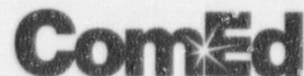


Commonwealth Edison Company  
Quad Cities Generating Station  
22710 206th Avenue North  
Cordova, IL 61242-9740  
Tel 309-654-2241



SVP-99-170

August 13, 1999

U.S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, D C 20555

Quad Cities Nuclear Power Station, Units 1 and 2  
Facility Operating License Nos. DPR-29 and DPR-30  
NRC Docket Nos. 50-254 and 50-265

Subject: Relief Requests CR-25, CR-26, CR-27, CR-28, CR-30, PR-11, PR-12  
and PR-13

- References:
- 1) American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, 1992 Edition, with the 1992 Addenda, IWE Inservice Inspection Plan (containment)
  - 2) American Society of Mechanical Engineers Boiler and Pressure Vessel Code Section XI, 1989 Edition, no Addenda, Inservice Inspection Plan

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The proposed relief requests are provided as an attachment to this letter.

Approval of the attached relief requests CR-25, CR-26, CR-27, CR-30 and PR-12 are needed prior to the start of our next Unit 2 refueling outage which is currently scheduled to start January 21, 2000. Relief requests CR-28, PR-11 and PR-13 are not needed to support this outage and should be reviewed accordingly.

In addition, as permitted by Regulatory Guide 1.147 "Inservice Inspection Code Case Acceptability - ASME Section 1, Division 1," (Revision 12) paragraph C.1., Acceptable Code Cases, the following code cases will be utilized by Quad Cities Station, N-458, N-496-1, N-498-1, N-504-1 and N-524 during the current Inservice Inspection Interval.

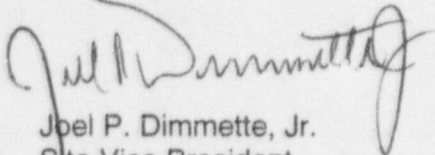
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August 13, 1999  
U.S. Nuclear Regulatory Commission  
Page 2

Should you have any questions concerning his letter, please contact  
Mr. C. C. Peterson at (309) 654-2241, extension 3609.

Respectfully,



Joel P. Dimmette, Jr.  
Site Vice President  
Quad Cities Nuclear Power Station

Attachments:

Attachment A: Relief Request CR-25  
Attachment B: Relief Request CR-26  
Attachment C: Relief Request CR-27  
Attachment D: Relief Request CR-28  
Attachment E: Relief Request CR-30  
Attachment F: Relief Request PR-11  
Attachment G: Relief Request PR-12  
Attachment H: Relief Request PR-13

cc: Regional Administrator – NRC Region III  
NRC Senior Resident Inspector – Quad Cities Nuclear Power Station

**Attachment A, Relief Request CR-25  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 2)**

**COMPONENT IDENTIFICATION**

Code Class:	MC
References:	IWE-2200 (g)
Examination Category:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternate Requirements for Preservice Examination for Re-applied Paint/Coating
Units	1 and 2
Component Numbers:	All Class MC Components

**CODE REQUIREMENT**

ASME Section XI, 1992 Edition, 1992 Addenda, paragraph IWE-2200 (g) requires that when paint or coatings are re-applied, the condition of the new paint or coating shall be documented in the preservice examination records.

**CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED**

Relief is requested from the requirement to perform a preservice inspection of new paint or coatings per IWE-2200 (g).

**BASIS FOR RELIEF:**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

In accordance with ASME Section III, NE-2110 (b), paint and coatings are not part of the containment pressure boundary under current Code rules because they are not associated with the pressure retaining function of the component. The interiors of the drywell and suppression chamber are painted to prevent corrosion. Neither paint nor coatings contribute to the structural integrity or leak tightness of the containment. Furthermore the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement per IWA-4111 (b)(5). The process of applying paint or coatings at Quad Cities Station is performed and controlled under existing station work control procedures. Quad Cities Station General Work Maintenance/Modification Work Specification R-4411, Section 993, provides the requirements for safety related paints and coatings applied to any substrate inside the primary containment. Procedures used to apply paint/coating repairs and perform coating inspections are reviewed and approved in accordance with these requirements. Coating applicators are qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations. The adequacy of applied coatings is verified through the inspections performed by qualified coating inspectors.

**Attachment A, Relief Request CR-27  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 2)**

Recording the condition of reapplied coating in the preservice record does not substantiate the containment structural integrity. Should deterioration of the coating in the re-applied area occur, the area will require additional evaluation regardless of the preservice record. Recording the condition of new paint or coating in the preservice records does not increase the level of quality and safety of the containment. The requirement to perform a preservice examination when paint or coatings are re-applied has been removed from Subsection IWE of ASME Section XI, 1998 Edition.

**PROPOSED ALTERNATIVE PROVISIONS**

The re-applied paint and coatings on the containment vessel will be examined as required by Quad Cities Station Specification R-4411, which was developed in accordance with our Quality Assurance Program requirements. If degradation of the coating is identified, additional measures will be applied to determine if the containment pressure boundary is affected. Repairs to the primary containment boundary, if required, would be conducted in accordance with applicable ASME Section XI Code rules.

**APPLICABLE TIME PERIOD**

Relief is requested for the first ten-year interval of the IWE Program for Quad Cities Units 1 and 2.

**Attachment B, Relief Request CR-26  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 2)**

**COMPONENT IDENTIFICATION**

Code Class:	MC
References:	IWE 2500(b) Table 2500-1
Examination Category:	Not Applicable
Item Numbers:	Not Applicable
Description:	Alternate Requirements for Visual examination to be performed in accordance with Table IWE-2500-1 prior to removal of paint or coatings.
Units	1 and 2
Component Numbers:	All Class MC Components

**CODE REQUIREMENT**

ASME Section XI, 1992 Edition, 1992 Addenda, IWE-2500 (b) requires a visual examination to be performed in accordance with Table IWE-2500-1 prior to removal of paint or coatings.

**CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED**

Relief is requested for the IWE-2500 (b) requirement to perform visual examinations in accordance with Table IWE-2500-1 prior to removal of paint or coatings.

**BASIS FOR RELIEF**

In accordance with to 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Paint and coatings are not part of the containment pressure boundary under current Code rules (ASME Section III, NE-2110 (b)) because they are not associated with the pressure retaining function of the component. The interiors of the drywell and suppression chamber are painted to prevent corrosion. Neither paint nor coatings contribute to the structural integrity or leak tightness of the containment. Furthermore, the paint and coatings on the containment pressure boundary were not subject to Code rules when they were originally applied and are not subject to ASME Section XI rules for repair or replacement per IWA-4111 (b)(5). The process of applying paint or coatings at Quad Cities Station is performed and controlled under existing station work control procedures. Quad Cities Station General Work Maintenance/Modification Work Specification R-4411, Section 993, provides the requirements for safety related paints and coatings applied to any substrate inside the primary containment. Procedures used to apply coating/coating repairs and perform coating inspections are reviewed and approved in accordance with these requirements. Coating applicators are qualified to demonstrate their ability to satisfactorily apply the coatings in accordance with the manufacturer's recommendations. The adequacy of applied coatings is verified through the inspections performed by qualified coating inspectors.

**Attachment B, Relief Request CR-26  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 2)**

Degradation or discoloration of the paint or coating materials on containment would be an indicator of potential degradation of the containment pressure boundary. Additional measures would have to be employed to determine the nature and extent of any degradation, if present. The application of ASME Section XI rules for removal of paint or coatings, when unrelated to an ASME Section XI repair or replacement activity, is a burden without a compensating increase in quality or safety. The requirement to perform a visual examination prior to removal when paint or coatings are reapplied has been removed from Subsection IWE of ASME Section XI, 1998.

**PROPOSED ALTERNATIVE PROVISIONS**

The paint and coatings in the containment will be examined as required by Quad Cities Station Specification R-4411, which was developed in accordance with our Quality Assurance Program requirements. If degradation of the coating is identified, additional measures will be applied to determine if the containment pressure boundary is affected. Repairs to the primary containment boundary, if required, would be conducted in accordance with ASME Section XI Code rules.

**APPLICABLE TIME PERIOD**

Relief is requested for the first ten year interval of the IWE Program for Quad Cities Units 1 and 2.

**Attachment C, Relief Request CR-27  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 3)**

**COMPONENT IDENTIFICATION**

Code Class:	1, 2, and 3
References:	IWB-2430, IWC-2430, and IWD-2430
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Alternative Rules for Determining Additional Examinations

**CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED**

Quad Cities Station requests relief to use the Additional Examination criteria of the 1998 Edition of ASME Section XI in lieu of the requirements of the 1989 Edition for determining expansion samples.

**BASIS FOR RELIEF**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that the proposed alternatives would provide an acceptable level of quality and safety.

Paragraphs IWB-2430 and IWC-2430 require additional examinations to be performed when examinations reveal indications exceeding the applicable acceptance standards of Table IWB-3410-1 for Class 1 and of article IWC-3000 for Class 2. If these additional examinations reveal indications exceeding the applicable acceptance standards of Table IWB-3410-1 for Class 1 and of article IWC-3000 for Class 2, further expansion criteria is also provided. The 1989 Edition of ASME Section XI does not address additional examinations for Class 3 components.

When examinations detect indications exceeding allowable acceptance standards, it is required to examine additional components to reveal if similar conditions exist in related components. The requirements of ASME Section XI, regarding examination sample expansion, have been greatly enhanced from the 1989 Edition with the development of later editions of the Code up to and including the 1998 Edition. The conditions of IWB-2430 and IWC-2430 in the 1989 Edition are vague regarding the number of components to expand to, the type (or expansion criteria) of components to be included, and the subsequent schedule impacts.

Beginning with the 1992 Edition, and continuing up to and including the 1998 Edition, these expansion criteria have been updated to provide further detail in the number and type of components to be examined following the detection of an indication that exceeds the applicable acceptance standards. The expansion methodology has also changed to incorporate considerations such as the material, service, flaw type, and the relevant conditions detected.

**Attachment C, Relief Request CR-27  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 3)**

**BASIS FOR RELIEF** (Con't)

The 1998 Edition of ASME Section XI defines the Class 1 additional examinations required, per IWB-2430, based on the number of examinations scheduled for the current inspection period. These additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaws or relevant conditions. If indications are then found in this expanded sample, further examinations shall be conducted on the remaining number of components with similar material and service that are subject to the same type of flaws or relevant conditions. These requirements are excluded from Category B-P. This philosophy is consistent with the examination requirements of this category since 100% of the components are inspected each period. Finally, the 1998 Edition of ASME Section XI adds guidance in subparagraph IWB-2430(e) on additional examinations for welded attachments that are inspected as a result of identified component support deformation.

The 1998 Edition of ASME Section XI defines the Class 2 additional examinations required per IWC-2430 based on 20% of the number of examinations scheduled for the current inspection *Interval* instead of those scheduled for the current Period. These additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaws or relevant conditions. If indications are then found in this expanded sample, further examinations shall be conducted on the remaining number of components with similar material and service that are subject to the same type of flaws or relevant conditions. This criteria clarifies what is meant by "similar components" in the 1989 Edition of ASME Section XI. These requirements are excluded from Category C-H. This philosophy is consistent with the examination requirements of that category since 100% of the components are inspected each period. Finally, the 1998 Edition of ASME Section XI adds guidance in subparagraph IWC-2430(d) on additional examinations for welded attachments that are inspected as a result of identified component support deformation.

As previously stated, the 1989 Edition of Section XI does not address Class 3 additional examinations in article IWD-2000. The 1998 Edition of Section XI defines the Class 3 additional examinations required per IWD-2430 based on 20% of the number of examinations scheduled for the current inspection Interval. These additional examinations shall be selected from welds, areas, or parts of similar material and service. This additional selection may require inclusion of piping systems other than the one containing the flaws or relevant conditions. If indications are then found in this expanded sample, the extent of further examinations shall be determined based upon an engineering evaluation of the root cause of the flaws or relevant conditions. The corrective measures shall be documented in accordance with IWA-6000. These requirements are excluded from Category D-B. This philosophy is consistent with the examination requirements of that category since



**Attachment C, Relief Request CR-27  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 3 of 3)**

**BASIS FOR RELIEF** (Con't)

100% of the components are inspected each period. Finally, the 1998 Edition of ASME Section XI adds guidance in subparagraph IWD-2430(d) on additional examinations for welded attachments that are inspected as a result of identified component support deformation.

Quad Cities Station Interval Schedule and Selection Document defines the number of examinations to be performed based on the station's selection criteria which is comprised of the applicable Code requirements, Code Cases, and Relief Requests. The number of scheduled examinations as detailed in these documents will be used as the basis for applying the expansion criteria detailed above.

Quad Cities Station believes that the expansion criteria of the 1998 Edition of ASME Section XI, as outlined above, adequately details the number and type of components to be examined following the detection of an indication that exceeds the applicable acceptance standards. The methodology of the expansion also incorporates the requirements to take into consideration including the material, service, flaw type, and relevant conditions, which will provide a sample much more characteristic of the detected indications and component conditions. This philosophy improves on the criteria of the 1989 Edition of ASME Section XI by taking advantage of an inspection for cause approach and focusing the additional examinations on those areas of similar conditions that could result in similar indications and flaws.

**PROPOSED ALTERNATE PROVISIONS**

As an alternative to the requirements of the 1989 Edition of ASME Section XI, Quad Cities Station will utilize the expansion criteria detailed in paragraphs IWB-2430, IWC-2430, and IWD-2430 (Additional Examinations) of the 1998 Edition of ASME Section XI for Class 1, Class 2, and Class 3 components, as appropriate using the schedule quantities detailed in the station Selection Document, when examinations reveal indications exceeding the applicable acceptance standards.

**APPLICABLE TIME PERIOD**

Relief is requested for the third ten-year inspection interval of the Inservice Inspection Program for Quad Cities Units 1 and 2.

**Attachment D, Relief Request CR-28  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 3)**

**COMPONENT IDENTIFICATION**

Code Class:	1, 2, and 3
References:	IWA-4800, IWA-6200, IWA-7520
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission

**CODE REQUIREMENTS FROM WHICH RELIEF IS REQUESTED**

Relief is requested from the requirement to submit the ISI Summary Report within 90 days of the completion of each refueling outage as required per IWA-6230.

**BASIS FOR RELIEF**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety.

ASME Section XI has recently reevaluated the Code criteria for reporting inservice inspection results, repairs and replacements, and has concluded that the current requirements are no longer effective. To address this issue, ASME Section XI has issued Code Case N-532, "Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 and IWA-6000". Code Case N-532 provides an alternative to the current ASME Section XI repair and replacement documentation requirements as well as regulatory reporting requirements relating to inservice inspection. This alternative will reduce the resources required to prepare and submit NIS-2 forms and the ISI Summary Report currently required after each refueling outage. This is a significant reduction in the "administrative" burden required by ASME Section XI, IWA-6000. The use of Code Case N-532 only affects documentation and reporting requirements and does not affect the level of quality or safety provided by the Inservice Inspection Program. Furthermore, the reporting requirements of the code case provide for a more comprehensive status report of the inservice inspection program with respect to both period and interval inspection requirements.

**Attachment D, Relief Request CR-28  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 3)**

**BASIS FOR RELIEF** (Cont'd)

Code Case N-532 was approved by the ASME Boiler and Pressure Vessel Code Committee on December 12, 1994, but is not yet included in the most recent listing of NRC approved code cases provided in Revision 12 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI Division 1".

The NRC Staff has made recommendations supporting the development of Code Case N-532 in SECY-94-093, "NRC Staff Assessment of Reporting Requirements for Power Reactor Licensees". The use of Code Case N-532 is consistent with the recommendations of SECY-94-093 and provides appropriate documentation to the regulatory and enforcement authorities having jurisdiction at the plant.

This request to use Code Case N-532 includes compliance with the Code Case with the following clarification regarding reporting of "corrective measures". ASME Section XI uses the term "corrective measures" in two different ways. One use of the term involves Code required activities such as repairs and replacements. The other use of the term, as found in IWX-3000, involves maintenance activities that do not involve repairs or replacements. With this clarification, Quad Cities Station proposes not to report corrective measures which only include routine maintenance activities such as tightening threaded fittings to eliminate leakage, torquing of fasteners to eliminate leakage at bolted connections, replacing valve packing due to unacceptable packing leakage, tightening loosened mechanical connections on supports, adjusting and realigning supports, cleaning up corrosion on components resulting from leakage, etc.

Including these routine maintenance activities in the Owner's Activity Report Form OAR-1 required by Code Case N-532 would be a significant expansion of current requirements. In addition, it would be an unnecessary reporting and review burden which would provide little benefit. Reporting of these minor maintenance corrective measures has no safety significance and detracts from the reporting of meaningful information on repairs, replacements, and evaluations performed to accept flaws and relevant conditions exceeding ASME Section XI acceptance criteria. Only corrective measures which refer to Code required repairs and replacements will be reported in compliance with Code Case N-532.

**Attachment D, Relief Request CR-28  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 3 of 3)**

**PROPOSED ALTERNATIVE CRITERIA**

Quad Cities Station considers the alternative documentation and reporting requirements of Code Case N-532 to be a reasonable alternative and an improvement to existing requirements. Therefore, Form OAR-1 will be submitted following the end of the Inspection Period. Because the use of this alternative only affects documentation and reporting requirements, Quad Cities Station considers this alternative to provide an acceptable level of quality and safety.

**APPLICABLE TIME PERIOD**

Relief is requested for the third period of the third ten-year interval of the Inservice Inspection Program for Quad Cities Units 1 and 2.

**Attachment E, Relief Request CR-30  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 3)**

**COMPONENT IDENTIFICATION**

Code Class:	MC
References:	Table 2500-1
Examination Category:	E-G
Item Number:	8.10
Description:	Alternate Requirements for VT-1 Visual Examination of Pressure Retaining Bolting
Units	1 and 2
Component Numbers:	All Class MC Components

**CODE REQUIREMENT**

ASME Section XI, 1992 Edition with the 1992 Addenda, Table IWE-2500-1, Examination Category E-G, Item No. 8.10 requires a VT-1 visual examination of 100% of each bolted connection during the Inspection Interval.

**CODE REQUIREMENTS FROM WHICH RELIEF IS REQUESTED**

Relief is requested for the ASME Section XI, 1992 Edition, Table IWE-2500-1, Examination Category E-G, Item No. E8.10, which requires a VT-1 visual examination of 100% of each bolted connection during the Inspection Interval.

**BASIS FOR RELIEF:**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

This proposed alternative examination method and frequency has been evaluated and we have determined that the implementation of the alternative requirement will provide an acceptable level of quality and safety for the following reasons:

1. A general visual examination of each pressure retaining bolted connection, including its bolts, studs, nuts, washers, etc., will be performed once each Inspection Period (i.e., three examinations in a ten year period). Performing the general visual examination at this frequency would detect and correct potential degradation prior to failure and is considered an enhancement to the current ASME Code, Section XI requirement. The ASME Code, Section XI only requires a visual examination to be performed once during this same time period.
2. The general visual examination is an acceptable examination method for detecting potential degradation of pressure retaining bolting. The general visual examination is not a cursory look at the pressure retaining bolting, but a thorough examination of the exposed surface areas and is performed by qualified and properly trained examiners. If an area is determined to be suspect, a more detailed visual examination to determine the magnitude and extent of the suspect areas will be performed.

**Attachment E, Relief Request CR-30  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 3)**

3. Both the general and/or detailed visual examination will be performed in accordance with procedures that will delineate the controls for ensuring sufficient illumination and resolution for detecting degradation are maintained.
4. The examiners will also be required to successfully complete approved training on the proper techniques for examining Code Class MC items. This level of qualification will ensure the capability and visual acuity of the examiners is sufficient to detect evidence of potential degradation of the pressure retaining bolting.
5. The level of quality and safety will not be decreased by the performance of the general visual examination of the accessible surface areas in place of the VT-1 visual examination. As clarified in paragraph IWA-2211, VT-1 visual examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, and erosion. The VT-1 visual examination requirements were primarily written for the examination of components and items within the reactor coolant pressure boundary (a VT-1 examination is not required for Class 2 and 3 components). The bolted connections associated with primary containment are not subject to the same service conditions (e.g., pressure, temperature, loading) as the bolting within the reactor coolant pressure boundary. These bolted connections, along with their bolting, are also not subject to conditions that could cause accelerated degradation or aging. For these reasons, a VT-1 examination is not warranted.
6. The level of quality and safety will not be decreased by not disassembling the bolted connection and performing the general visual examination of the accessible surface areas. The bolted connections associated with primary containment are also subject to testing in accordance with 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors." Appendix J requires that each of these bolted connections be tested on a routine basis. The purpose of the Appendix J test is to ensure the leak-tight integrity of the primary containment structure. Thus, the visual examination only needs to be performed to evaluate any inservice environmental effects that could adversely affect the performance of the bolted connection that have been adequately assembled and tested. For these reasons, the bolted connection need not be disassembled for the purpose of examination, and only those portions of bolting that are exposed to environmental conditions require examination.
7. The ASME Main Committee and the Board of Nuclear Codes and Standards have also determined that the VT-1 examination of pressure retaining bolting was not appropriate and have approved the rewrite of Subsection IWE which eliminated this requirement. This rewrite of Subsection IWE was published in the 1998 Edition of the ASME Code, Section XI. The alternative examination method and frequency proposed is consistent with the approved rewrite of Subsection IWE.

**Attachment E, Relief Request CR-30  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 3 of 3)**

**PROPOSED ALTERNATIVE PROVISIONS**

During the first containment Inspection Interval, we will perform a general visual examination of each pressure retaining bolted connection once per Inspection Period. The bolted connection will be examined in their "as-found" condition and will not be disassembled for the sole purpose of performing the general visual examination. If an area is determined to be suspect during the general visual examination, we will perform a detailed visual examination to determine the magnitude and extent of the suspect areas. If required, we will disassemble the bolted connection to support the performance of the detailed visual examination.

**APPLICABLE TIME PERIOD**

Relief is requested for the first ten-year interval of the IWE Program for Quad Cities Units 1 and 2.

**REFERENCES:**

1. ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 1992 Edition with 1992 Addenda and 1998 Edition.
2. 10 CFR 50, Appendix J, "Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactor."

**Attachment F, Relief Request PR-11  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 2)**

**COMPONENT IDENTIFICATION**

Code Class: 2  
References: Table IWC-2500-1  
Examination Category: C-H  
Item Number: C7.10, C7.20, C7.30, C7.40, C7.70, C7.80  
Description: Continuous Pressure Monitoring of the Control Rod Drive (CRD) System Accumulators.  
Component Numbers: CRD Accumulators and associate piping

**CODE REQUIREMENT FROM WHICH RELIEF IS REQUESTED**

Relief is requested from the Visual VT-2 examination requirements specified in Table IWC-2500-1 for the nitrogen side of the CRD System Accumulators on the basis that Quad Cities Station Technical Specification Surveillance requirements exceed the code requirement for a Visual VT-2 Examination.

**BASIS FOR RELIEF**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that the proposed alternatives provide an acceptable level of quality and safety.

As required by Quad Cities Station Technical Specifications, the CRD System Accumulator pressure must be greater than or equal to 940 psig to be considered operable. 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 (+/- 15 psig), an alarm is received in the control room. The pressure drop for the associated accumulator is then recorded in the control room log, and the accumulator is recharged by station procedure QCOP 0300-06, "CRD Accumulator Charging." If an accumulator requires charging more than twice in a thirty 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 QCOP 0300-06.

Since monitoring the nitrogen side of the accumulators is continuous, any degradation of 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 would require applying a leak detection solution to 177 accumulators resulting in additional radiation exposure without any added benefit in safety. This inspection would not be consistent with As Low As Reasonably Achievable (ALARA) practices.



**Attachment F, Relief Request PR-11  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 2)**

**PROPOSED ALTERNATE EXAMINATIONS**

As an alternate to the Visual VT-2 examination requirements of Table IWC-2500-1, Quad Cities Station will perform continuous pressure decay monitoring in conjunction with Technical Specification, 4.3.G – Control Rod Scram Accumulators, Surveillance Requirements for the nitrogen side of the CRD Accumulators including attached piping.

**APPLICABLE TIME PERIOD**

Relief is requested for the third ten-year interval of the Inservice Inspection Program for Quad Cities Units 1 and 2.

**Attachment G, Relief Request PR-12  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 1 of 2)**

**COMPONENT IDENTIFICATION**

Code Class:	1, 2, and 3
References:	IWA-5250(a)(2)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Alternative Rules for Corrective Measures if Leakage Occurs at Bolted Connections

**CODE REQUIREMENT FOR WHICH RELIEF IS REQUESTED**

IWA-5250(a)(2) states that if leakage occurs at a bolted connection, the bolting shall be removed, VT-3 visually examined for corrosion, and evaluated in accordance with IWA-3000.

**BASIS FOR RELIEF**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety.

Removal of pressure retaining bolting at mechanical connections for visual, VT-3 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 Code 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. Quad Cities Station considers this requirement to be unnecessarily restrictive.

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:

- Bolting materials
- Service age of joint bolting materials
- Leakage history at connection
- Leakage location
- Visual evidence of corrosion at connection (connection disassembled)
- Corrosiveness of process fluid
- Plant / Industry studies of similar bolting materials in a similar environment.
- Leakage monitoring

**Attachment G, Relief Request PR-12  
for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
(Page 2 of 2)**

**BASIS FOR RELIEF** (Cont)

Removal of bolting may not always be the most prudent course of action. An example at Quad Cities Station is the complete replacement of bolting materials (studs, bolts, nuts, washers, etc) at mechanical joints during plant outages. When the associated system process piping is pressurized during plant start-up, in some cases, leakage is identified at these joints. The root cause of this leakage is often due to thermal expansion of the piping and bolting materials at the joint and subsequent process fluid seepage at the joint gasket. Proper re-torquing of the joint bolting, in most cases, stops the leakage. Removal of any of the joint bolting to evaluate for corrosion would be unwarranted in this situation due to new condition of the bolting materials. ASME Section XI Interpretation XI-1-92-01 has recognized this situation exists, and has clarified that the requirements of IWA-5250(a)(2) do not apply.

**PROPOSED ALTERNATE PROVISIONS**

Quad Cities Station proposes the following alternative methodology to the requirements of IWA-5250(a)(2) which will provide an equivalent level of quality and safety when evaluating leakage and bolting material condition at Class 1, 2, and 3 bolted connections.

As an alternative to the to the requirements of IWA-5250(a)(2), one of the following requirements shall be met for leakage at bolted connections:

- (a) The leakage shall be stopped, and the bolting and component material shall be reviewed for joint integrity.
- (b) If the leakage is not stopped, the joint shall be evaluated in accordance with IWB-3142.4 for joint integrity. This evaluation should include bolting materials, service age of joint bolting materials, leakage history at connection, leakage location, visual evidence of corrosion at connection, corrosiveness of process fluid, plant/industry studies of similar bolting materials in a similar environment and leakage monitoring as detailed in the basis for this relief.

**PROPOSED ALTERNATE PROVISIONS**

If any of the above parameters indicates a need for further examination, the bolt closest to the source of leakage shall be removed, receive a VT-3 examination, and be evaluated in accordance with IWA-3100(a). If the leakage is identified when the bolted connection is in service, and the information in the evaluation is supportive, the removal of the bolt for VT-3 examination may be deferred to the next refueling outage. When the removed bolt has evidence of degradation, all remaining bolting shall be removed, VT-3 examined, and evaluated in accordance with IWA-3100(a).

**APPLICABLE TIME PERIOD**

Relief is requested for the third ten-year interval of the Inservice Inspection Program for Quad Cities Units 1 and 2.

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**COMPONENT IDENTIFICATION**

Code Class:	1, 2, and 3
Reference:	IWA-2300
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Alternative Requirements for Qualification of VT-2 Examination Personnel.
Component Numbers:	Class 1, 2, and 3 Pressure Retaining Components.

**CODE REQUIREMENTS FOR WHICH RELIEF IS REQUESTED**

ASME Section XI, Subarticle IWA-2300 and Paragraph IWA-2312 require personnel performing nondestructive examinations not listed in SNT-TC-1A, "Personnel Qualification and Certification in Nondestructive Testing," to be qualified and certified to a comparable level of qualification as defined in SNT-TC-1A and the Employer's written practice.

**BASIS FOR RELIEF**

In accordance with 10 CFR 50.55a(a)(3)(i) and (ii), relief is requested on the basis the proposed alternatives would provide an acceptable level of quality and safety.

ASME Section XI currently requires personnel conducting VT-2 examinations to be qualified and certified to comparable levels of qualification as defined in SNT-TC-1A and the Employer's written practice. However, unlike the nondestructive testing methods addressed within SNT-TC-1A, or VT-1 and VT-3 examination methods, VT-2 examination does not require any special knowledge of technical principals underlying its performance. It is only the straight forward examination for leakage. No special skills or technical training are required in order to observe leakage from a component or bubbles forming on a joint wetted with leak detection solution. As such, VT-2 personnel should not be subject to the same qualification and certification requirements that were established for nondestructive testing personnel. Code Case N-546 provides more applicable requirements for the qualification and certification of VT-2 inspection personnel.

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for Quad Cities Nuclear Power Station Unit(s) 1 and 2  
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**BASIS FOR RELIEF** (Con't)

Code Case N-546 requires that personnel performing VT-2 visual examinations have at least forty (40) hours of plant walkdown experience, receive a minimum of four (4) hours of training on Section XI requirements, and pass the vision test requirements of IWA-2321, 1995 Edition. This alternative to the existing Code requirements reduces the administrative burden of maintaining an ASME Section XI qualification and certification program for VT-2 examiners and allows for the use of personnel most familiar with the walkdown of plant systems, such as licensed and nonlicensed operators, local leak rate personnel, system engineers, and NDE personnel to perform VT-2 visual inspections. The quality of VT-2 visual examinations will be maintained by using the alternate qualification rules approved by ASME in Code Case N-546.

Code Case N-546 was approved by the ASME Boiler and Pressure Vessel Code Committee on August 24, 1995, but is not yet included in the most recent listing of NRC approved code cases provided in Revision 12 of Regulatory Guide 1.147, "Inservice Inspection Code Case Acceptability - ASME Section XI Division 1".

**PROPOSED ALTERNATE PROVISIONS**

Quad Cities Station will use the provisions of Code Case N-546 as alternatives to the requirements of Section XI, IWA-2300 for qualifying VT-2 visual inspectors.

**APPLICABLE TIME PERIOD**

Relief is requested for the third ten-year interval of the Inservice Inspection Program for Quad Cities Units 1 and 2.