sufficient safety margins are maintained, respectively. Assurance that the second and third principles are met is based on the application of the approved methodology and not on the particular inspection locations selected. The methodology used to develop the fourth interval RI-ISI program is unchanged from the methodology approved for use in third interval and therefore, the second and third principles are met.

The fourth principle that any increase in core damage frequency and risk are small and consistent with the NRC Safety Goal Policy Statement requires an estimate of the change in risk, and the change in risk estimate is dependent on the location of inspections in the proposed ISI program compared to location of inspections that would be inspected using the requirements of ASME Section XI from which relief was granted. The licensee has stated that the current fourth interval locations have not been changed from the third interval locations and so any increase in risk, barring changes to the probabilistic risk assessment (PRA), would remain as previously estimated and therefore, the fourth principle is met. As discussed in the staff's safety evaluation for the third ISI interval (letter from A. Mendiola to O. Kingsley dated February 5, 2002), the licensee stated that the RI-ISI program will be maintained and reviewed on an ASME-period basis. In the August 13, 2003, letter, the licensee clarified that the PRA was updated in late 2002, and a project is underway to compare the current RI-ISI selection against the updated PRA. Maintenance of a living program satisfies the fifth key principle which provides that risk-informed applications should include performance monitoring and feedback provisions.

The staff finds that the five key principles of risk-informed decision-making are ensured by the licensee proposed fourth interval RI-ISI program plan and therefore, the proposed program for the fourth interval is acceptable.

Conclusion

Based on the discussion above, the staff concludes that the proposed alternative for a risk informed ISI program provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-02, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.3 RELIEF REQUEST NO. I4R-03, ALTERNATIVE REQUIREMENTS TO ASME SECTION XI, APPENDIX VII, SUBSUBARTICLE VII-4240, ANNUAL TRAINING

Code Requirements for which Relief is Requested

The licensee stated that the 1995 Edition of ASME Section XI with the 1996 Addenda, Appendix VII, requires a minimum of 10 hours of annual training. In addition to the above, the licensee stated 10 CFR 50.55a(b)(2)(xiv), "Appendix VIII personnel qualification," requires that all personnel qualified to perform UT examinations in accordance with ASME Section XI, Appendix VIII, shall receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing UT examinations at a licensee's facility.

Licensee's Proposed Alternative to Code

The licensee stated that annual UT training shall be conducted in accordance with 10 CFR 50.55a(b)(2)(xiv) in lieu of Subsubarticle VII-4240 of ASME Section XI, 1995 Edition with the 1996 Addenda, Appendix VII. The annual UT training shall require that all personnel qualified for performing UT examination in accordance with ASME Section XI, Appendix VIII, shall receive 8 hours of annual hands-on training on specimens that contain cracks. This training must be completed no earlier than 6 months prior to performing UT examinations at a licensee's facility.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee stated that the proposed alternative will provide an acceptable level of quality and safety. The licensee stated that on September 22, 1999, the staff published a final rule in the *Federal Register* to amend 10 CFR 50.55a(b)(2) to incorporate by reference the 1995 Edition and Addenda through the 1996 Addenda of ASME Section XI. The change included the requirement to have a minimum of 10 hours of annual training contained in subsubarticle VII-4240 of ASME Section XI. Additionally, the amended 10 CFR 50.55a(b)(xiv) requires that all personnel qualified to perform UT examinations in accordance with Appendix VIII of ASME Section XI shall receive 8 hours of annual hands-on training on specimens that contain cracks. Paragraph 2.4.1.1.1 in the *Federal Register* notice contained the following statement which includes a discussion of the EPRI Performance Demonstration Initiative (PDI) program:

"The NRC had determined that this requirement (i.e. Subsubarticle VII-4240) was inadequate for two reasons. The first reason was that the training does not require laboratory work and examination of flawed specimens. Signals can be difficult to interpret and, as detailed in the regulatory analysis for this rulemaking, experience and studies indicate that the examiner must practice on a frequent basis to maintain the capability for proper interpretation. The second reason is related to the length of training and its frequency. Studies have shown that an examiner's capability begins to diminish within approximately 6 months if skills are not maintained. Thus, the NRC had determined that 10 hours of annual training is not sufficient practice to maintain skills, and that an examiner must practice on a more frequent basis to maintain proper skill level. The PDI program has adopted a requirement for 8 hours of training, but it is required to be hands-on practice. In addition, the training must be taken no earlier than 6 months prior to performing examinations at a licensee's facility. PDI believes that 8 hours will be acceptable relative to an examiner's abilities in this highly specialized skill area because personnel can gain knowledge of new developments, material failure modes, and other pertinent technical topics through other means. These changes are reflected in 10 CFR 50.55a(b)(2)(xiv) of the final rule."

Staff Evaluation

Subarticle VII-4240, Appendix VII of Section XI of the Code requires 10 hours of annual training to impart knowledge of new developments, material failure modes, and any pertinent technical topics as determined by the license. No hands-on training or practice is required to be included in the 10 hours of training. This training is required of all UT personnel qualified to perform examinations of ASME Code Class 1, 2, and 3 systems.

Independent of the ASME Code, 10 CFR 50.55a(b)(2)(xiv) imposes the requirement that 8 hours of hands-on training with specimens containing cracks be performed no earlier than 6 months prior to performing examinations at a licensee's facility. The licensee contends that its proposed alternative will simplify record keeping and satisfy the need to maintain skills.

As part of the staff's 1999 rulemaking effort to revise 10 CFR 50.55a(b)(2), the issue of UT annual training requirements was reviewed. This review was included in the summary of comments to the September 22, 1999 rule (64 *FR* 51370). In the review, the staff determined that the 10 hours of annual training was inadequate for two reasons. The first reason was that the training does not require practice with flawed specimens. Practice with flaws is necessary because signals can be difficult to interpret. The second reason is related to the length of training and its frequency. Studies have shown that an examiner's capability begins to diminish within 6 months if skills are not maintained. Therefore, examiners must practice on a frequent basis to maintain their capability for proper interpretation of flaws.

Based on resolution of public comments for the above rulemaking, the staff accepted an industry approach advanced by EPRI, which proposed 8 hours of hands-on practice with flawed specimens containing cracks. The practice would occur no earlier than 6 months prior to performing examinations at a licensee's facility. This approach was reflected in 10 CFR 50.55a(b)(2)(xiv) in the September 22, 1999, rulemaking for personnel maintaining their Appendix VIII qualifications. Therefore, the staff has determined that the proposed alternative to use 10 CFR 50.55a(b)(2)(xiv) in lieu of Subarticle VII-4240 will maintain the skill and proficiency of UT personnel at or above the level provided in the Code for annual UT training, thereby providing an acceptable level of quality and safety.

Conclusion

Based on the discussion above, the staff concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-03, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.4 RELIEF REQUEST NO. I4R-04, ALTERNATIVE REQUIREMENTS TO APPENDIX VIII, SUPPLEMENT 4 QUALIFICATION REQUIREMENTS

Code Requirements for which Relief is Requested

The licensee stated that the 1995 Edition of ASME Section XI, Appendix VIII, Supplement 4, subparagraph 3.2(c) requires that the UT performance demonstration results be plotted on a two dimensional plot with the measured depth plotted along the ordinate axis and the true depth plotted along the abscissa axis. For qualification, the plot must satisfy the statistical parameters identified in subparagraph 3.2(c).

Licensee's Proposed Alternative to Code

The licensee stated that the Root Mean Square (RMS) calculations of subparagraph 3.2(c) of ASME Section XI, Appendix VIII, Supplement 4, which utilizes an RMS value of 0.15 depth and the RMS calculations of subparagraph 3.2(b), which utilizes an RMS value of 0.75 length shall be used in lieu of the statistical parameters of subparagraph 3.2(c) of ASME Section XI, Appendix VIII, Supplement 4.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee is seeking relief from the statistical parameters identified in subparagraph 3.2(c) of ASME Section XI, Appendix VIII, Supplement 4 because the proposed alternatives provide an acceptable level of quality and safety.

The licensee stated that the 1999 amendment to the *Federal Register* amended 10 CFR 50.55a(b)(2)(xv)C)(1) which requires a depth sizing acceptance criterion of 0.15 inch RMS be used in lieu of the requirements of subparagraph 3.2(a) and a length sizing requirement of 0.75 inch RMS to be used in lieu of the requirements of 3.2(b) of ASME Section XI, Appendix VIII, Supplement 4.

The licensee stated the statistical parameters rely upon the depth sizing acceptance criteria used in subparagraph 3.2(a) and the length sizing acceptance criteria used in subparagraph 3.2(b) of Supplement 4. For Supplement 4 UT performance demonstrations, the linear regression line of the data required by subparagraph 3.2(c) is not applicable because the performance demonstrations are performed on test specimens with flaws located on the inner 15 percent through-wall. Additionally, the licensee stated the subparagraph 3.2(c) specified value for evaluating the mean deviation of flaw depth is not restrictive enough for evaluating flaw depths within the inner 15 percent of wall thickness. The use of 10 CFR 50.55a(b)(2)(xv)(C)(1) RMS calculations of subparagraph 3.2(a), which utilizes an RMS value of 0.15 inch depth, and RMS calculations of subparagraph 3.2(b), which utilizes an RMS value of 0.75 inch length, in lieu of the statistical parameters of 3.2(c) is appropriate because it is more restrictive.

Staff Evaluation

Supplement 4, Subparagraph 3.2(c), requires that the UT performance demonstration results be plotted on a two-dimensional plot with the measured depth plotted along the ordinate axis and the true depth plotted along the abscissa axis. For qualification, the plot must satisfy the following statistical parameters: (1) slope of the linear regression line is not less than 0.7; (2) the mean deviation of flaw depth is less than 0.25 inches; and (3) correlation coefficient is not less than 0.70.

The licensee proposed eliminating the use of Supplement 4, Subparagraph 3.2(c) which imposes three statistical parameters for depth sizing. The first parameter, 3.2(c)(1), pertains to the slope of a linear regression line. The linear regression line is the difference between actual versus true value plotted along a through-wall thickness. For Supplement 4 performance demonstrations, a linear regression line of the data is not applicable because the performance demonstrations are performed on test specimens with flaws located in the inner 15 percent through-wall. The differences between actual versus true value produce a tight grouping of results which resemble a shotgun pattern. The slope of a regression line from such data is extremely sensitive to small variations, thus making the parameter of Subparagraph 3.2(c)(1) a poor and inappropriate acceptance criterion. The second parameter, 3.2(c)(2), pertains to the mean deviation of flaw depth. The value used in the code is too lax with respect to evaluating flaw depths within the inner 15 percent of wall thickness. Therefore, the licensee proposed to use the more appropriate criterion of 0.15 inch RMS of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies Subparagraph 3.2(a), as the acceptance criterion. The third parameter, 3.2(c)(3), pertains to a correlation coefficient. The value of the correlation coefficient in Subparagraph 3.2(c)(3) is inappropriate for this application since it is based on the linear regression from Subparagraph 3.2(c)(1).

PDI was aware of the inappropriateness of Subparagraph 3.2(c) early in the development of their program. They brought the issue before the appropriate ASME committee which formalized eliminating the use of Supplement 4, Subparagraph 3.2(c) in Code Case N-622. The NRC staff representatives participated in the discussions and consensus process of the code case. Based on the above, the NRC staff believes that the use of Subparagraph 3.2(c) requirements in this context is inappropriate and that the proposed alternative to use the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), specifically 0.15 inch RMS, which modifies the criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c), will provide an acceptable level of quality and safety.

Conclusion

Based on the discussion above, the staff has concluded that the proposed use of the RMS value of 10 CFR 50.55a(b)(2)(xv)(C)(1), which modifies the depth sizing criterion of Appendix VIII, Supplement 4, Subparagraph 3.2(a), in lieu of Subparagraph 3.2(c) will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-04, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.5 RELIEF REQUEST NO. I4R-05, REV. 01, PRESSURE TESTING REACTOR PRESSURE VESSEL HEAD FLANGE SEAL LEAK DETECTION SYSTEM

Code Requirements for which Relief is Requested

The licensee stated that Table IWC-2500-1 of the 1995 Edition of the ASME Code requires that a Visual VT-2 examination be performed during a system leakage test.

Licensee’s Proposed Alternative to Code

The licensee stated that a VT-2 visual examination will be performed on the subject line during reactor vessel flood-up during a refueling outage. The static head developed due to the water above the reactor vessel flange during flood-up will allow for the detection of any gross indications in the line. This examination will be performed with the frequency specified by Table IWC-2500-1 for a system leakage test which corresponds to once each inspection period.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee is seeking relief from the pressure testing requirements for the RPV Head Flange Leak Detection System under Table IWC-2500-1 of ASME Section XI because of the possibility of damage to the reactor vessel O-ring seals during the pressure test.

The licensee stated the RPV Leak Detection line is separated from the reactor pressure boundary by one passive membrane, a silver plated O-ring which is located on the vessel flange. A second O-ring is located on the opposite side of the tap in the vessel flange. This line is required (to be in service) during plant operation in order to indicate failure of the inner flange seal O-ring. Failure of the O-ring would result in the annunciation of a High Level Alarm in the control room. Control room operators would then quantify the leakage rate from the O-ring and then isolate the leak detection line from the drywell sump by closing valve AO 1(2)-220-51. This action is taken in order to prevent steam cutting of the O-ring and the vessel flange. Failure of the inner O-ring is the only condition under which this line is pressurized.

The licensee explained that the configuration of this system precludes manual testing while the vessel head is removed because the odd configuration of the vessel tap, combined with the small size of the tap and the high test pressure required by the Code (1000 psig minimum) prevents the tap in the flange from being temporarily plugged. The opening in the flange is only 3/16" in diameter and is smooth walled making a high pressure temporary seal very difficult. Failure of this seal could possibly cause ejection of the device used for plugging into the vessel. The licensee stated that a pneumatic test performed with the head installed is precluded due to the configuration of the top head. The top head of the vessel contains two grooves that hold the O-rings. The O-rings are held in place by a series of retainer clips spaced 15° apart. The retainer clips are contained in a recessed cavity in the top head. If a pressure test was performed with the head installed, the inner O-ring would be pressurized in a direction opposite to what it would see in normal operation. This test pressure would result in a net inward force on the O-ring that would tend to push it into the recessed cavity that houses the retainer clips. The O-ring material is only 0.050" thick with a silver plating thickness of 0.004" to 0.006" and could very likely be damaged by this deformation into the recessed areas on the top head.

The licensee stated that in addition to the problems associated with the O-ring design that preclude this testing it is also questionable whether a pneumatic test is appropriate for this line. Although the line will initially contain steam if the inner O-ring leaks, the system actually detects leakage rate by measuring the level of condensate in a collection chamber, making the system medium water at the level switch. Finally, the use of a pneumatic test performed at a minimum of 1000 psig would represent an unnecessary risk in safety for the inspectors and test engineers in the unlikely event of a test failure, due to the large amount of stored energy contained in air pressurized to 1000 psig. System leakage testing of this line is precluded because the line will only be pressurized in the event of a failure of the inner O-ring. It is impractical to purposely fail the inner O-ring in order to perform a test.

Staff Evaluation

The subject relief request was resubmitted as Rev. 01 by letter dated August 13, 2003. Pursuant to 10 CFR 50.55a(g)(5)(iii), the licensee is seeking relief from the Visual VT-2 examination requirements for the reactor pressure vessel head flange seal leak detection system piping identified under Table IWC-2500-1 of ASME Section XI because the required testing is impractical.

The purpose of the system is to determine if the inner reactor vessel flange O-ring has failed. If the inner O-ring seal fails, an alarm sounds and the line is isolated to minimize steam cutting to the reactor vessel flange. During operation, the line typically stays depressurized and therefore, the line and the Class 2 piping welds have seen little if any service use during operation. The staff concludes that since the line experiences little if any service use, the likelihood of degradation to the welds and service related failures are low.

The licensee has determined that both personnel safety and equipment damage issues could occur if an alternative form of testing (pneumatic testing) is deployed due to certain design aspects of the line. The staff agrees that equipment damage to the O-ring due to reverse pressurization is likely and ejection of the testing plugs is a risk to personnel. The staff concludes that requiring the licensee to redesign the reactor vessel flange leak detection system in order to perform the testing is impractical and that the static head test visual VT-2 examination performed by personnel provide reasonable assurance of the line's integrity.

Conclusion

The staff concludes that requiring the licensee to perform a design modification in order to perform a pressure test of the Class 2 piping in the reactor vessel flange leak detection system is impractical and that the alternative testing using static head pressure provides adequate assurance of the continued structural integrity of the welds. Therefore, relief is granted pursuant to 10 CFR 50.55a(g)(6)(i) for the fourth ISI interval for Relief Request No. I4R-05, Rev. 01, for Quad Cities Nuclear Power Station, Units 1 and 2. This grant of relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this relief request remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.6 RELIEF REQUEST NO. I4R-06, CONTINUOUS PRESSURE MONITORING OF THE CONTROL ROD DRIVE SYSTEM ACCUMULATORS

Code Requirements for which Relief is Requested

The licensee stated that IWC-2500-1 requires a Visual VT-2 examination to be performed during a system leakage test.

Licensee's Proposed Alternative to Code

As an alternative to the Visual VT-2 examination requirements specified in Table IWC-2500-1, the licensee will perform continuous pressure decay monitoring in conjunction with technical specifications for the nitrogen side of the Control Rod Drive (CRD) System Accumulators including attached piping.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee is seeking relief from the Visual VT-2 examination requirements for the CRD Accumulators and associated piping identified under Table IWC-2500-1 of ASME Section XI because the proposed alternative provides an acceptable level of quality and safety.

The licensee stated that the CRD System Accumulator pressure must be greater than or equal to 940 psig to be considered operable as required by the technical specifications for Quad Cities Nuclear Power Station, Units 1 and 2. The accumulator pressure is continuously monitored by system instrumentation. Since the accumulators are isolated from the source of make up nitrogen, the continuous monitoring of the CRD accumulators functions as a pressure decay type test. Should accumulator pressure fall below 1000 psig, an alarm is received in the control room. The pressure drop for the associated accumulator is then recorded, and the accumulator is recharged in accordance with the station's procedures. If an accumulator requires charging more than twice in a 30 day period, then a leak check is performed to determine the cause of the pressure loss. When leakage is detected, corrective actions are taken to repair the leaking component as required by station procedures.

The licensee stated that since monitoring the nitrogen side of the accumulators is continuous, any leakage from the accumulator would be detected by normal system instrumentation. An additional Visual VT-2 examination performed once per inspection period would not provide an increase in safety, system reliability, or structural integrity. In addition, performance of a Visual VT-2 examination would require applying a leak detection solution to 177 accumulators per unit resulting in additional radiation exposure without any added benefit in safety and is not consistent with minimizing radiation dose to personnel.

Staff Evaluation

The 1995 Edition of ASME Section XI, 1995 Addenda, Table IWC-2500-1, Examination Category C-H, requires that pressure retaining components have a system leakage test, with a Visual VT-2 examination each inspection period. The licensee is requesting relief from the Visual VT-2 examination for the nitrogen side only of the CRD system accumulators and its associated piping. IWA-5221(a) states that a system leakage test is conducted during operation at nominal operating pressure, or when pressured to nominal operating pressure and temperature. IWA-5212(b) states: "When conducting a system leakage test described in IWA-5211(a), the level of system pressure and temperature indicated or recorded by normal operating system instrumentation or by test instrumentation is acceptable." IWA-5241(a) requires that the VT-2 visual examination be conducted by examining the accessible external exposed surfaces of pressure retaining components for evidence of leakage.

The licensee's alternative provides continuous monitoring of the system pressure by monitoring the pressure decay of the system versus inspecting for evidence of leakage by performing a VT-2 visual inspection. In practice, the VT-2 visual inspection to check for the integrity of a pressure retaining boundary is performed while the system is at operating pressure by walking down and inspecting the boundary prior to full power operation. In this instance, the alternative to continuously monitor the accumulators' system pressure with instrumentation approved by the Code under IWA-5212(b) exceeds the requirement to check for boundary integrity only during the system leakage test VT-2 visual inspection. Secondly, should any of the 177 accumulator experience a pressure drop below a prescribed limit due to leakage, an alarmed function provides immediate notification that action is required. The staff concludes the licensee's continuous monitoring of the 177 CRD system accumulators' pressure with normal operating system instrumentation and alarm function provides an acceptable level of safety and quality and, is therefore, acceptable.

Conclusion

Based on the discussion above, the staff concludes that the proposed alternative, continuous pressure monitoring of the nitrogen side of the CRD system accumulators and its associated piping, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-06, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.7 RELIEF REQUEST NO. I4R-07, ALTERNATIVE RULES FOR CORRECTIVE MEASURES IF LEAKAGE OCCURS AT BOLTED CONNECTIONS

Code Requirements for which Relief is Requested

The licensee stated that IWA-5250(a)(2) requires that if leakage occurs at a bolted connection, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. The bolt selected shall be the one closest to the source of leakage. When the removed bolt has evidence of degradation, all remaining bolting in the connection shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100.

Licensee's Proposed Alternative to Code

As an alternative to the requirements of IWA-5250(a)(2), the licensee stated that, consistent with the methodology of Code Case N-566-2, the following requirements of either (a) or (b) will be met for leakage at bolted connections:

(a) The leakage will be stopped, and the bolting and component material will be reviewed for joint integrity as described in (c) below.

(b) If the leakage is not stopped, the licensee will evaluate the structural integrity and consequences of continuing operation, and the effect on the system operability of continued leakage. This engineering evaluation will include the considerations listed in (c) below.

(c) The evaluation of (a) and (b) above is to determine the susceptibility of the bolting to corrosion and failure. This evaluation will include the following:

(1) the number and service age of the bolts;
(2) bolt and component material;
(3) corrosiveness of process fluid;
(4) leak location and system function;
(5) leakage history at the connection or other system components;
(6) visual evidence of corrosion at the assembled connection.

The licensee stated if any of the above parameters indicates a need for further examination, the corrective action will be taken in accordance with IWA-5250(a)(2).

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee stated the proposed alternative provides an acceptable level of quality and safety. The licensee stated that removal of pressure retaining bolting at mechanical connections for VT-3 visual examination and subsequent evaluation in locations where leakage has been identified is not always the most prudent course of action to determine condition of the bolting and/or the root cause of the leak. The requirement to remove, examine and evaluate bolting in this situation does not allow consideration of other factors which may indicate the condition of mechanical joint bolting. The licensee stated that other factors which should be considered in an evaluation of bolting condition when leakage has been identified at a mechanical joint include, but should not be limited to:

(1) Bolting materials
(2) Corrosiveness of process fluid
(3) Service age of joint bolting materials
(4) Leakage location
(5) Leakage history at connection
(6) Visual evidence of corrosion at connection (connection assembled)
(7) Plant/Industry studies of similar bolting materials in a similar environment
(8) Condition and leakage history of adjacent components

The licensee cited an example where strict application of IWA-5250(a) would be excessive, such as in the case when complete replacement of bolting materials occurs at a mechanical joint during a plant outage. In some cases, leakage is identified at these joints during pressurization and the cause of this leakage is often due to thermal expansion of the piping and bolting materials at the joint. In this case, the licensee stated that removal of any joint bolting to evaluate for corrosion would be unwarranted.

Staff Evaluation

In accordance with the 1995 Edition with 1996 Addenda of the ASME Code, Section XI, when leakage occurs at bolted connections, one of the bolts shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100. In lieu of the Code-required removal of bolting to perform a VT-3 visual examination, the licensee has proposed to perform an evaluation of the bolted connection to determine the susceptibility of the bolting to corrosion and the potential for failure. Most leakage at bolted connections are caused due to thermal expansion of the piping and bolting materials and subsequent fluid seepage at the joint gasket. Therefore, proper torquing of the bolting in most cases, would stop the leakage. Following stoppage of a leak, the bolting and component material will be evaluated to determine the susceptibility of the bolting to corrosion and failure considering all of the following factors:

- The number and service age of the bolts;
- Bolt and component material;
- Corrosiveness of process fluid;
- Leakage location and system function;

- Leakage history at connection or other system components;
- Visual evidence of corrosion at the assembled connection.

If the evaluation determines that the leaking condition has not degraded the fasteners, then no further action is necessary. If the initial evaluation indicates the need for a more in-depth evaluation, the bolt closest to the source of leakage will be removed, VT-3 examined, and evaluated in accordance with IWA-3100. In a situation where leakage is not stopped, the licensee indicated it would evaluate structural integrity consequences of continued operation, and the effect on the system operability of continued leakage along with the consideration of the above six factors. Such an evaluation will be the basis for deferral of removing the bolt closest to the source of leakage to the next outage. This alternative allows the licensee to utilize a systematic approach and sound engineering judgement, provided that as a minimum, all of the six evaluation factors listed in the licensee's proposed alternative are considered.

This proposed alternative engineering evaluation considers all the factors necessary to identify degradation of the bolts in any leaking bolted connection. Accordingly, the use of this type of engineering evaluation is expected to result in the identification of such degradation and corrective actions when appropriate, and avoids unnecessary joint disassembly when the bolts are fit for service. As a result, the licensee's alternative to the Code-required removal of bolting at a joint when leakage occurs will provide an acceptable level of quality and safety since the integrity of the joint will be maintained.

Conclusion

Based on the discussion above, the staff has concluded that the proposed alternative of corrective measures for bolting at a leaking connection will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-07, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.8 RELIEF REQUEST NO. I4R-08, EVALUATION CRITERIA FOR TEMPORARY ACCEPTANCE OF FLAWS FOR MODERATE ENERGY CLASS 2 AND 3 PIPING

Code Requirements for which Relief is Requested

The licensee stated that IWC-3122.3 requires that a component whose volumetric or surface examination detects flaws may be acceptable for continued service without a repair/replacement activity if an analytical evaluation is performed in accordance with IWC-3600. Similar requirements for visual examinations are contained in IWC-3132.3. The licensee also stated that in the 1995 Edition with the 1996 Addenda of ASME Section XI, IWC-3600, Analytical Evaluation of Flaws, and IWD-3000, Acceptance Standards, are in the course of preparation and state that the requirements of IWB may be used.

Licensee's Proposed Alternative to Code

The licensee proposes that when using analytical evaluation as the method of acceptance for flaws in moderate energy Class 2 or 3 piping, the provisions of Code Case N-513-1 will be followed without performing a repair/replacement activity. This acceptance will be temporary and will remain in effect for a limited time, not exceeding the time to the next scheduled outage.

The licensee stated that the alternative may implement this method or one of the other methods contained in ASME Section XI to accept detected flaws; however, in no case will the temporary evaluation process be applied to:

- components other than pipe or tube,
- leakage through a gasket,
- threaded connections with nonstructural seal welds for leakage prevention, or
- degraded socket welds.

The licensee stated that when applying the methods of Code Case N-513-1, the specific safety factors contained in Paragraph 4.0 of the Case will be satisfied. These conditions are consistent with those contained in 10 CFR 50.55a(b)(2)(xiii) regarding the use of Code Case N-513.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee stated the proposed alternative provides an acceptable level of quality and safety. ASME Section XI Code Case N-513 is conditionally approved for use in Revision 13 of Regulatory Guide 1.147; however, this Case is not applicable to the 1996 Addenda which is the code of record for the fourth ISI interval. Code Case N-513-1 has since been issued in Supplement 11 of the 1998 Edition and is currently applicable through the 2001 Edition. This revision of the Code Case has not yet been approved for general use by the NRC in Regulatory Guide 1.147.

The licensee stated that Code Case N-513-1 revises the base case to expand the temporary acceptance methodology from Class 3 moderate energy piping to Class 2 and 3 moderate energy piping. Both cases provide requirements which may be followed for temporary acceptance of flaws in ASME Section III, ANSI B31.1, and ANSI B31.7 piping designated as Class 2 or 3. This acceptance is limited to moderate energy piping defined as piping whose maximum operating temperature does not exceed 200 °F and whose maximum operating pressure does not exceed 275 psig. The provisions of the case demonstrate the integrity of the item containing the flaw for a limited period of time until appropriate repair/replacement or additional examination activities can be performed.

Staff Evaluation

The licensee's alternative requests the use of Code Case N-513-1 as an alternative to the requirements of IWC-3122.3. IWC-3122.3 allows the continued use of a component without a repair/replacement activity provided an engineering evaluation is performed. The licensee proposes that the requirements of Code Case N-513-1 be used for performing the engineering evaluation of moderate energy ASME Class 2 and 3 components. Code Case N-513-1 requires that for through-wall flaws, leakage shall be observed by daily walkdowns to confirm the analysis conditions used in the evaluation remain valid. A repair/replacement activity may be deferred on accepted flaws including through wall flaws, for a limited time not exceeding the time to the next scheduled outage.

Code Case N-513-1 has been reviewed by the staff and found it to be conditionally acceptable, but this acceptability has not yet been reflected in the latest revision of Regulatory Guide 1.147. The staff concludes that the licensee's use of N-513-1 provides an acceptable level of quality and safety, and is therefore, acceptable for use as an alternative.

<u>Conclusion</u>

Based on the discussion above, the staff has concluded that the proposed alternative to use Code Case N-513-1 for the performance of temporary acceptance of flaws in moderate energy ASME Code Class 2 and 3 piping systems for a limited time not to exceed the time until the next scheduled outage will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-08, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2, or until such time as the Code Case is referenced in 10 CFR 50.55a. At that time, if the licensee intends to continue to implement this code case, the licensee must follow all provisions referenced in Code Case N-513-1, with limitations as referenced in 10 CFR 50.55a, if any. All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

2.9 RELIEF REQUEST NO. I4R-09, PRESSURE RETAINING WELDS IN PIPING, SUBJECT TO APPENDIX VIII, SUPPLEMENT 11

<u>Code Requirements for which Relief is Requested</u>

The licensee is requesting relief from the weld overlay requirements in the following paragraphs to Section XI, Appendix VIII, Supplement 11.

Paragraph 1.1(d)(1) requires that all base metal flaws be cracks.

Paragraph 1.1(e)(1) requires that at least 20 percent but less than 40 percent of the flaws shall be oriented within ±20 degrees of the axial direction.

Paragraph 1.1(e)(1) also requires that the rules of IWA-3300 shall be used to determine whether closely spaced flaws should be treated as single or multiple flaws.

Paragraph 1.1(e)(2)(a)(1) requires that a base grading unit shall include at least 3 inches of the length of the overlaid weld and the outer 25 percent of the overlaid weld and base metal on both sides.

Paragraph 1.1(e)(2)(a)(3) requires that for unflawed base grading units, at least 1 inch of unflawed overlaid weld and base metal shall exist on either side of the base grading unit.

Paragraph 1.1(e)(2)(b)(1) requires that an overlay grading unit shall include the overlay material and the base metal-to-overlay interface of at least 6 sq. inches. The overlay grading unit shall be rectangular, with minimum dimensions of 2 inches.

Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.1 inch are reported as being intrusions into the overlay material.

Licensee's Proposed Alternative to Code

The proposed alternative is to use the EPRI-PDI program in lieu of the requirements of Section XI, Appendix VIII, Supplement 11.

Licensee's Basis for Relief

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee stated the proposed alternative to use the EPRI-PDI program in lieu of the Appendix VIII, Supplement 11 provides an acceptable level of quality and safety. The licensee stated that paragraph 1.1(d)(1), requires that all base metal flaws be cracks. As illustrated in the subject relief request, implanting a crack requires excavation of the base material on at least one side of the flaw. While this may be satisfactory for ferritic materials, it does not produce a useable axial flaw in austenitic materials because the sound beam, which normally passes only through base material, must now travel through weld material on at least one side, producing an unrealistic flaw response. To resolve this issue, the PDI program revised this paragraph to allow use of alternative flaw mechanisms under controlled conditions. For example, alternative flaws shall be limited to when implantation of cracks precludes obtaining an effective UT response, flaws shall be semi-elliptical with a tip width of less than or equal to 0.002 inches, and at least 70 percent of the flaws in the detection and sizing test shall be cracks and the remainder shall be alternative flaws.

The licensee stated relief is requested to allow closer spacing of flaws provided they did not interfere with detection or discrimination. The existing specimens used to date for qualifications to the Tri-party (NRC/BWROG/EPRI) agreement have a flaw population density greater than allowed by the current Code requirements. These samples have been used successfully for all previous qualifications under the Tri-party agreement program to Supplement 11. The PDI Program has merged the Tri-party test specimens into their weld overlay program. For example: the requirement for using IWA-3300 for proximity flaw evaluation in paragraph 1.1(e)(1) was excluded, instead indications will be sized based on their individual merits; paragraph 1.1(d)(1) includes the statement that intentional overlay fabrication flaws shall not interfere with UT detection or characterization of the base metal flaws; paragraph 1.1(e)(2)(a)(1) was modified to require that a base metal grading unit include at least 1 in. of the length of the overlaid weld, rather than 3 inches; paragraph 1.1(e)(2)(a)(3) was modified to require sufficient unflawed overlaid weld and base metal to exist on all sides of the grading unit to preclude interfering reflections from adjacent flaws, rather than the 1 inch requirement of Supplement 11; paragraph 1.1(e)(2)(b)(1) was modified to define an overlay fabrication grading unit as including the overlay material and the base metal-to-overlay interface for a length of at least 1 inch rather than the 6 square inch requirement of Supplement 11; and paragraph 1.1(e)(2)(b)(2) states that overlay fabrication grading units designed to be unflawed shall be separated by unflawed overlay material and unflawed base metal-to-overlay interface for at least 1 inch at both ends, rather than around its entire perimeter.

In addition, the licensee stated that the requirement for axially oriented overlay fabrication flaws in paragraph 1.1(e)(1) was excluded from the PDI Program as an improbable scenario. Weld overlays are typically applied using automated gas tungsten arc welding techniques with the filler metal being applied in a circumferential direction. Because resultant fabrication induced discontinuities would also be expected to have major dimensions oriented in the circumferential direction, axial overlay fabrication flaws are unrealistic.

The PDI Program revised paragraph 2.0 to permit the overlay fabrication flaw test and the base metal flaw tests be performed separately.

The requirement in paragraph 3.2(b) for reporting all extensions of cracking into the overlay is omitted from the PDI Program because it is redundant to the (root mean square) RMS calculations performed in paragraph 3.2(c) and it's presence adds confusion and ambiguity to depth sizing as required by paragraph 3.2(c). This also makes the weld overlay program consistent with the Supplement 2 depth sizing criteria.

Finally, the licensee indicated that the PDI Program omits the phrase "and base metal on both sides," in paragraph 1.1(e)(2)(a)(1) because some of the qualification samples included flaws on both sides of the weld. To avoid confusion, several instances of the term "cracks" or "cracking" were changed to the term "flaws" because of the use of alternative flaw mechanisms.

Staff Evaluation

The nuclear power industry tasked PDI with the implementation of a Section XI, Appendix VIII, Supplement 11 performance demonstration program. The PDI program is routinely assessed by the staff for consistency with Code and proposed Code changes. In order to meet the scheduled implementation date of November 22, 2001, specified in 10 CFR 50.55a(g)(6)(ii)(C), PDI evaluated the applicability of using test specimens from an existing weld overlay program for its Supplement 11 performance demonstration program. Their evaluation identified differences with Paragraphs 1.1(e)(1), 1.1(e)(2)(a)(1), 1.1(e)(2)(a)(3), 1.1(e)(2)(b)(1), and 3.2(b). PDI proposed through Code that these paragraphs be changed to permit using the existing weld overlay test specimens.

Paragraph 1.1(e)(1) requires that at least 20 percent but less than 40 percent of the flaws shall be oriented within ±20 degrees of the axial direction. In the PDI program the flaws satisfy the requirement and specifies that the flaws must be in the base metal. This is a tightening of the requirements. Hence, PDIs application of flaw angles to the axial direction is acceptable.

Paragraph 1.1(e)(1) also requires that the rules of IWA-3300 shall be used to determine whether closely spaced flaws should be treated as single or multiple flaws. PDI treats each flaw as an individual flaw and not as part of a system of closely spaced flaws. PDI controls the flaws going into a test specimen set such that the flaws are free of interfering reflections from adjacent flaws. In some cases, this would permit flaws to be closer together than what is allowed by IWA-3300, thus making the performance demonstration more challenging. Hence, PDI's application for closely spaced flaws is acceptable.

Paragraph 1.1(e)(2)(a)(1) requires that a base grading unit shall include at least 3 inches of the length of the overlaid weld, and the base grading unit includes the outer 25 percent of the overlaid weld and base metal on both sides. The PDI program reduced the criteria to 1 inch of the length of the overlaid weld and eliminated from the grading unit the need to include both sides of the weld. The test specimens from the existing weld overlay program have flaws on both sides of the welds which prevents them from satisfying the base grading unit requirements. These test specimens have been used successfully for testing the proficiency of personnel for over 16-years. This is a more challenging test because the individual must locate the flaw on the correct side of the weld. Hence, PDI's application of the 1 inch length of the overlaid weld base grading unit and elimination from the grading unit the need to include both sides of the weld is acceptable.

Paragraph 1.1(e)(2)(a)(3) requires that for unflawed base grading units, at least 1 inch of unflawed overlaid weld and base metal shall exist on either side of the base grading unit. This is to minimize the number of false identification of extraneous reflectors. The PDI program stipulates that unflawed overlaid weld and base metal exists on all sides of the grading unit and be free of interfering reflections from adjacent flaws which addresses the same concerns as Code. Hence PDI's application of the variable flaw free area adjacent to the grading unit is acceptable.

Paragraph 1.1(e)(2)(b)(1) requires that an overlay grading unit shall include the overlay material and the base metal-to-overlay interface of at least 6 square inches. The overlay grading unit shall be rectangular, with minimum dimensions of 2 inches. The PDI program reduces the base metal-to-overlay interface to at least 1 inch (in lieu of a minimum of 2 inches) and eliminates the minimum rectangular dimension. This criterion is more challenging than Code because of the variability associated with the shape of the grading unit. Hence, PDI's application of the grading unit is acceptable.

Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.1 inch are reported as being intrusions into the overlay material. The PDI program omits this criteria. The PDI program requires that cracks be sized to the tolerance specified in Code which is 0.125 inches. Since the Code tolerance is close to the 0.1 inch value of Paragraph 3.2(b), any crack extending beyond 0.1 inch into the overlay material would be identified from its dimensions. The reporting of an extension in the overlay material is redundant for performance demonstration testing. Hence, PDI's omission of highlighting a crack extending beyond 0.1 inch into the overlay material is acceptable.

In addition to the changes for flaw locations, PDI determined that certain Supplement 11 requirements pertaining to location and size of cracks would be extremely difficult to achieve. In an effort to satisfy the requirements, PDI developed a process for fabricating flaws that exhibited crack like reflective characteristics. Instead of all flaws being cracks as required by Paragraph 1.1(d)(1), the PDI weld overlay performance demonstrations contain at least 70 percent cracks with the remainder being fabricated flaws exhibiting crack like reflective characteristics. The NRC has reviewed the flaw fabrication process, and has compared the reflective characteristics between cracks and fabricated flaws. NRC found the fabricated flaws acceptable for the application.

Conclusion

Based on the discussion above, the staff has concluded that the proposed alternative to use the EPRI-PDI program in lieu of the requirements of Section XI, Appendix VIII, Supplement 11 will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative under ISI Relief Request No. I4R-09, is authorized for the fourth 10-year ISI interval for Quad Cities Nuclear Power Station, Units 1 and 2. All other

ASME Code, Section XI requirements for which relief was not specifically requested and approved in this safety evaluation remain applicable, including third party review by the Authorized Nuclear Inservice Inspector.

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