FPL Energy Duane Arnold, LLC
3277 DAEC Road
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FPLEnergy.

Duane Arnold Energy Center

June 30, 2006

NG-06-0439
10 CFR 50.55a

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Duane Arnold Energy Center
Docket No: 50-331
Op. License No: DPR-49

Fourth 10-Year Inservice Inspection Plan

The Duane Arnold Energy Center (DAEC) will begin the fourth inservice inspection (ISI) interval on November 1, 2006. This interval will continue through February 21, 2014. FPL Energy Duane Arnold, LLC herewith submits the DAEC ISI plan for this upcoming interval (Attachment).

We intend to use this plan to perform inservice inspections during the upcoming refueling outage (RFO) 20. As discussed in the plan, FPL Energy Duane Arnold, LLC is requesting approval of several relief requests. The relief requests are contained in Section H of the Attachment. Approval is requested as follows:

- NDE-R003 was approved as NDE-R047 by the NRC in a safety evaluation dated January 6, 2005. That approval included relief through the fourth 10-year interval, therefore, FPL Energy Duane Arnold, LLC is not requesting NRC approval of this relief request. The relief request is included in Section H for completeness. Please note that the contents of NDE-R003 have been editorially updated to reflect the current edition and addenda of Section XI of the fourth 10-year interval ISI Program and to reflect the fact that DAEC is no longer operated by Nuclear Management Company, LLC.
- Relief requests NDE-R001, NDE-R002, NDE-R005, NDE-R006 and NDE-R008 support inspections planned for RFO 20 currently scheduled for February, 2007. Therefore, approval of these relief requests by January 30, 2007 is requested.
- FPL Energy Duane Arnold, LLC requests approval of the remaining two relief requests by October 1, 2007.

A047

Should you have any questions regarding this matter, please contact Steve Catron, Duane Arnold Energy Center Licensing Manager, at (319) 851-7234.

Gary Van Middlesworth

Gary Van Middlesworth
Vice President, Duane Arnold Energy Center
FPL Energy Duane Arnold, LLC

Attachment: DAEC Fourth Interval Inservice Inspection Plan for Duane Arnold Energy Center, Palo, IA

cc: Administrator, Region III, USNRC
Project Manager, DAEC, USNRC
Senior Resident Inspector, DAEC, USNRC

DAEC Fourth Interval

Inservice Inspection Plan

FOR
DUANE ARNOLD ENERGY CENTER
PALO, IA

This procedure applies to Safety-Related equipment.

Usage Level
REFERENCE USE

Effective Date:

TECHNICAL REVIEW			
Prepared by:	Russell A Ripman ISI Administrator	Date:	6/28/06
Verified by:	[illegible] Peer Review	Date:	6/28/06
Reviewed by:	Bruce Kat Inspection and Material Supervisor	Date:	6-28-06

PROCEDURE APPROVAL			
Approved by Procedure Owner:	[illegible] Program Engineering Manager	Date:	6/28/06
Reviewed by:	JBL Authorized Nuclear Inservice Inspector	Date:	6-28-06

TABLE OF CONTENTS

1.0 INTRODUCTION

(1) This Inservice Inspection Plan outlines the requirements for the Non-Destructive Examination of Class 1, 2, and 3 pressure retaining components and their supports at Duane Arnold Energy Center (DAEC).

(2) This Inservice Inspection Plan will be effective from November 1, 2006 through and including February 21, 2014, which represents the fourth ten-year interval of the Inservice Inspection Program for DAEC. The interval ends on February 21, 2014 to coincide with the end of the current license, thus making the interval duration of approximately 99 months as opposed to the standard 120 months.

(3) The key features of this Plan are the Introduction and Plan Description, Relief Requests, Technical Approach and Positions, and Summary Tables. The details of the Inservice Inspection Program are addressed in other documents that are available at DAEC. These documents include, but are not limited to, component detail drawings, piping and instrumentation diagrams, piping isometric drawings, a component listing of each weld, valve, support, etc., procedures, calibration blocks, schedules, and other records required to define and execute the Inservice Inspection Plan at the DAEC.

2.0 BASIS OF INSERVICE INSPECTION PLAN

(1) The commercial operation date for Duane Arnold Energy Center is February 1, 1975. The end of the first interval was extended from February 1, 1985, to October 31, 1985, due to a recirculation inlet nozzle safe-end replacement outage that lasted from June 17, 1978 through March 10, 1979. The extended interval is consistent with the ASME B&PV Code Section XI, Paragraph IWA-2400(c) of the ASME Section XI, 1980 Edition with the 1981 Addenda, and IES letters dated December 13, 1983 (NG-83-4036) and January 24, 1984 (NG-84-0213). The end of the second interval was originally scheduled for November 1, 1995. The second interval was also extended 1 year, as permitted by IWA-2430(d) of the ASME Section XI 1989 Edition and the revised rule making of 10CFR50.55a(g)(6)(A)(3)(v). The end of the second interval was extended into the third inspection interval, up to the end of refueling outage (RFO-14) scheduled for October 1996. The third period interval started November 1, 1996 and is scheduled to end October 31, 2006.

(2) The three inspection periods during the fourth inspection interval are as follows:

First Period: November 1, 2006 - October 31, 2009 (36 Months)
Second Period: November 1, 2009 – October 31, 2012 (36 Months)
Third Period: November 1, 2012 - February 21, 2014 (27 Months)

(3) This Plan was developed in accordance with the requirements delineated in 10 CFR 50.55a and the 2001 Edition through 2003 Addenda of the American Society of Mechanical Engineers (ASME)

Boiler and Pressure Vessel Code, Section XI, Subsections IWA, IWB, IWC, IWD, and IWF for Inspection Program B.

(a) Class 1 Category B-F and B-J, and Class 2 Category C-F-2 welds will be selected utilizing the risk-informed process described by EPRI TR-112657B-A (ref. Relief Request NDE-R005).

(b) An ISI Plan per Subsections IWE and IWL is not included in this submittal.

(c) As allowed by USNRC Regulatory Guide 1.147, Revision 14, certain ASME Section XI Code Cases have been determined to be acceptable for application to ISI Programs. The following Code Cases are being adopted by DAEC and incorporated in the 4th interval ISI Program Plan. Additionally, DAEC wishes to use Code Cases not currently identified in Regulatory Guide 1.147 and has requested relief to use specific Code Cases, where the relief is identified in Section H.

Code Case	Title	Governing Document	Limitations
N-460	*Alternative Examination Coverage for Class 1 and Class 2 Welds, Section XI Division 1*	*RG 1.147 Rev. 14*	*None*
N-513-1	*Evaluation Criteria or Temporary. Acceptance of Flaws in Class 3 Piping Section XI Division 1*	*RG 1.147 Rev. 14*	*(1) Specific safety factors in paragraph 4.0 must be satisfied. (2) Code Case N-513 may not be applied to: (a) Components other than pipe and tube (b) Leakage through a gasket (c) Threaded connections employing nonstructural seal welds for leakage prevention (through seal weld leakage is not a structural flaw; thread integrity must be maintained). (d) Degraded socket welds.*
N-526	*Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-532-1	*Alternative Requirements to Repair and Replacement Documentation Requirements and Inservice Summary Report Preparation and Submission as Required by IWA-4000 I and IWA-6000, Section XI Division 1*	*RG 1.147 Rev. 14*	*Code Case N-532-1 requires an Owner's OAR- 1 to be prepared and certified upon completion of each refueling outage. The OAR forms must be submitted to the NRC within 90 days of the completion of the refueling outage.*

N-537	*Location of Ultrasonic Depth-Sizing Flaws, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-545	*Alternative Requirements for Conduct of Performance Demonstration Detection Test of Reactor Vessel, Section XI*	*RG 1.147 Rev. 14*	*None*
N-552	*Alternative Methods – Qualification for Nozzle Inside Radius Section from the Outside Surface, Section XI, Division 1*	*RG 1.147 Rev. 14*	*To achieve consistency with the 10 CFR 50.55a rule change published September 22, 1999 (64 FR 51370), incorporating Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," to Section XI, add the following to the specimen requirements: "At least 50 percent of the flaws in the demonstration test set must be cracks and the maximum misorientation must be demonstrated with cracks. Flaws in nozzles with bore diameters equal to or less than 4 inches may be notches. Add to detection criteria, "The number of false calls must not exceed three."*

N-597-1	*Requirements for Analytical Evaluation of Pipe Wall Thinning, Section XI, Division 1*	*RG 1.147 Rev. 14*	*1) Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, April 1999, "Recommendation for Effective Flow Accelerated Corrosion Program," for developing the inspection requirements, the method of predicting the rate of wall thickness loss, and the value of the predicted remaining wall thickness. As used in NSAC-202L-R2, the terms "should" and "shall" have the same expectation of being completed.* *2) Components affected by flow accelerated corrosion to which this Code Case are applied must be repaired or replaced in accordance with the construction code of record and Owner's requirements or a later NRC-approved edition of Section III of the ASME Code prior to the value of tp reaching the allowable minimum wall thickness tmin, as specified in -3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to NRC review and approval.* *3)For Class 1 piping not meeting the criteria of -3221, the use of evaluation methods and criteria is subject to NRC review and approval.* *4) For those components that do not require immediate repair or replacement, the rate of wall thickness loss is to be used to determine a suitable inspection frequency so that repair or replacement occurs prior to reaching allowable minimum wall thickness tmin.* *5) For corrosion phenomena other than flow-accelerated corrosion, use of the Code Case is subject to NRC review and approval. Inspection plans and wall thinning rates may be difficult to justify for certain degradation mechanisms such as MIC and pitting.*

N-606-1	*"Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique"*	*RG 1.147 Rev. 14*	*Prior to welding, an examination or verification must be performed to ensure proper preparation of the base metal, and that the surface is properly contoured so that an acceptable weld can be produced. The surfaces to be welded, and surfaces adjacent to the weld, are to be free from contaminants, such as rust, moisture, grease, and other foreign material or any other condition that would prevent proper welding and adversely affect the quality or strength of the weld. This verification is to be required in the welding procedures.*
N-613-1	*Alternative requirements for welding procedure qualification*	*RG 1.147 Rev. 14*	*None*
N-624	*Successive Inspections, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-629	*Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-639	*Alternative Calibration Block Material, Section XI, Division 1*	*RG 1.147 Rev. 14*	*Chemical ranges of the calibration block may vary from the materials specification if (1) it is within the chemical range of the component specification to be inspected, and (2) the phase and grain shape are maintained in the same ranges produced by the thermal process required by the material specification.*
N-658	*Qualification Requirements for Ultrasonic Examination of Wrought Austenitic Piping Weld, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-661	*Alternate methods to restore wall thickness for Class 2 and 3 Raw Water applications*	*RG 1.147 Rev. 14*	*(a) If the root cause of the degradation has not been determined, the repair is only acceptable for one cycle.* *(b) Weld overlay repair of an area can only be performed once in the same location.* *(c) When through-wall repairs are made by welding on surfaces that are wet or exposed to water, the weld overlay repair is only acceptable until the next refueling outage.*

N-686	*Alternate requirements for visual examinations, VT-1, VT-2, and VT-3 Section XI.*	*Relief Request*	
N-695	*Qualification Requiremens for Dissimilar Metal Piping Welds, Section XI, Division 1*	*RG 1.147 Rev. 14*	*None*
N-700	*Alternative for selecting vessel attachment welds.*	*Relief Request*	

3.0 SYSTEM CLASSIFICATION

(1) Per IWA-1400(a) of the 2001 Edition through the 2003 Addenda of Section XI, it is the owner's responsibility to determine the appropriate Code Classes for each component and to identify the system boundaries subject to inspection. IWA-1300 states that components identified for inspection and testing shall be included in the inservice inspection plan, and that the selection of components for the inservice inspection plan is subject to review by the regulatory and enforcement authorities having jurisdiction at the plant site. IWA-1320(a) states that the system group classification criteria of the regulatory authorities having jurisdiction at the power plant site governs the application of the rules of Section XI. IWA-1400(a), footnote 1, states that classification criteria are specified in 10CFR50. This reference is to footnote 9 of 10 CFR50.55a which specifies that Regulatory Guide 1.26 and Section 3.2.2 of NUREG-0800 may be used for this purpose. Section 3.2.2 of NUREG-0800 allows the use of either the NRC Group Classification system of Regulatory Guide 1.26 or the ANS Safety Classification system (referring to the method described in ANSI/ANS-52.1-1983) which can be cross-referenced to Regulatory Guide 1.26.

(2) The component classifications of the ASME Code (Class 1, 2, or 3) determine the rules and requirements for inspection and testing and define the Section XI examination boundaries. Because early vintage nuclear plants were designed and constructed before Section III of the ASME Boiler and Pressure Vessel Code was incorporated into 10CFR50.55a, the ASME Section XI Code classifications for ISI may differ from the original design classifications. Therefore, while the ASME Code classifications determine the rules for repairs and replacements and the component inspection requirements, all repairs and replacements are performed to meet, at a minimum, the specifications of the original design code.

(3) Historically, the safety-related classification process and criteria have not been clearly defined. Various documents used in this process have alluded to such phrases as "safety-related" or "important to safety" but no complete, consistent guideline existed as to why some equipment is more important to nuclear safety than other equipment or what documents are applicable. As a result, various interpretations/inconsistencies have evolved in the use of the term "safety-related", often times confusing regulatory and other non-functional requirements as to its applicability.

(4) Other phrases widely used in codes, standards, and other documents have also been correctly and incorrectly interpreted to be synonymous to "safety-related". "Basic component" defined in 10CFR21 is equivalent to "safety-related".

(5) The Updated Final Safety Analysis Report (UFSAR) uses the term "safety" in a broader context than "safety-related". The UFSAR uses phrases such as "safety functions", "nuclear safety systems", "instruments required for safety" and others. The relationship of the term "safety-related" to those other commonly referred to terms such as "safety", "protection systems" etc. is not necessarily synonymous with the term "safety-related".

(6) There also exists further confusion regarding the term "safety-related". This confusion results from the different uses and interpretations applied to this term. The term safety-related is typically used in the following ways:

(a) From a design engineering standpoint, the term "safety-related" is used to identify items which are (1) part of the reactor coolant pressure boundary, (2) required to shut down the reactor and maintain it in a safe shutdown condition, or (3) required to prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to 10CFR100.11 guidelines.

(b) Typically, three methods of procurement are utilized, commonly referred to as: safety-related, commercial grade, and non-safety-related. A safety-related procurement refers to the purchase of an item under the provisions of 10CFR21 from a vendor with a quality assurance program that meets the requirements of 10CFR50 Appendix B. A commercial grade procurement refers to an item which will be dedicated for safety-related use, but is not purchased to an approved 10CFR50 Appendix B Quality Assurance program nor are 10CFR21 requirements imposed on the vendor. Once a commercial grade item is dedicated it becomes a basic component. A non-safety-related procurement refers to an item which does not have a safety-related function.

(c) Also, selected items may be classified as safety-related even though their function is non-safety-related. This is done to institute greater controls over procurement, maintenance, or replacement of such items.

(d) As a result, it is important to understand the context in which the term "safety-related" is used and what is meant. For this document the term "safety-related" pertains to the function a system or component performs.

(7) The NRC issued the construction permit for the Duane Arnold Energy Center (DAEC) in June 1968. The plant design was completed when IES Utilities Inc. (IES) applied for an operating license for DAEC and submitted the Final Safety Analysis Report (FSAR) for the facility to the NRC in March 1971. This license was issued by the NRC in January 1974. The United States of America Standards (USAS) used for the original design and construction of DAEC were B31.1 (1967), Code for Power Piping, and B31.7 (1969 edition with 1970/1971 addenda), Code for Nuclear Power Piping. The "General Design Criteria for Nuclear Power Plant Construction Permits" was published for comment in the Federal Register in July 1967. The final version of these design criteria was not incorporated into the Code of Federal Regulations (10CFR50, Appendix A) until February 1971, approximately the same time that the FSAR was submitted to the NRC. The FPL Energy license for DAEC is based, in part, on design and construction of the plant to USAS B31.1, USAS B31.7, and the interpretation of the intent of the Draft General Design Criteria published in July 1967.

The piping and pressure retaining components of all DAEC systems were both functionally and seismically classified according to service and location prior to construction by Bechtel (Architect Engineer) and/or General Electric (the plant Engineer-Constructor). These design classifications are as follows:

Quality Group A - Piping and equipment pressure parts within the reactor coolant pressure boundary through the outer most isolation valves, inclusive.

Quality Group B - Piping and equipment pressure parts downstream of the outer most isolation valves, extensions of the containment, and the emergency core cooling system.

Quality Group C - Auxiliaries to the emergency core cooling system or radioactive waste process piping and equipment pressure parts, excluding power generation systems.

Quality Group D - Balance of plant piping and equipment pressure parts, including power generation systems. Certain piping in this group such as Off Gas, Well Water and portions of the Fuel Pool Cooling is designated "non-critical" . This "non-critical" piping is identified by the "D" designation.

Quality Group D+QA - Balance of plant piping and equipment pressure parts, including power generation systems. Certain piping in this group such as feedwater and main steam outside of the containment are designated "B or D" to establish a category for added quality controls. QA Level 1 or 2, for B designation, and Level 3 or 4 for D designated quality group D+QA piping.

(8) The current ASME Code component classifications did not exist at the time of DAEC design and construction. The ASME Code Class 1, 2, and 3 designations were added and defined in more recent editions of the ASME Boiler and Pressure Vessel Code. The scope of earlier editions of the ASME Code was limited to systems and portions of systems that comprised the reactor coolant pressure boundary. Hence the unique wording of 10CFR50.55a(g)(1) for nuclear power facilities whose construction permit was issued prior to January 1, 1971:

> "Components which are part of the reactor coolant pressure boundary and their supports must meet the requirements applicable to components which are classified as ASME Code Class 1. Other safety-related pressure vessels, piping, pumps and valves must meet the requirements applicable to components which are classified as ASME Code Class 2 or Class 3."

(9) The initial DAEC ISI program was based on the 1970 Edition of Section XI. This program was submitted as part of the original FSAR (Appendix J), which was accepted by the NRC. However, the inspection rules and requirements of the 1970 Edition of Section XI were minimal and have changed significantly since then. Federal regulations require that ISI programs be updated, to the extent practical, to comply with the inspection and testing requirements of the edition and addenda of the ASME Code incorporated by reference in 10CFR50.55a one year prior to the start of each ten-year inspection interval.

(10) During subsequent revisions of the ISI program, other safety-related systems were added to the ISI program and ASME Code Class designations were assigned to establish the examination boundaries and define the required inspections and tests for the associated components. Systems, or portions of systems, were considered safety-related if they were determined to mitigate the consequences of an accident based on the analyses contained in Section 15 of the UFSAR. Although the General Electric Design Classifications do not directly correlate to ASME Code Class 1, 2, and 3, and NRC Quality Groups A, B, C, and D of Regulatory Guide 1.26, they were used as the basis for establishing the ASME Section XI examination boundaries. For the purposes of ISI, the DAEC Safety Class (SC) I safety-related components were designated ASME Section XI Code Class 1, the SC II safety-related components were designated ASME Section XI Code Class 2, and the SC III safety-related components were designated ASME Section XI Code Class 3. The DAEC D+QA systems, including both safety-related and nonsafety-related systems, (except the Main Steam lines outside MSIV to Stop valves, and portions of the Emergency Service Water piping) were generally designated Non-Code Class.

(11) Because DAEC was designed and constructed prior to the issuance of Regulatory Guide 1.26 (safety guide 26) and NUREG-0800, these documents were not used to establish the original Section XI examination boundaries, however, in accordance with the requirements of ASME Section XI 1974 Summer 75 IWA-1000 footnote 2, these guidance documents were used during the first ten year ISI program update. DAEC has formally committed to the use of either Regulatory Guide 1.26 or NUREG-0800, Section 3.2.2. The DAEC ISI program for the Fourth ten-year inspection interval will continue to employ Regulatory Guide 1.26, NUREG-0800 and other approved American Nuclear Society guidance documents to determine the applicability of component inspections and to determine examination boundaries. DAEC UFSAR was used for guidance and provides the basis for establishing the applicable system safety classifications contained in this document.

4.0 AUGMENTED INSERVICE INSPECTION REQUIREMENTS

(1) The following augmented inservice inspection requirements are being implemented under a separate program not included in this submittal. DAEC's augmented inspection program is implemented in accordance with the latest licensing agreements pertaining to these requirements:

(a) Generic Letter 88-01, NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping or BWRVIP-75 "Technical Basis for Revision to Generic Letter 88-01 Inspection Schedules (Ref. AR 19005 and 23055)

(b) Generic Letter 94-03, Intergranular Stress Corrosion Cracking of Core Shrouds In Boiling Water Reactors or BWRVIP-76 "BWR Core Shroud Inspection and Flaw Evaluation Guidelines"

(c) NUREG 0313, Rev. 2, Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping.

(d) NUREG 0619, BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking.

(e) USNRC Regulatory Guide 1.150, Revision 1, Examination of Reactor Pressure Vessel Welds during Preservice and Inservice Inspection.

(f) NRC IE Bulletin 80-13, Cracking in Core Spray Spargers. These examinations have been upgraded to implement the recommended inspections contained in BWRVIP-18.

(g) Generic Letter 87-11, Relaxation in Arbitrary Intermediate Pipe Rupture Requirements.

5.0 REPAIR AND REPLACEMENT PROGRAM REQUIREMENTS

(1) The DAEC Repair/Replacement program requirements are being implemented in accordance with the 2001 Edition of ASME Section XI, as amended by the 2003 Addenda and the latest 10 CFR 50.55(a) requirements. The ASME Section XI Repair/Replacement program for DAEC will be administered in accordance with the aforementioned rules and maintained in accordance with IWA-1400, IWA-6200 and available for review on-site.

6.0 SNUBBER TESTING PROGRAM REQUIREMENTS

(1) The DAEC Snubber Examination/Testing Program requirements are being implemented under separate program document not included in this submittal. DAEC's Snubber Examination/Testing Program is implemented in accordance with the DAEC Technical Requirements Manual (Reference NDE-R001). This Program will be administered in accordance with the aforementioned rules and available for review on-site.

7.0 CONTENTS OF INSERVICE INSPECTION PLAN

(1) The Inservice Inspection Plan addresses the requirements for inservice inspection of components and system pressure testing separately, although some Sections of the Plan are common to both. The applicability of each of the Sections identified in this Plan are as follows:

Inservice Inspection- Sections B, E, F, G, H, and I

(2) Cover and Table of Contents

Provides the organizational format and revision status for the Inservice Inspection Plan.

(3) Section A - Introduction and Plan Description

Provides details on the scope, basis and contents of the Inservice Inspection Plan, system classifications, and augmented inservice inspection requirements.

(4) Section B - Application of Exemption Criteria

Provides the basis for determining the Class 1, 2, and 3 exempted components from surface and volumetric examination requirements per IWB, IWC, and IWD-1200.

(5) Section C - List of Applicable Piping and Instrumentation Diagrams (P&IDs)

Provides a listing of P&IDs corresponding to each system that contains components subject to examination under this Plan.

(6) Section D - List of Applicable Piping Isometric Drawings

Provides a listing of piping isometric drawings corresponding to each system that contains components subject to volumetric, surface, VT-1, or VT-3 examinations under this Plan.

(7) Section E - List of Applicable Calibration Standards Drawings

Provides a listing of ultrasonic calibration block standards currently available for performance of volumetric examinations under this Plan.

(8) Section F - Inservice Inspection Summary Table

The DAEC Inservice Inspection Summary Table provides the following information:

(a) Examination Category

Provides the examination category as identified in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWF-2500-1, and identification of specific Code Cases being implemented. Only those examination categories applicable to DAEC are identified.

(b) Item Number and Item Description

Provides the item number and description as defined in ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, IWF-2500-1, and identification of specific Code Cases being implemented. Only those item numbers applicable to DAEC are identified. The following Item Numbers do not apply to the DAEC:

Class 1

B2.10, B2.11, B2.20, B2.21, B2.22, B2.30, B2.31, B2.32, B2.40, B2.50, B2.51, B2.52, B2.60, B2.70, B2.80, B3.10, B3.20, B3.30, B3.50, B3.70, B3.80, B3.110, B3.130,

B3.150, B3.160, B5.40, B5.50, B5.60, B5.70, B5.80, B5.90, B5.100, B5.110, B5.120, B6.60, B6.70, B6.80, B6.90, B6.100, B6.110, B6.120, B6.130, B6.140, B6.150, B6.160, B6.170, B6.210, B6.220, B6.230, B7.20, B7.30, B7.40, B9.22, B10.40, B12.10, B12.30, B12.40, B13.50, B13.60, B13.70, 16.10, and B16.20

Class 2

C1.30, C2.10, C2.11, C2.20, C2.30, C2.31, C2.32, C2.33, C3.30, C3.40, C4.10, C4.20, C4.30, C4.40, C5.10, C5.11, C5.20, C5.21, C5.30, C5.40, C5.41, C5.60. C5.61, C5.70, C6.10, and C6.20

Class 3

D1.10, D1.30, and D1.40

(c) Total Number of Components

Provides the total population of components potentially subject to examination. The number of components actually examined during the inspection interval will be as indicated in Section I, based upon the Code requirements for the subject item number .

(d) Exam Requirements

Provides the examination method(s) required by ASME Section XI, Tables IWB-2500-1, IWC-2500-1, IWD-2500-1, and IWF-2500-1.

(e) Relief Request

Provides a listing of relief requests or alternative examinations applicable to the item number.

(f) Technical Approach and Position

Provides a listing of technical approach and positions applicable to the item number

(9) Section G - Inservice Inspection Technical Approach and Positions

When the requirements of ASME Section XI are not easily interpreted, DAEC has reviewed general licensing/regulatory requirements and industry practice to determine a practical method of implementing the Code requirements. The technical approach and position documents contained in this section have been provided to clarify DAEC's implementation of ASME Section XI requirements for inservice inspection.

(10) Section H - Inservice Inspection Relief Requests

This section contains relief requests written in accordance with 10 CFR 50.55a(a)(3)(i), (a)(3)(ii), or (g)(6)(i) when specific ASME Section XI requirements for inservice inspection are considered impractical or an alternative is identified. The enclosed relief requests are subject to change throughout the inspection interval. If examination requirements are determined to be impractical during the course of the interval, additional or modified relief requests will be submitted in accordance with 10 CFR 50.55a.

(11) Section I - Component Examination Summary Listing

This section contains the tables and schedule for selection and examination of components in accordance with the requirements of ASME Section XI.

1.0 APPLICATION OF EXEMPTION CRITERIA

(1) Section XI Class 1 Exemptions:

(a) Subparagraph IWB-1220(a) gives specific guidance permitting exemption of components from the volumetric and surface examination requirements of IWB-2500 if they are connected to the reactor coolant system (RCS) and are part of the reactor coolant pressure boundary, and are of such a size and shape so that upon postulated rupture, the resulting flow of coolant from the RCS under normal plant operating conditions is within the capacity of makeup systems which are operable from on-site emergency power.

DAEC requested General Electric (GE) perform an analysis to determine the applicability of IWB-1220(a) and identify those systems and piping line sizes that could be exempted. This analysis was performed by GE document 22A2750, and results documented in section 5.2.5.3.3 to the Updated Final Safety Analysis Report.

The calculation identifies and provides that those portions of steam piping with an inside diameter of 2.24 inches and water piping with an inside diameter of 1.12 inches may be exempted from the surface and volumetric examination requirements of Table IWB-2500-1. The systems credited in this calculation with providing normal makeup are the Reactor Core Isolation Cooling (RCIC) and Control Rod Drive (CRD) systems.

In determining the size of the water and steam lines excluded from surface and volumetric examination, water lines were defined as those which penetrate the reactor pressure vessel (RPV) below the normal water level and steam lines as those which penetrate the RPV above the normal water level.

The reactor coolant makeup system consists of the following system(s):

System	Pump Flow Rate	Maximum Fluid Temp.	Emergency Power
CRD System	42 GPM	140^{o} F	Yes, On-site
RCIC System	425 GPM	140^{o} F	Yes, On-site

Water flow rates from a liquid line break are taken as 8000 lbs/sec/ft^2 at 1000 psi. Steam flow rates from a steam line are taken as 2000 lbs/sec/ft^2 at 1000 psi. Make-up water weighs 8.33 lbs per gallon at 70^{o} F. On this basis, the exclusion diameters based on reactor coolant make-up system capacity are as follows:

$$D_W = \frac{\sqrt{M_{70} \left[\frac{V_{70}}{V_{140}} \right]}}{17.8}$$

$$D_s = 2 \quad D_w$$

Dw= Inside diameter of piping containing water which may be exempted from examination

Ds= Inside diameter of piping containing steam which may be exempted from examination

m= Total make-up flow rate of water in gallons per minute.

Using RCIC as the minimum make-up flow.

$$\mathbf{Dw} = \frac{\sqrt{400}}{17.8} = 1.12\text{" water}$$

$$\mathbf{Ds} = 2 \times 1.12 = 2.24\text{" steam}$$

(b) Piping that is NPS 1 and smaller, and the components and connections in piping that is NPS 1 and smaller, are exempt from the volumetric and surface examination requirements of IWB-2500 per IWB-1220(b).

(c) The supports connected to components which are exempt from examination under IWB-1220 are also exempt from the examination requirements of IWF-2500 and Table IWF-2500-1.

(d) The integral attachments of supports connected to components which are exempt from examination under IWB-1220 are also exempt from the examination requirements of IWB-2500 and Table IWB-2500-1.

(e) The above exemptions effect portions of the following systems:

System	P&ID
Nuclear Boiler	M-114
Rx Vessel Instrumentation	M-115
Rx Recirculation	M-116
CRD Hydraulic	M-117,M-118
Residual Heat Removal	M-119,M-120
Core Spray	M-121
HPCI	M-122,M-123
RCIC	M-124,M-125
Standby Liquid Control	M-126
Reactor Water Cleanup	M-127
MSIV Leakage Control	M-184

(2) Section XI Class 2 Exemptions

(a) Components Within RHR, ECC, and CHR Systems (or Portions of Systems)

(i) Vessels, piping, pumps, valves and other components that are NPS 4 and smaller are exempt from the volumetric and surface examination requirements of IWC-2500 per IWC-1221(a).

(ii) Piping and other components of any size beyond the last shutoff valve in open ended portions of systems that do not contain water during normal plant operations are exempt from the volumetric and surface examination requirements of IWC-2500 per IWC-1221(d).

(iii) The supports connected to components which are exempt from examination under IWC-1220 are also exempt from the examination requirements of IWF-2500 and Table IWF-2500-1.

(iv) The integral attachments of supports connected to components which are exempt from examination under IWC-1220 are also exempt from the examination requirements of IWC-2500 and Table IWC-2500-1.

(b) Components Within Systems (Or Portions of Systems) Other than RHR, ECC, and CHR Systems

(i) Vessels, piping, pumps, valves and other components that are NPS 4 and smaller are exempt from the volumetric and surface examination requirements of IWC-2500 per IWC-1222(a).

(ii) Piping and other components of any size beyond the last shutoff valve in open ended portions of systems that do not contain water during normal plant operations are exempt from the volumetric and surface examination requirements of IWC-2500 per IWC-1222(d).

(iii) Vessels, piping, pumps, valves and other components of any size in systems or portions of systems that operate (when system function is required) at a pressure less than or equal to 275 psig and at a temperature less than or equal to 200°F are exempt from the surface and volumetric examination requirements of IWC-2500 per IWC-1222(c).

(iv) The supports connected to components which are exempt from examination under IWC-1220 are also exempt from the examination requirements of IWF-2500 and Table IWF-2500-1.

(v) The integral attachments of supports connected to components which are exempt from examination under IWC-1220 are also exempt from the examination requirements of IWC-2500 and Table IWC-2500-1.

(vi) Piping support members and piping support components that are encased in concrete shall be exempted from the surface examination requirements of IWC-2500 per IWC-1223.

(3) The above exemptions effect portions of the following systems:

System	P&ID
Main Steam	M-103
Turbine Seal	M-104
Condensate Demineralizer	M-109
Rx Building Cooling Water	M-112
Residual Heat Removal	M-113,M-119,M-120
Nuclear Boiler	M-114
Rx Vessel Instrumentation	M-115
CRD Hydraulic	M-118
Core Spray	M-121
HPCI	M-122,M-123
RCIC	M-124,M-125
Standby Liquid Control	M-126
Compressed Air	M-130
Fuel Pool Cooling & Cleanup	M-134
Radwaste Sump	M-137
Containment Atmosphere Control	M-143

Turbine Bldg Sample	M-147
Drywell Cooling	M-157
Aux. Heating Boiler & Main Loop	M-160
Containment Atmosphere Monitoring	M-181
MSIV Leakage Control	M-184
Radwaste Liquid Waste	M-186
Post Accident Sampling	M-187

(4) Section XI Class 3 Exemptions:

(a) The integral attachments of supports and restraints to components that are NPS 4 and smaller within the system boundaries of Examination Categories D-A, and D-B of Table IWD-2500-1 will be exempted from visual examination (VT-1).

(b) The integral attachments of supports and restraints to components that exceed NPS 4 will be exempted from visual examination (VT-1) of Table IWD-2500-1 provided the components are located within systems (or portions of systems) whose function is not required in support of RHR, ECC, and CHR systems and the components operate at a pressure of 275 psig or less and at a temperature of 200°F or less.

(c) The supports connected to components which are exempt from examination under IWD-1220 are also exempt from the examination requirements of IWF-2500 and Table IWF-2500-1.

(d) The above exemptions effect portions of the following systems:

System	P&ID
RHR Service Water	M-113
Emergency Service Water	M-113
Nuclear Boiler	M-114
Residual Heat Removal	M-119,M-120
River Water	M-129
Diesel Generator	M-132
Fuel Pool Cooling	M-134
Circulation Water	M-142
Well Cooling Water	M-144
Service Water Pumphouse	M-146
Control Bldg Cooling	M-169
Rx Bldg HVAC	M-171
Air Flow Standby Filter Unit Control	M-173

(1) LIST OF APPLICABLE PIPING AND INSTRUMENTATION DIAGRAMS (P&IDS)

P & ID #	DRAWING TITLE	ASME CLASS
M-100	LEGEND	N/A
M-101	LEGEND	N/A
M-102	LEGEND	N/A
M-103	MAIN STEAM TURBINE STOP & CONTROL VALVES SH 1	2 & AUG
M-104	TURBINE STEAM SEAL SH 1	2
M-105	STEAM AIR EJECTOR	AUG
M-106	CONDENSATE FEEDWATER SH1	NONCLASS
M-107	CONDENSATE FEEDWATER SH2	NONCLASS
M-108	CONDENSATE DEMINERALIZER	NONCLASS
M-109	CONDENSATE DEMINERALIZER	2 & AUG
M-110	MAKE-UP DEMINERALIZER	NONCLASS
M-111	GENERAL SERVICE WATER	NONCLASS
M-112	REACTOR BUILDING COOLING WATER	2
M-113	RHR SERVICE WATER & EMERG. SERVICE WATER	2, 3
M-114	NUCLEAR BOILER	1, 2 & 3 (AUG)
M-115	REACTOR VESSEL INSTRUMENTATION	1 & 2
M-116	REACTOR RECIRCULATION	1
M-117	CRD HYDRAULIC, SH 1	1
M-118	CRD HYDRAULIC, SH 2	1, 2
M-119	RESIDUAL HEAT REMOVAL	1, 2 & 3
M-120	RESIDUAL HEAT REMOVAL	1, 2 & 3
M-121	CORE SPRAY	1 & 2
M-122	HIGH PRESSURE COOLANT INJECTION SH 1	1, 2 & AUG
M-123	HIGH PRESSURE COOLANT INJECTION SH 2	1, 2, AUG
M-124	REACTOR CORE ISOLATION COOLING SH 1	1, 2 & AUG
M-125	REACTOR CORE ISOLATION COOLING SH 2	1 & 2
M-126	STANDBY LIQUID CONTROL	1 & 2
M-127	REACTOR WATER CLEANUP	1, AUG
M-128	REACTOR WATER FILTER DEMINERALIZER	NONCLASS
M-129	RIVER WATER SUPPLY & INTAKE STRUCTURE	3
M-130	COMPRESSED AIR Sheets 1 - 8 & 10	NONCLASS
M-130	COMPRESSED AIR Sheets 9	2
M-131	TURBINE LUBE OIL	NONCLASS
M-132	DIESEL GENERATOR SH's 1, 2, & 3	3
M-133	FIRE PROTECTION	NONCLASS
M-134	FUEL POOL COOLING & CLEANUP	2 & 3
M-135	FUEL POOL DEMINERALIZER	NONCLASS
M-136	SERVICE CONDENSATE	NONCLASS
M-137	RADWASTE SUMP SYSTEM Sheet 1	2
M-138	EQUIPMENT RADWASTE	NONCLASS
M-139	FLOOR DRAIN RADWASTE	NONCLASS
M-140	RADWASTE SOLIDS HANDLING	NONCLASS
M-141	OFF GAS	NONCLASS
M-142	CIRCULATION WATER	3

(2) LIST OF APPLICABLE PIPING AND INSTRUMENTATION DIAGRAMS Cont.

P&ID #	DRAWING TITLE	ASME CLASS
M-143	CONTAINMENT ATMOSPHERE CONTROL SH 1, 2, 3, and 4	2
M-144	WELL COOLING WATER SH 1	3
M-144	PRODUCTION WELL 1, 2, 3, AND 4	NONCLASS
M-145	MISC. TURBINE GENERATOR	NONCLASS
M-146	SERVICE WATER PUMPHOUSE	3
M-147	TURBINE BUILDING SAMPLE	2
M-148	AREA RADIATION MONITORING	NONCLASS
M-149	OFF GAS RECOMBINER	NONCLASS
M-150	HVAC PLANT AIR FLOW	NONCLASS
M-151	CONTROL BUILDING & TSC AIR FLOW	NONCLASS
M-152	REACTOR BUILDING AIR FLOW	NONCLASS
M-153	TURBINE BUILDING AIR FLOW	NONCLASS
M-154	HVAC RADWASTE BUILDING AIR FLOW	NONCLASS
M-156	DRYWELL AIR FLOW	NONCLASS
M-157	DRYWELL COOLING WATER SH 1	2
M-158	HVAC AIR FLOW AND STANDBY GAS TREATMENT	NONCLASS
M-159	VENTILATION TURBINE BUILDING	NONCLASS
M-160	AUX. HEATING BOILER & MAIN LOOP SH 1	2
M-161	AIR CONDITIONING CONTROL BUILDING	NONCLASS
M-162	AUX. HEATING REACTOR BUILDING	NONCLASS
M-163	AUX. HEATING TURBINE BUILDING	NONCLASS
M-164	VENTILATION RADWASTE BUILDING	NONCLASS
M-165	MAIN PLANT AIR INTAKE & M.G. ROOM	NONCLASS
M-166	COOLING & HEATING PLANT AIR SUPPLY	NONCLASS
M-167	ADM. BUILDING HEATING AND COOLING	NONCLASS
M-168	ADM. BUILDING HEATING AND COOLING	NONCLASS
M-169	CONTROL BUILDING COOLING & PLANT CHILLED WTR SH 2, 3	3
M-170	HVAC MISC. CONTROL	NONCLASS
M-171	REACTOR BUILDING HVAC COOLING	3
M-172	AIR FLOW, HTG. CLG. MACH SHOP OFF GAS RETENTION BLDG.	NONCLASS
M-173	AIR FLOW STANDBY FILTER UNIT CONTROL	3
M-174	DRYWELL HEATING & VENTILATION FAN	NONCLASS
M-175	AIR FLOW PUMPHOUSE	NONCLASS
M-176	VENTILATION & OFF GAS STACK REACTOR BUILDING	NONCLASS
M-177	INTAKE, TSC, & WELL HS. HTG. AND VENTILATION CONTROL	NONCLASS
M-178	HVAC, MISC. CONTROL ROOM	NONCLASS
M-179	LEGEND (HVAC)	N/A
M-180	CHLORINATION & ACID FEED	NONCLASS
M-181	CONTAINMENT ATMOSPHERE MONITORING	2
M-182	RADWASTE EVAPORATION	NONCLASS
M-183	RADWASTE SAMPLE	NONCLASS
M-184	MSIV LEAKAGE CONTROL	1 & 2
M-185	FIRE PROTECTION CARBON MONOXIDE	NONCLASS
M-186	RADWASTE LIQUID WASTE STORAGE & HANDLING	2
M-187	POST ACCIDENT SAMPLING	2

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-114	Main Steam 'A'	1.2-1
M-114	Main Steam 'B'	1.2-2
M-114	Main Steam 'C'	1.2-3
M-114	Main Steam 'D'	1.2-4
M-114	Feedwater 'A' and 'B'	1.2-5
M-114	Feedwater 'C' and 'D'	1.2-6
M-121	Core Spray 'A'	1.2-7
M-121	Core Spray 'B	1.2-8
M-122	HPCI - Steam Side	1.2-9
M-123	HPCI - Water Side	1.2-10
M-127	RWCU - Suction Side	1.2-11A
M-127	RWCU - Discharge Side	1.2-11B
M-117	CRD Return	1.2-12A
M-117	CRD Return	1.2-12B
M-114	RHR Head Spray	1.2-13
M-119	RHR 18B	1.2-14
M-120	RHR-20A	1.2-15
M-119	RHR-20B	1.2-16
M-124	RCIC - Steam	1.2-17
M-125	RCIC - Water	1.2-18
M-116	Recirc 'A' - Bypass 'A'	1.2-19A
M-116	Recirc 'A' - Drain Line	1.2-19B

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-116	Recirc Manifold 'A' and Risers E, F, G, H	1.2-20
M-116	Recirc 'B' - Bypass 'B'	1.2-21A
M-116	Recirc 'B' Drain Line	1.2-21B
M-116	Recirc Manifold 'B' and Risers A, B, C, D	1.2-22
M-114	RPV Head Spray Spare	1.2-23
M-114	RPV Head Vent	1.2-24
M-115	Jet Pump Inst. 'A'	1.2-25
M-115	Jet Pump Inst. 'B'	1.2-26
M-126	SBLC	1.2-27
M-115	Vessel Instr. N-11A	1.2-28
M-115	Vessel Instr. N-11B	1.2-29
M-115	Vessel Instr. N-12A	1.2-30
M-115	Vessel Instr. N-12B	1.2-31
M-116 M-127	Bottom Head Drain	1.2-32
M-115	Vessel Instr. N-16A	1.2-33
M-115	Vessel Instr. N-16B	1.2-34
M-114	Main Steam Drain	1.2-35
M-116	Recirc Pump	1.3-01
M-116	Recirc Pump A Supports	1.3-02
M-116	Recirc Pump B Supports	1.3-03
M-119 M-120	RHR Ht. Exchanger A&B	2.1-01

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-120	RHR Pump Suction (S.E.)	2.2-32
M-119	RHR Pump Suction (N.W.)	2.2-33
M-119 M-120	RHR Pump Shutdown	2.2-34
M-120	RHR Pump Discharge(S.E.)	2.2-36
M-120	RHR Heat Exchanger Discharge (S.E.)	2.2-37A
M-120	RHR Heat Exchanger Discharge (S.E.)	2.2-37B
M-120	RHR Heat Exchanger Discharge (S.E.)	2.2-38
M-119	RHR Pump Discharge(N.W.)	2.2-39
M-119	RHR Heat Exchanger Discharge (N.W.)	2.2-40
M-119	RHR Heat Exchanger Discharge (N.W.)	2.2-41
M-119 M-134	RHR Fuel Pool Cooling and Cleanup	2.2-43
M-123	HPCI Pump Suction	2.2-44
M-123	HPCI Pump Discharge	2.2-45
M-122	HPCI Turbine Steam Inlet	2.2-46
M-122	HPCI Turbine Steam Exhaust	2.2-47
M-121	Core Spray Suction (S.E.)	2.2-48
M-119 & M-121	Core Spray Discharge (S.E.)	2.2-49
M-119	Core Spray Discharge (S.E.)	2.2-50
M-121	Core Spray Suction (N.W.)	2.2-51
M-120 & M-121	Core Spray Discharge (N.W.)	2.2-52A

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-120 & M-121	Core Spray Discharge (N.W.)	2.2-52B
M-103	Main Steam Loop 'A'	2.2-53
M-103	Main Steam Loop 'B'	2.2-54
M-103	Main Steam Loop 'C'	2.2-55
M-103	Main Steam Loop 'D'	2.2-56
M-103	Main Steam Bypass	2.2-57
M-103	Main Steam Bypass	2.2-58
M-118	Scram Discharge HDR (South)	2.2-60
M-118	Scram Discharge HDR(North)	2.2-61
M-109 M-119 M-125	RCIC Pump Suction	2.2-62
M-113	HPCI, RCIC and Reactor Building - ESW	3.1-1
M-146	Water Pumphouse - ESW	3.1.2
M-113	HPCI and Reactor Building - ESW	3.1-3
M-146	Water Pumphouse - ESW	3.1-4
M-113	Turbine Building - ESW	3.1-5
M-113	Turbine Building - ESW	3.1-6
M-113	Reactor Building - ESW	3.1-7
M-113	Turbine Building - ESW	3.1-8
M-113	Turbine Building - ESW	3.1-9
M-146	Water Pumphouse - RW	3.1-10

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-146	Water Pumphouse - RW	3.1-11
M-146	Water Pumphouse - RW	3.1-12
M-129	Intake Structure - RW	3.1-13
M-129	Intake Structure - RW	3.1-14
M-142	Water Pumphouse - RW	3.1-15
M-114	Reactor Building -Main Steam	3.1-16
M-114	Reactor Building -Main Steam	3.1-17
M-114	Reactor Building -Main Steam	3.1-18
M-114	Reactor Building -Main Steam	3.1-19
M-114	Reactor Building -Main Steam	3.1-20
M-114	Reactor Building -Main Steam	3.1-21
M-113	Reactor Building -RHR Service Water	3.1-22
M-113	HPCI Building - RHR Service Water	3.1-23
M-113	HPCI and Reactor Building - RHRSW	3.1-24
M-129	Intake Structure - RW	3.1-25
M-129	Intake Structure - RW	3.1-26
M-146	Water Pumphouse - RW	3.1-27
M-113	HPCI and Reactor Building - RHRSW	3.1-28
M-146	Water Pumphouse - RHRSW	3.1-29
M-146	Water Pumphouse - RHRSW	3.1-30

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
M-113	HPCI and Reactor Building - RHRSW	3.1-31
M-113	HPCI, RCIC and Reactor Building - ESW	3.1-32
M-113	HPCI Building - RHRSW	3.1-33
M-146	Water Pumphouse - RHRSW	3.1-34
M-146	Water Pumphouse - RHRSW	3.1-35
M-113	Reactor Building - RHRSW	3.1-36
M-113	HPCI Building - RHRSW	3.1-37
M-113	Reactor Building - ESW	3.1-38
N/A	Top Head Assembly	VS-01-06
N/A	Bottom Head Assembly	VS-01-07
N/A	SKIRT KNUCKLE & SKIRT EXTENSION	VS-01-08
N/A	CORE SPRAY BRACKET	VS-01-26
N/A	FEEDWATER SPARGER BRACKET	VS-01-27
N/A	GUIDE ROD BRACKET	VS-01-28
N/A	STEAM DRYER SUPPORT BRACKET	VS-01-29
N/A	STEAM DRYER HOLD-DOWN BRACKETS	VS-01-30
N/A	SURVEILLANCE SPECIMEN BRACKET	VS-01-31
N/A	JET PUMP RISER SUPPORT PAD	VS-01-34

(1) List of Applicable Piping Isometric Drawings

P&ID No.	DESCRIPTION	ISI ISOMETRIC DWG. No.
N/A	VESSEL SHELL RING LAYOUT	VS-01-41
N/A	SHROUD SUPPORT	VS-02-10
N/A	LEDGE SEGMENT PLATES	VS-02-11
N/A	CORE PLATE	VS-03-02
N/A	PERIPHERAL FUEL SUPPORT	VS-03-09
N/A	TOP GUIDE	VS-04-01
N/A	SURVEILLANCE PROGRAM	VS-10-02
N/A	CRD GUIDE TUBE	VS-12-04
N/A	DRY TUBE	VS-13-02

(1) List of Applicable Calibration Standards Drawings - Class 1 Carbon Steel

Cal. Block ID#	Nominal Pipe Size	Pipe Schedule	Thickness (inches)	Heat No.	Cal. Block Dwg. No.
IE-01	2"	Sch. 80	0.218	L4449	---------
IE-02	3"	Sch. 80	0.300	N55489	131C7903
IE-03	4"	Sch. 80	0.337	84A711	131C7903
IE-04	4"	Sch. 160	0.531	J616162	LMT-362
IE-05	6"	Sch. 80	0.432	L40321	131C7903
IE-58	6"	Sch. 160	0.719	23250	LMT-428
IE-06	8"	Sch. 80	0.500	123748	131C7903
IE-59	9", 508	Sch. 160	1.6195	523477	LMT-419
IE-07	10"	Sch. 80	0.593	62163	131C7903
IE-08	11"	---	1.090	51122	LMT-358
IE-09	12"	---	0.687	DXR8155	131C7903
IE-10	16"	---	0.843	49069	131C7903
IE-11	18"	---	0.937	89C753	131C7903
IE-51	20"	Sch. 80	1.031	N72753	-------
IE-12	20"	---	1.5	M52851	166B7258
IE-13	22"	Sch 80	1.125	L20112	LMT-357
IE-60	6.375"	---	1.27	17528	LMT-474
ISI-1018	12.25"	---	0.694 to 0.75	121SNH2	ISI-1018
5538	Alternative	---	0.50 to 2.00	T649	10052

(2) List of Applicable Calibration Standards Drawings - Class 1 Stainless Steel

Cal. Block ID#	Nominal Pipe Size	Pipe Schedule	Thickness (inches)	Heat No.	Cal. Block Dwg. No.
IE-14	1.5"	Sch. 80	0.200	432346	LMT-355
IE-15	2"	Sch. 80	0.215	308028	-------
IE-55	2.5"	Sch. 80	0.276	74835	LMT-407
IE-16	3"	Sch. 80	0.300	M6445	131C7903
IE-17	4"	Sch. 160	0.531	M2458	LMT-361
IE-18	4"	Sch. 80	0.337	80359	131C7903
IE-56	4"	Sch. 40	0.220	14241	LMT-430
IE-57	4" (316L)	Sch. 80	0.337	AJ9219	LMT-429
IE-19	8"	Sch. 80	0.500	80407	131C7903
IE-20	10"	Sch. 80	0.985	10SS 304WOL	-------
IE-21	10"	---	0.594	651345	131C7903
IE-22	16"	---	0.844	132002	131C7903
IE-54	18"	---	0.935	A3533	-------
IE-23	18"	Sch. 80	1.300	67695-A	-------
IE-24	20"	---	1.500	3160816A	166B7258
IE-25	20"	Sch. 80	1.031	10093	-------
IE-26	22"	Sch. 80	1.125	28730	-------
WE-Size-03-SS	Alternative	---	---	23561	C-2367-624C
5539	Alternative	---	0.50 to 2.00	A13554	10052

(3) **List of Applicable Calibration Standards Drawings - Class 2 Carbon Steel**

Cal. Block ID#	Nominal Pipe Size	Pipe Schedule	Thickness (inches)	Heat No.	Cal. Block Dwg. No.
IE-38	6"	Sch. 40	0.280	N8023	LMT-7-28-77
IE-39	8"	Sch. 40	0.322	CS001	-------
IE-40	10"	Sch. 40	0.365	N8024	LMT-7-28-77
IE-41	12"	Sch. 40	0.406	N8025	LMT-7-28-77
IE-42	16"	Sch. 40	0.500	N8026	LMT-7-28-77
IE-43	18"	Sch. 40	0.562	N8027	LMT-7-28-77
IE-44	20"	Sch. 40	0.594	N14071	LMT-7-28-77
IE-45	8"	Sch. 100	0.594	L20632	LMT-084
IE-46	10"	Sch. 100	0.719	58205	LMT-087
IE-47	12"	Sch. 100	0.844	57083	LMT-086
IE-48	14"	---	0.375	L02777	LMT-167
IE-49	16"	---	0.375	L80611	LMT-168
IE-50	18"	---	0.375	N15689	LMT-169
IE-52	20"	---	0.375	N94046-20	LMT-170
IE-53	24"	---	0.375	N94046-24	LMT-171

(4) List of Applicable Calibration Standards Drawings - Class 1 Inconel 600

Cal. Block ID#	Nominal Pipe Size	Pipe Schedule	Thickness	Heat No.	Cal. Block Dwg. No.
IE-27	12"	--------	0.75"	NX9724	LMT-038
IE-28	8.7"	--------	0.66"	72534	LMT-360
IE-29	11"	--------	1.09"	72614	LMT-359

(5) **List of Applicable Calibration Standards Drawings - Step Wedge Blocks**

Cal. Block ID#	Material Type	Heat No.	Thickness Range	Cal. Block Dwg. No.
QD-UT-1	SA 516 GR 70 CS	432L0241-L216703	.250"-2.000"	N/A
QD-UT-2	A216 WCB CS	N/A	.900"-3.900"	N/A
QD-UT-3	316 SS	89764	.500"-2.500"	N/A
QD-UT-4	304 SS	89908	.500"-2.500"	N/A
QD-UT-5	A-36 CS	Y75453	.500"-2.500"	N/A
QD-UT-6	Cast CF8M SS	N/A	.250"-2.000"	N/A
QD-UT-7	304 SS	89908	.101"-.500"	N/A
QD-UT-8	Inconel	N/A	.428"-1.591"	N/A
QD-UT-9	Aluminum	N/A	.208"-.728"	N/A
QD-UT-10	AISI 1018 CS	S/N 798705	.100"-.500"	N/A
QD-UT-11	Copper	N/A	.1"-.4700"	N/A
QD-UT-12	1018 CS	A08146	.1"-.5"	N/A
QD-UT-13	1018 CS	A07588	.250"-1.00"	N/A

(6) **List of Applicable Calibration Standards Drawings - RPV Blocks and Studs**

Cal. Block ID#	Nominal Pipe Size	Pipe Schedule	Thickness (inches)	Heat No.	Cal. Block Dwg. No.
IE-30	PLATE	N/A	5.5"	B0402	LMT-SK
IE-31	PLATE	N/A	6.625"	P2112	SK-4-7-78
IE-32	PLATE	N/A	6.625"	P2130	SK-4-7-78
IE-33	PLATE	N/A	6.625"	T1937	SK-4-7-78
IE-34	PLATE	N/A	6.625"	P2076	SK-4-7-78
IE-35	PLATE	N/A	4.0"	B0390	SK-4-7-78
IE-36	STUDS	N/A	5.187"	8083916	IE-36
IE-37	STUDS	N/A	2.75"	61994	N/A
IE-61	NOZ TO SAFEND	N/A	.8445"	40580-1	IOWA-N201
IE-62	RPV STUD	N/A	5.375	PC23-1	IE-62
83B (N2)	NOZ-THERMALSLEEVE	N/A	N/A	125M	103E1034
58B (N1)	NOZ TO SAFEND	N/A	N/A	218993	D2371-175

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
B-A	B1.11	Circumferential Shell Welds	4	Volumetric	NDE-R003	
	B1.12	Longitudinal Shell Welds	8	Volumetric		
	B1.21	Circumferential Head Welds	2	Volumetric		
	B1.22	Meridional Head Welds	15	Volumetric		
	B1.30	Shell-to-Flange Weld	1	Volumetric	NDE-R008	
	B1.40	Head-to-Flange Weld	1	Volumetric & Surface	NDE-R008	
B-D	B3.90	Nozzle-to-Vessel Welds in Reactor Vessel	34	Volumetric		
	B3.100	Nozzle Inside Radius Section in Reactor Vessel	34	Volumetric		
B-F	B5.10	Reactor Vessel Nozzle-to-Safe End Butt Welds NPS 4 or Larger	N/A	Volumetric & Surface	NDE-R005	
	B5.20	Reactor Vessel Nozzle-to-Safe End Butt Welds Less Than NPS 4	N/A	Surface	NDE-R005	
	B5.30	Reactor Vessel Nozzle-to-Safe End Socket Welds	N/A	Surface	NDE-R005	
B-G-1	B6.10	Reactor Vessel Closure Head Nuts	60	Visual, VT-1	NDE-R006	
	B6.20	Reactor Vessel Closure Studs, in Place	60	Volumetric		
	B6.40	Threads in Reactor Vessel Flange	60	Volumetric		

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
B-G-1	B6.50	Reactor Vessel Closure Washers, Bushings	60	Visual, VT-1	NDE-R006	
	B6.180	Bolts & Studs in Pumps	2 Sets of 16	Volumetric		
	B6.190	Flange Surface, When Connection Disassembled, in Pumps	2 sets of 16	Visual, VT-1	NDE-R006	
	B6.200	Nuts, Bushings, & Washers in Pumps	2 Sets of 16	Visual, VT-1	NDE-R006	
B-G-2	B7.10	Bolts, Studs, & Nuts in Reactor Vessel	3 locations	Visual, VT-1	NDE-R006	
	B7.50	Bolts, Studs, & Nuts in Piping	5 locations	Visual, VT-1	NDE-R006	
	B7.60	Bolts, Studs, & Nuts in Pumps	2 Sets of 16	Visual, VT-1	NDE-R006	
	B7.70	Bolts, Studs, & Nuts in Valves	47 valves	Visual, VT-1	NDE-R006	
	B7.80	Bolts, Studs, & Nuts in CRD Housings	89 Sets	Visual, VT-1	NDE-R006	
B-J	B9.11	Circumferential Piping Welds NPS 4 or Larger	N/A	Volumetric & Surface	NDE-R005	
	B9.21	Circumferential Piping Welds	N/A	Surface	NDE-R005	
B-J	B9.31	Branch Pipe Connection Welds NPS 4 or Larger	N/A	Volumetric & Surface	NDE-R005	
	B9.32	Branch Pipe Connection Welds Less Than NPS 4	N/A	Surface	NDE-R005	
	B9.40	Socket Welds	N/A	Surface	NDE-R005	

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
B-K	B10.10	Integrally Welded Attachments to Pressure Vessels	5 (one vessel)	Surface	NDE-R004	
	B10.20	Integrally Welded Attachments to Piping	32	Surface		
	B10.30	Integrally Welded Attachments to Pumps	8	Surface		
B-L-2	B12.20	Pump Casings	2	Visual, VT-3	NDE-R006	
B-M-2	B12.50	Valve Bodies, Exceeding NPS 4	45	Visual, VT-3	NDE-R006	
B-N-1	B13.10	Vessel Interior	12	Visual, VT-3	NDE-R006	
B-N-2	B13.20	Interior Attachments within Beltline Region in Reactor Vessel	11	Visual, VT-1	NDE-R006	
	B13.30	Interior Attachments beyond Beltline Region in Reactor Vessel	27	Visual, VT-3	NDE-R006	
	B13.40	Core Support Structure in Reactor Vessel	5	Visual, VT-3	NDE-R006	
B-O	B14.10	Welds in CRD Housing, Peripheral CRDs	28	Volumetric or Surface		
B-P	B15.10	System Leakage Test	1	Visual, VT-2	NDE-R006	
C-A	C1.10	Circumferential Shell Welds	2 (one vessel)	Volumetric		
	C1.20	Circumferential Head Welds	1 (one vessel)	Volumetric		
C-B	C2.21	Nozzle-to-Shell (or Head) Weld without Reinforcing Plates in Vessels > 1/2" Nominal Thickness	2 (one vessel)	Volumetric & Surface		

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
C-B	C2.22	Nozzle Inner Radius	2 (one vessel)	Volumetric		
C-C	C3.10	Integrally Welded Attachments to Pressure Vessels	5 (one vessel)	Surface	NDE-R004	
	C3.20	Integrally Welded Attachments to Piping	61	Surface		
C-F-2	C5.51	Circumferential Welds in Carbon or Low Alloy Steel Piping > or = 3/8" Nominal Wall Thickness For Piping > NPS 4	N/A	Volumetric & Surface	NDE-R005	
	C5.81	Circumferential Welds in Carbon or Low Alloy Steel Branch Connections Piping For Piping > NPS 4 (Reference Table IWC-2500-1, Note 1).	N/A	Surface	NDE-R005	
C-H	C7.10	System Leakage Test of Pressure Retaining Components	7	Visual, VT-2	NDE-R006	
D-A	D1.20	Integral Attachments - Piping	68	Visual, VT-1	NDE-R006	
D-B	D2.10	System Leakage Test	3	Visual, VT-2	NDE-R006 NDE-R007	
F-A	F1.10	Class I Component Supports	165	Visual, VT-3	NDE-R006	
	F1.20	Class II Component Supports	330	Visual, VT-3	NDE-R006	
F-A	F1.30	Class III Component Supports	238	Visual, VT 3	NDE-R006	
	F1.40	Supports Other Than Piping Supports (Class 1, 2, and 3)	29	Visual, VT-3	NDE-R006	

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
R-A		Risk-Informed weld selection for Categories B-F, B-J, and C-F-2 based on EPRI methodology			NDE-R005	
	R1.10 R1.11 R1.14 R1.16 R1.18	No Degradation Mode Thermal Fatigue Corrosion Cracking Intergranular Stress Corrosion Cracking Flow-Acclerated Corrosion	70	Volumetric		

1.0 INDEX

Position	Revision	Type	Description
TAP-I001	0	DAEC	Weld Reference System.
TAP-I002	0	DAEC	Valve discs that are considered pressure retaining and requiring repair/replacement activities in accordance with IWA-4000.
TAP-I003	0	DAEC	Preservice examinations for Risk-informed Welds.

(1) TECHNICAL APPROACH AND POSITION NUMBER: TAP-I001

COMPONENT IDENTIFICATION

Code Classes: 1 and 2
References: IWA-2600
Examination Category: Not Applicable
Item Number: Not Applicable
Description: Weld Reference System

CODE REQUIREMENT

IWA-2610, "Weld Reference System" states a reference system shall be established for all welds and areas subject to surface or volumetric examination.

Each such weld and area shall be located and identified by a system of reference points. The system shall permit identification of each weld, location of each weld centerline, and designation of regular intervals along the weld length.

POSITION

At the time DAEC was constructed, datum reference markings nor a reference system were required by Code. Application of such physical markings to each and every area subject to surface or volumetric examination (in an operating plant) would require significant expenditure of resources and result in additional, unnecessary personnel radiation exposure. In many instances, limited or no physical access is available to permit such markings.

It is DAEC's position to continue using the present weld identification method successfully employed during the three previous 10 year inspection intervals. This is accomplished by procedurally describing datum or reference points such that subsequent location of the examination area can be repeatedly achieved.

During the course of performing examinations for the fourth inspection interval, in accordance with the requirements of the Inservice Inspection Program Plan, weld reference points will be physically applied to welds where flaw indications are detected and determined to be relevant. Flaw indications or relevant conditions qualified for continued service through evaluation shall be reexamined during subsequent inspection periods in accordance with IWX-2420.

Where new welds are installed as a result of repair and replacement and require preservice inspection the requirements of IWA-2600 will be met.

(2) TECHNICAL APPROACH AND POSITION NUMBER: TAP-I002

COMPONENT IDENTIFICATION

Code Classes: 1, 2, and 3
References: IWA-4000

Examination Category: Not Applicable
Item Number: Not Applicable
Description: Valve discs that are considered pressure retaining and requiring repair/replacement activities in accordance with IWA-4000.

CODE REQUIREMENT

IWA-4110(b) states in part "This article provides requirements for repair/replacement activities associated with pressure retaining components and their supports, including appurtenances, subassemblies, parts of a component, core support structures, metal containments and their integral attachments, and metallic portions of Class CC containments and their integral attachments."

POSITION

There are many valves used in nuclear power plants. The type of valves used depends on the design and function required. Valves are used for on-off service, modulating/throttling service, to protect components against overpressure, and to prevent backflow from occurring. Some of the valve types used are :

1) Gate Valves
2) Ball Valves
3) Butterfly Valves
4) Globe Valves
5) Check Valves
6) Plug Valves

In general, valves may be categorized within the following four groups

1) Isolation Valves: Used for on-off service (including throttled position) with local or remote actuation. Depending on the particular application and operating conditions isolation valves can be either gate, globe, butterfly, ball, plug, or diaphragm valves.
2) Control Valves: Used for modulating or throttling service. Their operation is automatic in response to continuous monitoring of some parameter in the controlled system. In general, control valves require no manual operator action. A control valve functions as a variable resistance in a pipeline.

3) Pressure Relief Valves: Used to provide protection against excessive pressure. The valve opens automatically when pressure exceeds a preset level and closes after pressure recedes below a preset level. Power-operated relief valves that open or close in response to command signals are also utilized.
4) Check Valves: Used to allow flow in the normal flow direction and to prevent flow in the opposite flow direction (reverse flow). Check valves are typically opened and closed by the flow forces.

The following table shows the types of valves used for each of the above-mentioned functions:

Valve Functions and Types			
Isolation	**Control**	**Check**	**Pressure Relief**
Gate	Globe	Swing Check	Self-Acting Relief
Globe	Butterfly	Lift Check	Self-Acting Safety
Butterfly	Ball Including Cam Type	Tilting-Disc Check	Self-Acting Safety/Relief
Ball	Self-Contained Regulators	Double-D Check	Power-Operated Relief
Plug		Silent/Nozzle Check	
Diaphragm		Stop Check	

Those valves that would be used for Isolation, Check, and Pressure Relief could be those considered to have discs that would be pressure retaining based on the function. However, further investigation is needed to define which valve discs would be required to "act" as a pressure retaining disc. This is accomplished by reviewing 10CFR50.1. The definition of a *Basic Component* when applied to a nuclear power plant is any plant structure, system, component, or part thereof necessary to assure:

1) The integrity of the reactor coolant pressure boundary.
2) The capability to shut down the reactor and maintain it in a safe condition, or
3) The capability to prevent or mitigate the consequences of accidents which could result in potential offsite exposures comparable to those referred to in 50.34(a)(1), 50.67(b)(2), or 100.11 of this chapter, as applicable.

The definition goes on to state "In all cases, *basic component* includes safety related design, analysis, inspection, testing, fabrication, replacement parts, or consulting services that are associated with the component hardware, whether these services are performed by the component supplier or other supplier."

Therefore, for those valves, the IST Program is a good source for which ones would be used for meeting the three criteria mentioned above. The OM Code uses the same criteria for Subsection ISTC in defining the program scope.

In addition, the ISI Program (IWA-1400[a]) provides the criteria and basis of the ASME Section XI Code classification of systems and components for the purposes of Inservice Inspection (ISI) and repair/replacement activities. In that classification process, valves will be designated as "boundary" valves, meaning the valve will be the class break. For example, a valve will be the boundary between Class 1 and 2 systems. This valve would also have a disc that would be used for pressure retaining function.

Conclusion

For implementation of repair/replacement activities on valve discs, the following position is taken. Only those valves identified in the ISI Classification that have been designated as valves between classes will be designated with pressure retaining disc assemblies. In addition, all valves that are required to close to perform a specific function in shutting down the reactor to the cold shutdown condition, in maintaining the cold shutdown condition, or mitigating the consequences of an accident shall be designated as valves with pressure retaining disc assemblies (review the IST Program to determine which valves these are). All other boundary valve internal components and disc are considered as non-structural internal components.

References

EPRI Report (TR-105852-V2), *Valve Application, Maintenance, and Repair Guide*,
10 CFR (Code of Federal Regulation) Part 50.1.

ASME OM Code-2001, Code for Operation and Maintenance of nuclear power plants.

(3) TECHNICAL APPROACH AND POSITION NUMBER: TAP-I003

COMPONENT IDENTIFICATION

Code Classes:	1 and 2
References:	Risk-Informed ISI Program
Examination Categories:	B-F, B-J, C-F-2, R-A
Item Numbers:	All

EXAMINATION REQUIREMENTS

Inservice examinations, for those plants using the EPRI methodology, are performed on piping welds in accordance with approved Relief Requests for Risk-Informed Inservice Inspection (RI-ISI) and the EPRI Topical Report, TR-112657, Rev. B-A.

Preservice examinations are performed on piping welds in accordance with the requirements of the applicable Edition and Addenda of ASME Section XI.

POSITION

The examination of piping welds is required to be performed on a periodic basis. In the case of DAEC, a RI-ISI Program has been approved for use as described in the NRC SER dated January 17, 2003 for the third interval and by relief request for the fourth interval submitted here-in. The number and selection of welds and how often they are to be examined is provided by a combination of the submittal and the SER, which reference the EPRI topical report. There are several different issues to be considered as to what type of examination is to be performed on each weld.

The scope of the RI-ISI alternative is for the Inservice Inspection of piping. The objective of the submittal is to request the use of a risk-informed process for the Inservice Inspection of Class 1 and 2 piping. Relief was not requested from the preservice examination requirements of ASME Section XI.

Section XI Preservice Inspection requirements are essentially the same as Inservice Inspection requirements. The difference is that the NDE examinations are performed prior to placing the components in service. In addition, all class 1 components are examined in accordance with the Inservice Inspection requirements. For class 2 and 3 components, examination is limited to only those components that will be examined in the future. Since these examinations are performed prior to the placement of the welds into service, no in-service time has been accumulated and any degradation mechanism present would not have had any time to degrade the new weld.
New piping welds installed due to repair/replacement activities shall be evaluated for inclusion in the RI-ISI Program. If the repair/replacement activity is part of a Modification Activity, the evaluation shall be

performed as part of that Modification Activity. The evaluation for all other repair/replacement activities shall be performed as part of the Repair/Replacement Plan. In accordance with the approved Relief Request, a periodic review of the RI-ISI Program is required to evaluate any changes in the plant, operating experience, and any other issues that may affect the initial evaluations.

Butt Welds

For inservice examination of butt welds, the code alternate requires a volumetric examination instead of the surface and volumetric examinations as required by the Code. The examination volume is increased for some welds due to the effects of the degradation mechanism, but as a conservative policy, the most restrictive examination area should be examined on every weld to the extent practicable.

For preservice examination of new piping butt welds, surface and volumetric examinations as required by the applicable Section XI Code shall be performed. The most conservative volumetric examination requirements of the RI-ISI Program shall be followed. Surface examinations are usually required as part of the Construction Code and will be part of the repair/replacement activity, so this would probably not add any additional work scope. The surface examinations procedures shall meet the applicable Section XI Code requirements. A VT-2 visual examination shall be performed as part of the repair/replacement activity of the weld.

Socket Welds

For inservice examination of socket welds, the submittal says "it should be noted that non-socket welds are subject to volumetric examination, so this percentage does not rely upon welds that are solely subject to a VT-2 visual examination." This statement would require a VT-2 examination instead of the surface examination as required by the Code. EPRI Topical Report 1000701 (MRP) gives guidance for performing volumetric examination of socket welds. The examination technique would look at accessible susceptible areas near and around the welds.

For preservice examination of new piping socket welds, surface examinations as required by the applicable Section XI Code shall be performed. Surface examinations are usually required as part of the Construction Code and will be part of the repair/replacement activity, so this would probably not add any additional work scope. The surface examinations procedures shall meet the applicable Section XI Code requirements. A VT-2 visual examination shall be performed as part of the repair/replacement of the weld.

<u>CONCLUSION</u>

Inservice Inspection of welds shall be a volumetric examination. Any welds that cannot meet this requirement should not be examined until an examination technique is developed or an acceptable alternative is approved by the NRC.

Preservice examination of welds shall be in accordance with the applicable ASME Section XI Code with volumetric examination areas extended to meet the most conservative requirements of the submittal, the SER and the EPRI Topical Report.

New piping welds installed due to repair/replacement activities shall be evaluated for a degradation mechanism. If the repair/replacement activity is part of a Modification Activity, the evaluation shall be performed as part of that Modification Activity. The evaluation for all other repair/replacement activities shall be performed as part of the Repair/Replacement Plan.

REFERENCES

EPRI Topical Report, TR-112657, Rev. B-A.

Interim Thermal Fatigue Management Program (MRP) EPRI TR 1000701, dated January 2001.
Standard template for RI-ISI Relief Requests.

SER 2003-0009 from PBNP.

ASME Code Case N-578, *Risk-Informed Requirements for Class 1, 2, and 3 Piping*, Method B.

Inservice Engineering Record of Conversation ROC-00, from PBNP, dated November 29-30, 2001.

1.0 PURPOSE

The following section contains an index and the relief requests written in accordance with 10CFR50.55a(a)(3) and (g)(6) when specific ASME Section XI requirements for inservice inspection are considered impractical or pose an undo burden on the licensee. The relief requests contained in Section H are subject to change during the course of the ten year inspection interval as a result of changes in technology, plant design or as a result of installed modifications. If examinations or tests are determined to be impractical, or result in hardship or unusual difficulty without a commensurate increase in the level of quality or safety, during the course of the interval, additional or modified relief requests will be submitted in accordance with 10CFR50.55(a)(3) and (g)(6).

The following Table is an index which summarizes each relief request and provides for sequential numbering to maintain continuity for the remaining inspection intervals for DAEC.

2.0 INDEX

Relief Request	Status	Rev	Summary
NDE-R001	Submitted	0	Use of the DAEC Technical Requirements Manual for Snubber Visual Examination & Testing.
NDE-R002	Submitted	0	Approved use of PDI for Overlays in lieu of Supplement 11 to Appendix VIII.
NDE-R003	Approved	0	Requested to use BWRVIP-05 recommendations for reduced circumferential vessel weld exams.
NDE-R004	Submitted	0	Request use of Code Case N-700 "Selection of Class 1, 2, and 3 Vessel Welded Attachments"
NDE-R005	Submitted	0	Risk Informed ISI for Class I B-F & B-J Welds and Class 2 & C-F-2 Welds
NDE-R006	Submitted	0	Request use of Code Case N-686 "Alternate Requirements for Visual Examinations"
NDE-R007	Submitted	0	Alternative for pressure testing and visual examination of buried piping and components.
NDE-R008	Submitted	0	Request approval to use Appendix VIII examinations for the Fourth 10-Year Interval reactor vessel-to-flange weld and head-to-flange weld in lieu of the existing requirements to use Section V.

(1) Relief Request NDE–R001

COMPONENT IDENTIFICATION

Code Classes:	1, 2, and 3
References:	ASME, Section XI, IWF-5000 (2001 Edition through 2003 Addenda) ASME / ANSI OM-1987, Part 4, First Addenda (1988)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	Continue use of DAEC's Technical Requirements Manual for visual examination & functional testing of snubbers.
Component Numbers:	All Class 1, 2, and 3 Component Snubbers

CODE REQUIREMENT

Paragraphs IWF-5200(a) and IWF-5300(a) require Preservice and Inservice examinations to be performed in accordance with ASME/ANSI OM-1987, Part 4, using VT-3 visual examination method described in IWA-2213. Additionally, paragraphs IWF-5200(b) and IWF-5300(b) respectively require Preservice and Inservice tests to be performed in accordance with ASME/ANSI OM, Part 4. Table IWA-1600-1 specifies use of the 1987 Edition, with OMa-1988 Revision of ASME/ANSI OM, Part 4. Paragraph IWF-5300(c) requires integral and non-integral attachments for snubbers, including lugs, bolting pins, clamps to be visually examined in accordance with ASME Section XI, Subsection IWF.

The regulation in 10 CFR 50.55a(b)(3)(v) permits the use of Subsection ISTD, titled "*Inservice Testing of Dynamic Restraints (Snubbers) in Light-water Reactor Power Plants,*" ASME OM Code, 1995 Edition up to and including the 2003 Addenda, in lieu of the requirements for snubbers in Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to their technical specifications or licensee controlled documents. Preservice and inservice examinations shall be performed using the VT-3 visual examination method described in IWA-2213.

BASIS FOR RELIEF

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety.

The Duane Arnold Energy Center (DAEC) Technical Requirements Manual (TRM) Section 3.7.2 imposes alternative surveillance requirements for both visual inspections and functional testing of all safety related snubbers. Functional testing provides a 95 percent confidence level that 90 percent to 100 percent of the snubbers operate within the specified acceptance limits. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability. Visual examination requirements are based on NRC Generic Letter 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions."

For clarification, this 10CFR50.55a Request includes only the snubber and its pin-to-pin connections and does not include the remaining portion of the Section III NF support containing a snubber. As required by IWF-5200(c) and IWF-5300(c) the examination of the remaining portion of the support, including integral and nonintegral attachments for supports containing snubbers will be performed in accordance with Section XI Subsection IWF as part of the Inservice Inspection Program.

Implementation of TRM requirements for snubber visual examination and functional testing has maintained a reliable snubber population. The TRM requirements provide an equivalent level of quality and safety. These alternative requirements were previously reviewed and approved by the staff in amendment 203 to the DAEC Technical Specifications (TS).

The mechanical and hydraulic snubbers were constructed and installed in accordance with the requirements of the DAEC Updated Final Safety Analysis Report (UFSAR). Documentation of fabrication and installation examinations is stored at the plant site. Subsequent to the plant going into operation, these snubbers have been and continue to be visually inspected and functionally tested in accordance with the applicable requirements. The regulation in 10CFR 50.55a(b)(3)(v) permits the use of Subsection ISTD, titled "Inservice Testing of Dynamic Restraints (Snubbers) in Light-Water Reactor Power Plants" ASME OM Code, 1995 Edition up to and including the 2001 Edition through the 2003 Addenda, in lieu of the requirements for snubbers in Section XI, IWF-5200(a) and (b) and IWF-5300(a) and (b), by making appropriate changes to their technical specifications or licensee controlled documents. The attached TRM/ISTD Comparison Table allows comparison of specific key criteria between the TRM and ISTD.

The current TRM snubber visual examination and functional testing requirements have not been changed since they were originally removed from DAEC's Technical Specification as part of Improved Technical Specifications (ITS) implemented on August 1, 1998.

ALTERNATE EXAMINATION

DAEC proposes to continue to use the DAEC's Technical Requirements Manual Section 3.7.2 requirements for visual examination & functional testing for all ASME Class 1, 2, and 3 component snubbers in lieu of those contained in OMa-1988a Part 4 as referenced in ASME Section XI, Subarticles IWF-5200(a) and IWF-5300(b)

Visual examiners, who are qualified to the applicable rules of ASME Section XI, Article IWA-2300 "Qualifications of Nondestructive Examination Personnel" will perform the examinations and tests of Class 1, 2, and 3 component snubbers. Visual examination and testing results will be recorded and reported in accordance with the applicable rules of ASME Section XI, Article IWA-6000.

APPLICABLE TIME PERIOD

Relief is requested for the fourth ten-year interval of the Inservice Inspection Program for DAEC.

PRECEDENCE

The NRC approved McGuire Nuclear Station Unit 2 Relief Request RR-03-002 on November 22, 2004 to allow the use of their Selected Licensee Commitment 16.9.15 for their third 10-year inservice inspection interval for snubber testing/inspection (TAC No. MC2384)

Susquehanna Steam Electric Station Units 1 and 2 Relief Request 3RR-03 was approved by the NRC on September 24, 2004 to allow the use of their TRM snubber program for their third 10-year inservice inspection interval for snubber testing/inspection. (TAC No. MC1185 and MC1186).

TRM/ISTD Comparison Table

Criteria	ASME OM Code 2003 Addenda (Subsection ISTD) Requirements	DAEC TRM Section T3.7.2 Requirements
Snubber sample size	ISTD-5261 requires that each defined test plan group shall be tested using either a 10% sampling plan; or a 37 testing sample plan each refueling outage.	At least every 24 months, a representative sample of 10% of the total of each type of snubber in use shall be functionally tested either in place or in a bench test.
Examination requirements	ISTD-4210 states that snubber visual examinations shall identify physical damage, leakage, corrosion, or degradation. Also ISTD-5210 and ISTD-5120 require that operational readiness tests shall verify activation, release rate, and drag force by either an in-place or bench test. IWA-2213 also provides requirements for VT-3 examinations of snubbers.	TSR 3.7.2.2 requires that visual inspections shall verify that there are: (1) no visible indications of damage or impaired operability; (2) attachments to foundation or supporting structures are secure; and (3) fasteners for the attachment of the snubber to component and snubber anchorage are secure. TSR 3.7.2.3 b requires in place or bench tests to verify activation, snubber bleed or release rate, and TSR 3.7.2.3.c requires in place or bench testing to verify activation, drag force and release rate for hydraulic and mechanical snubbers.
Failure evaluation	Snubbers not meeting test requirements shall be evaluated to determine the root cause for the failure in accordance with ISTD-5271.	TLCO 3.7.2B requires: for snubbers that fail to lock-up or to move (frozen) during performance of functional testing, determine the cause of failure is not due to manufacturer or design deficiency or all snubbers subject to the same defect, shall be functionally tested.

Criteria	ASME OM Code 2003 Addenda (Subsection ISTD) Requirements	DAEC TRM Section T3.7.2 Requirements
Additional sampling	Additional snubbers are to be tested based on the number of failures in accordance with ISTD-5273.	TLCO 3.7.2B.3 requires that an additional 5% of the type of snubber that failed functional testing be tested.
Corrective Actions	ISTD-5280 requires that unacceptable snubbers shall be adjusted, repaired, modified, or replaced.	TLCO 3.7.2B.1 requires that inoperable snubbers would be replaced or restored to operable status.
Subsequent examination intervals.	ISTD-4250 provides guidance for examination intervals. Intervals are to be based on Table ISTD-4252-1.	TRM Table T3.7.2-1 provides a snubber visual inspection interval based on the number of unacceptable snubbers discovered. Requirements are similar to Table ISTD-4252-1.

(2) Relief Request NDE-R002

COMPONENT IDENTIFICATION

Code Class:	1
References:	IWB-2500, Table IWB-2500-1
Examination Categories:	B-F, B-J
Item Number:	B5.10, B5.20, B5.30, B9.11, B9.21, or B9.31 that are overlayed
Description:	Relief to use the Performance Demonstration Initiative (PDI) Program for implementation of Appendix VIII, Supplement 11 requirements.

SYSTEM/COMPONENT(S) FOR WHICH RELIEF IS REQUESTED

The affected components are Duane Arnold Energy Center (DAEC) pressure retaining welds in piping, subject to American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, 2001 Edition, Appendix VIII, Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds," examination.

CODE REQUIREMENTS

Fourth interval examinations will be performed per the requirements of ASME Section XI, 2001 Edition through the 2003 Addenda, as amended by 10 CFR 50.55a.

Per 10 CFR 50.55a(b)(2)(xxiv), the use of Appendix VIII and supplements to Appendix VIII of Section XI of the 2002 Addenda through the 2003 Addenda is prohibited. Therefore, for Appendix VIII and supplements to Appendix VIII the 2001 Edition of Section XI (no addenda) will be used.

The following paragraphs are examples of the code requirements for which relief is requested, all of which are contained within 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% of the flaws shall be oriented within $\pm$20 degrees of the pipe 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 three inches of the length of the overlaid weld.

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 six-square-inches. The overlay grading unit shall be rectangular, with minimum dimensions of two inches.

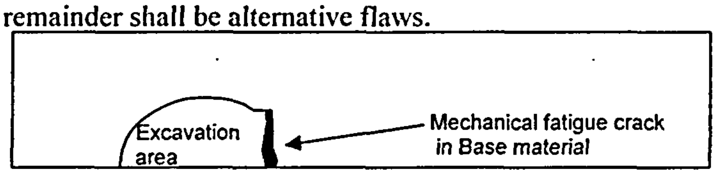
Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.1 inches be reported as being intrusions into the overlay material.

RELIEF REQUESTED

Pursuant to 10 CFR 50.55a(a)(3)(i), relief is requested to use the enclosed Performance Demonstration Initiative (PDI) Program for implementation of Appendix VIII, Supplement 11 requirements.

BASIS FOR RELIEF

Paragraph 1.1(d)(1), requires that all base metal flaws be cracks. As illustrated below, 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 ultrasonic 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.



Relief is requested to allow closer spacing of flaws provided the flaws do not interfere with detection or discrimination. The existing specimens used to date for qualification to the Tri-party (Nuclear Regulatory Commission (NRC)/Boiling Water Reactor Owners Group (BWROG)/Electrical Power Research Institute (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 facilitate their use and provide continuity from 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 ultrasonic 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 one inch of the length of the overlaid weld, rather than three 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 one 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 one inch, rather than the six-square-inch requirement of Supplement 11.

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 one inch at both ends, rather than around its entire perimeter.

Additionally, 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 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 its 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.

These changes are contained in Code Case N-653. A comparison between the 2001 Edition of Supplement 11, Code Case N-653, and the PDI Program is attached as supporting documentation. The first column identifies the code requirements, while the second (middle) column identifies the changes made by the Code Case.

There are, however, some additional changes that were inadvertently omitted from Code Case N-653. In paragraph 1.1(e)(2)(a)(1) the phrase "and base metal on both sides" was inadvertently included in the description of a base metal grading unit. The PDI program intentionally excludes this requirement because some of the qualification samples include 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.

Additionally, to avoid confusion, the overlay thickness tolerance contained in paragraph 1.1(b) last sentence, was reworded and the phrase "*and the remainder shall be alternative flaws*" was added to the next to the last sentence in paragraph 1.1(d)(1). Additional editorial changes were made to the PDI program to address an earlier request for additional information. The changes described above are identified by **bold** print in the third column of the table 1.

ALTERNATIVE EXAMINATION

In lieu of the requirements of ASME Section XI, 2001 Edition, Appendix VIII, Supplement 11, the PDI Program shall be used.

PERIOD FOR WHICH RELIEF IS REQUESTED

DAEC requests approval of the proposed alternative for the Fourth Ten-Year Interval of the Inservice Inspection Program for the DAEC.

PRECEDENCE

This Relief Request was approved during the DAEC Third Ten-Year Interval Inspection Program (NDE-R032, TAC NO. MC2182)

TABLE 1
APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS

APPENDIX VIII SUPPLEMENT 11 Current Requirements	CODE CASE N-653 (Provided for Information Only)	PDI PROGRAM The Proposed Alternative to Supplement 11 Requirements
1.0 SPECIMEN REQUIREMENTS		
Qualification test specimens shall meet the requirements listed herein, unless a set of specimens is designed to accommodate specific limitations stated in the scope of the examination procedure (e.g., pipe size, weld joint configuration, access limitations). The same specimens may be used to demonstrate both detection and sizing qualification.	No Change	No Change
1.1 General. The specimen set shall conform to the following requirements.	No Change	No Change
(a) Specimens shall have sufficient volume to minimize spurious reflections that may interfere with the interpretation process.	No Change	No Change
(b) The specimen set shall consist of at least three specimens having different nominal pipe diameters and overlay thicknesses. They shall include the minimum and maximum nominal pipe diameters for which the examination procedure is applicable. Pipe diameters within a range of 0.9 to 1.5 times a nominal diameter shall be considered equivalent. If the procedure is applicable to pipe diameters of 24 in. or larger, the specimen set must include at least one specimen 24 in or larger but need not include the maximum diameter. The specimen set must include at least one specimen with overlay thickness within -0.1 in. to +0.25 in. of the maximum nominal overlay thickness for which the procedure is applicable.	No Change	(b) The specimen set shall consist of at least three specimens having different nominal pipe diameters and overlay thicknesses. They shall include the minimum and maximum nominal pipe diameters for which the examination procedure is applicable. Pipe diameters within a range of 0.9 to 1.5 times a nominal diameter shall be considered equivalent. If the procedure is applicable to pipe diameters of 24 in. or larger, the specimen set must include at least one specimen 24 in or larger but need not include the maximum diameter. The specimen set **shall include specimens with overlays not thicker than 0.1 in. more than the minimum thickness, or thinner that 0.25 in. of the maximum nominal overlay thickness for which the examination procedure is applicable.**
(c) The surface condition of at least two specimens shall approximate the roughest surface condition for which the examination procedure is applicable.	No Change	No Change
(d) *Flaw Conditions*		

TABLE 1 APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS		
APPENDIX VIII SUPPLEMENT 11 **Current Requirements**	**CODE CASE N-653** **(Provided for Information Only)**	**PDI PROGRAM** **The Proposed Alternative to Supplement 11 Requirements**
(1) Base metal flaws. All flaws must be cracks in or near the butt weld heat-affected zone, open to the inside surface, and extending at least 75% through the base metal wall. Flaws may extend 100% through the base metal and into the overlay material; in this case, intentional overlay fabrication flaws shall not interfere with ultrasonic detection or characterization of the cracking. Specimens containing IGSCC shall be used when available.	(1) Base metal flaws. All flaws must be in or near the butt weld heat-affected zone, open to the inside surface, and extending at least 75% through the base metal wall. Intentional overlay fabrication flaws shall not interfere with ultrasonic detection or characterization of the cracking. Specimens containing IGSCC shall be used when available. (a) At least 70 percent of the flaws in the detection and sizing tests shall be cracks. Alternative flaw mechanisms, if used, shall provide crack-like reflective characteristics and shall be limited by the following: (1) Flaws shall be limited to when implantation of cracks precludes obtaining a realistic ultrasonic response. (2) Flaws shall be semielliptical with a tip width of less than or equal to 0.002 inches.	(1) Base metal flaws. All flaws must be in or near the butt weld heat-affected zone, open to the inside surface, and extending at least 75% through the base metal wall. Intentional overlay fabrication flaws shall not interfere with ultrasonic detection or characterization of the **base metal flaws**. Specimens containing IGSCC shall be used when available. At least 70 percent of the flaws in the detection and sizing tests shall be cracks **and the remainder shall be alternative flaws**. Alternative flaw mechanisms, if used, shall provide crack-like reflective characteristics and shall be limited by the *following:* (a) **The use of Alternative flaws** shall be limited to when **the implantation of cracks produces spurious reflectors that are uncharacteristic of actual flaws.** (b) Flaws shall be semi-elliptical with a tip width of less than or equal to 0.002 inches.
(2) *Overlay fabrication flaws*. At least 40% of the flaws shall be non-crack fabrication flaws (e.g., sidewall lack of fusion or laminar lack of bond) in the overlay or the pipe-to-overlay interface. At least 20% of the flaws shall be cracks. The balance of the flaws shall be of either type.	**No Change**	**No Change**
(e) *Detection Specimens*		
(1) At least 20% but less than 40% of the flaws shall be oriented within ±20 deg. Of the pipe axial direction. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access. The rules of IWA-3300 shall be used to determine whether closely spaced flaws should be treated as single or multiple flaws.	(1) At least 20% but less than 40% of the base metal flaws shall be oriented within +20 deg. Of the pipe axial direction. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access.	(1)At least 20% but less than 40% of the base metal flaws shall be oriented within +20 deg. Of the pipe axial direction. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access.
(2) Specimens shall be divided into base and overlay grading units. Each specimen shall contain one or both types of grading units.	(2) Specimens shall be divided into base metal and overlay fabrication grading units. Each specimen shall contain one or both types of grading units. Flaws shall not interfere with ultrasonic detection or characterization of other flaws.	(2) Specimens shall be divided into base metal and overlay fabrication grading units. Each specimen shall contain one or both types of grading units. Flaws shall not interfere with ultrasonic detection or characterization of other flaws.
(a)(1) A base grading unit shall include at least 3 in. of the length of the overlaid weld. The base grading unit includes the outer 25% of the overlaid weld and base metal on both sides. The base grading unit shall not include the inner 75% of the overlaid weld and base metal overlay material, or base metal-to-overlay interface.	(a)(1) A base metal grading unit shall include at least 1 in. of the length of the overlaid weld. The base metal grading unit includes the outer 25% of the overlaid weld and base metal on both sides. The base metal grading unit shall not include the inner 75% of the overlaid weld and base metal overlay material, or base metal-to-overlay interface.	(a)(1) **A base metal grading unit includes the overlay material and the outer 25% of the original overlaid weld. The base metal grading unit shall extend circumferentially for at least 1 in. and shall start at the weld centerline and be wide enough in the axial direction to encompass one half of the original weld crown and a minimum of 0.50" of the adjacent base material.**

TABLE 1 APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS		
APPENDIX VIII SUPPLEMENT 11 **Current Requirements**	**CODE CASE N-653** **(Provided for Information Only)**	**PDI PROGRAM** **The Proposed Alternative to Supplement 11 Requirements**
(a)(2) When base metal cracking penetrates into the overlay material, the base grading unit shall include the overlay metal within 1 in. of the crack location. This portion of the overlay material shall not be used as part of any overlay grading unit.	(a)(2) When base metal cracking penetrates into the overlay material, the base metal grading unit shall not be used as part of any overlay fabrication grading unit.	(a)(2) When base metal **flaws penetrate** into the overlay material, the base metal grading unit shall not be used as part of any overlay fabrication grading unit.
(a)(3) When a base grading unit is designed to be unflawed, at least 1 in. of unflawed overlaid weld and base metal shall exist on either side of the base grading unit. The segment of weld length used in one base grading unit shall not be used in another base grading unit. Base grading units need not be uniformly spaced around the specimen.	(a)(3) Sufficient unflawed overlaid weld and base metal shall exist on all sides of the grading unit to preclude interfering reflections from adjacent flaws.	(a)(3) Sufficient unflawed overlaid weld and base metal shall exist on all sides of the grading unit to preclude interfering reflections from adjacent flaws.
(b)(1) An overlay grading unit shall include the overlay material and the base metal-to-overlay interface of at least 6 sq. in. The overlay grading unit shall be rectangular, with minimum dimensions of 2 in.	(b)(1) An overlay fabrication grading unit shall include the overlay material and the base metal-to-overlay interface for a length of at least 1 in.	(b)(1) An overlay fabrication grading unit shall include the overlay material and the base metal-to-overlay interface for a length of at least 1 in.
(b)(2) An overlay grading unit designed to be unflawed shall be surrounded by unflawed overlay material and unflawed base metal-to-overlay interface for at least 1 in. around its entire perimeter. The specific area used in one overlay grading unit shall not be used in another overlay grading unit. Overlay grading units need not be spaced uniformly about the specimen.	(b)(2) 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 in. at both ends. Sufficient unflawed overlaid weld and base metal shall exist on both sides of the overlay fabrication grading unit to preclude interfering reflections from adjacent flaws. The specific area used in one overlay fabrication grading unit shall not be used in another overlay fabrication grading unit. Overlay fabrication grading units need not be spaced uniformly about the specimen.	(b)(2) 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 in. at both ends. Sufficient unflawed overlaid weld and base metal shall exist on both sides of the overlay fabrication grading unit to preclude interfering reflections from adjacent flaws. The specific area used in one overlay fabrication grading unit shall not be used in another overlay fabrication grading unit. Overlay fabrication grading units need not be spaced uniformly about the specimen.
(b)(3) Detection sets shall be selected from Table VIII-S2-1. *The minimum detection* sample set is five flawed base grading units, ten unflawed base grading units, five flawed overlay grading units, and ten unflawed overlay grading units. For each type of grading unit, the set shall contain at least twice as many unflawed as flawed grading units.	(b)(3) Detection sets shall be selected from Table VIII-S2-1. *The minimum* detection sample set is five flawed base metal grading units, ten unflawed base metal grading units, five flawed overlay fabrication grading units, and ten unflawed overlay fabrication grading units. For each type of grading unit, the set shall contain at least twice as many unflawed and flawed grading units. For *initial procedure qualification*, detection sets shall include the equivalent of three personnel qualification sets. To qualify new values of essential variables, at least one personnel qualification set is required.	(b)(3) Detection sets shall be selected from Table VIII-S2-1. *The minimum detection sample set is* five flawed base metal grading units, ten unflawed base metal grading units, five flawed overlay fabrication grading units, and ten unflawed overlay fabrication grading units. For each type of grading unit, the set shall contain at least twice as many unflawed as flawed grading units. For initial procedure qualification, detection sets shall include the equivalent of *three personnel qualification sets. To qualify* new values of essential variables, at least one personnel qualification set is required.
(f) Sizing Specimen		

TABLE 1 APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS		
APPENDIX VIII SUPPLEMENT 11 **Current Requirements**	**CODE CASE N-653** **(Provided for Information Only)**	**PDI PROGRAM** **The Proposed Alternative to *Supplement 11 Requirements***
(1) The minimum number of flaws shall be ten. At least 30% of the flaws shall be overlay fabrication flaws. At least 40% of the flaws shall be cracks open to the inside surface.	(1) The minimum number of flaws shall be ten. At least 30% of the flaws shall be overlay fabrication flaws. At least 40% of the flaws shall be cracks open to the inside surface. For initial procedure qualification, sizing sets shall include the equivalent of three personnel qualification sets. To qualify new values of essential variables, at least one personnel qualification set is required.	(1) The minimum number of flaws shall be ten. At least 30% of the flaws shall be overlay fabrication flaws. At least 40% of the flaws **shall be open to the inside surface. Sizing sets shall contain a distribution of flaw dimensions to assess sizing capabilities.** For initial procedure qualification, sizing sets shall include the equivalent of three personnel qualification sets. To qualify new values of essential variables, at least one personnel qualification set is required.
(2) At least 20% but less than 40% f the flaws shall be oriented axially. The remainder shall be oriented circumferentially. Flaws shall not be open to any surface to which the candidate has physical or visual access.	**No Change**	**No Change**
(3) Base metal cracking used for length sizing demonstrations shall be oriented circumferentially.	**No Change**	(3) Base metal **flaws** used for length sizing demonstrations shall be oriented circumferentially.
(4) Depth sizing specimen sets shall include at least two distinct locations where cracking in the base metal extends into the overlay material by at least 0.1 in. in the through-wall direction.	**No Change**	(4) Depth sizing specimen sets shall include at least two distinct locations where **a base metal flaw extends** into the overlay material by at least 0.1 in. in the through-wall direction.
2.0 CONDUCT OF PERFORMANCE DEMONSTRATION		
The specimen inside surface and identification shall be concealed from the candidate. All examinations shall be completed prior to grading the results and presenting the results to the candidate. Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited.	The specimen inside surface and identification shall be concealed from the candidate. All examinations shall be completed prior to grading the results and presenting the results to the candidate. Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited. The overlay fabrication flaw test and the base metal flaw test may be performed separately.	The specimen inside surface and identification shall be concealed from the candidate. All examinations shall be completed prior to grading the results and presenting the results to the candidate. Divulgence of particular specimen results or candidate viewing of unmasked specimens after the performance demonstration is prohibited. The overlay fabrication flaw test and the base metal flaw test may be performed separately.
2.1 Detection Test		
Flawed and unflawed grading units shall be randomly mixed. Although the boundaries of specific grading units shall not be revealed to the candidate, the candidate shall be made aware of the type or types of grading units (base or overlay) that are present for each specimen.	Flawed and unflawed grading units shall be randomly mixed. Although the boundaries of specific grading units shall not be revealed to the candidate, the candidate shall be made aware of the type or types of grading units (base metal or overlay fabrication) that are present for each specimen.	Flawed and unflawed grading units shall be randomly mixed. Although the boundaries of specific grading units shall not be revealed to the candidate, the candidate shall be made aware of the type or types of grading units (base metal or overlay fabrication) that are present for each specimen.
2.2 Length Sizing Test		
(a) The length sizing test may be conducted separately or in conjunction with the detection test.	**No Change**	**No Change**

TABLE 1 APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS		
APPENDIX VIII SUPPLEMENT 11 **Current Requirements**	**CODE CASE N-653** **(Provided for Information Only)**	**PDI PROGRAM** **The Proposed Alternative to Supplement 11 Requirements**
(b) When the length sizing test is conducted in conjunction with the detection test and the detected flaws do not satisfy the requirements of 1.1(f), additional specimens shall be provided to the candidate. The regions containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the length of the flaw in each region.	No Change	No Change
(c) For a separate length sizing test, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the length of the flaw in each region.	No Change	No Change
(d) For flaws in base grading units, the candidate shall estimate the length of that part of the flaw that is in the outer 25% of the base wall thickness.	(d) For flaws in base metal grading units, the candidate shall estimate the length of that part of the flaw that is in the outer 25% of the base metal wall thickness.	(d) For flaws in base metal grading units, the candidate shall estimate the length of that part of the flaw that is in the outer 25% of the base metal wall thickness.
2.3 Depth Sizing Test		
For the depth sizing test, 80% of the flaws shall be sized at a specific location on the surface of the specimen identified to the candidate. For the remaining flaws, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. *The candidate shall determine the maximum* depth of the flaw in each region.	The candidate shall determine the depth of the flaw in each region.	**(a) The depth sizing test may be conducted separately or in conjunction with the detection test.**
		(b) When the depth sizing test is conducted in conjunction with the detection test and the detected flaws do not satisfy the requirements of 1.1(f), additional specimens shall be provided to the candidate. The regions containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region.
		(c) For a separate depth sizing test, the regions of each specimen containing a flaw to be sized shall be identified to the candidate. The candidate shall determine the maximum depth of the flaw in each region.
3.0 ACCEPTANCE CRITERIA		
3.1 Detection Acceptance Criteria		

TABLE 1 APPENDIX VIII SUPPLEMENT 11 REQUIREMENTS VS. PDI PROGRAM REQUIREMENTS		
APPENDIX VIII SUPPLEMENT 11 **Current Requirements**	**CODE CASE N-653** **(Provided for Information Only)**	**PDI PROGRAM** **The Proposed Alternative to Supplement 11 Requirements**
Examination procedures, equipment, and personnel are qualified for detection when the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for both detection and false calls. The criteria shall be satisfied separately by the demonstration results for base grading units and for overlay grading units.	Examination procedures are qualified for detection when all flaws within the scope of the procedure are detected and the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for false calls. Examination equipment and personnel are qualified for detection when the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for both detection and false calls. The criteria shall be satisfied separately by the demonstration results for base metal grading units and for overlay fabrication grading units.	**(a) Examination procedures are qualified for detection when;**
		(1) All flaws within the scope of the procedure are detected and the results of the performance demonstration satisfy the acceptance criteria of Table VII-S2-1 for false calls.
		(a) At least one successful personnel demonstration has been performed meeting the acceptance criteria defined in (b).
		(b) Examination equipment and personnel are qualified for detection when the results of the performance demonstration satisfy the acceptance criteria of Table VIII-S2-1 for both detection and false calls.
		(c) The criteria in (a), (b) shall be satisfied separately by the demonstration results for base metal grading units and for overlay fabrication grading units
3.2 Sizing Acceptance Criteria.		
Examination procedures, equipment, and personnel are qualified for sizing when the results of the performance demonstration satisfy the following criteria.	**No Change**	**No Change**
(a) The RMS error of the flaw length measurements, as compared to the true flaw lengths, is less than or equal to 0.75 inch. The length of base metal cracking is measured at the 75% through-base-metal position.	**No Change**	(a) The RMS error of the flaw length measurements, as compared to the true flaw lengths, is less than or equal to 0.75 inch. The length of base metal flaws is measured at the 75% through-base-metal position.
(b) All extensions of base metal cracking into the overlay material by at least 0.1 in. are *reported as being intrusions into the overlay* material.	**This requirement is omitted.**	**This requirement is omitted.**
(c) The RMS error of the flaw depth measurements, as compared to the true flaw depths, is less than or equal to 0.125 in.	*(b)* The RMS error of the flaw depth measurements, as compared to the true flaw depths, is less than or equal to 0.125 in.	*(b)* The RMS error of the flaw depth measurements, as compared to the true flaw depths, is less than or equal to 0.125 in.

(3) Relief Request NDE-R003

ASME Code Components Affected

Code Class:	1
References:	IWB-2500, Table IWB-2500-1
Examination Categories:	B-A
Item Number:	B1.11
Description:	Relief from Volumetric Examination of All Pressure Retaining Reactor Pressure Vessel Shell Circumferential Welds Class I
Component Numbers:	VCB-B001, VCB-A002, VCB-B003, and VCB-B004

Applicable Code Edition and Addenda

Fourth interval examinations will be performed per the requirements of ASME Section XI, 2001 Edition through the 2003 Addenda, as amended by 10 CFR 50.55a.

Per 10 CFR 50.55a(b)(2)(xxiv), the use of Appendix VIII and supplements to Appendix VIII of Section XI of the 2002 Addenda through the 2003 Addenda is prohibited. Therefore, for Appendix VIII and supplements to Appendix VIII the 2001 Edition of Section XI (no addenda) will be used.

Applicable Code Requirement

DAEC requests relief from the inspection of Reactor Vessel Circumferential (B-A) Welds, Item B1.11, for the remaining term of the current license for the DAEC.

In accordance with the provisions of 10 CFR 50.55a(a)(3)(i), DAEC requests permanent relief for the remaining term of the operating license for the DAEC from the following requirements:

a. Volumetric examination of all RPV shell circumferential welds in the Reactor Pressure Vessel in accordance with the requirements of ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, 2001 Edition through the 2003 Addenda, Examination Category B-A, Item B1.11.

b. Successive Inspections for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 2001 Edition through the 2003 Addenda, Paragraph IWB-2420.

c. Additional Examinations for RPV shell circumferential welds in accordance with the requirements of ASME Section XI, 2001 Edition through the 2003 Addenda, Paragraph IWB-2430.

Reason for Request

DAEC requests this relief to reduce inspections and conserve radiological dose, while still maintaining an acceptable level of quality and safety for examination of the affected welds.

Proposed Alternative and Basis for Use

I. Alternative Provisions:

Pursuant to 10 CFR 50.55a(a)(3)(i), the DAEC will implement the following alternate provisions for the subject weld examinations. Unless stated otherwise, all references to the ASME code are to ASME Section XI, 2001 Edition through the 2003 Addenda.

a. Inservice Inspection Scope

The failure frequency for ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Reactor Pressure Vessel Shell Circumferential Welds, is sufficiently low to justify their elimination from the ISI requirement of 10 CFR 50.55a(g) based on the NRC Safety Evaluation (Reference 2).

The ISI examination requirements of ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12 Reactor Pressure Vessel Shell Longitudinal," shall be performed, to the extent possible, and shall include inspection of the circumferential welds at the intersection of these welds with the longitudinal welds, or approximately 2 to 3% of the RPV shell circumferential welds.

The procedures for these examinations shall be qualified such that flaws relevant to reactor pressure vessel integrity can be reliably detected and sized, and the personnel implementing these procedures shall be qualified in the use of the procedures.

b. Successive Examination of Flaws

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, "Reactor Pressure Vessel Shell Circumferential Welds," at intersections with longitudinal welds, successive examinations per IWB-2420 "Successive Inspections," are not required for non-threatening flaws such as embedded flaws from material manufacturing or vessel fabrication which experience negligible or no growth during the design life of the vessel, provided that the following conditions are met:

1. The flaw is characterized as subsurface in accordance with BWR Vessel and Internals Project Report, BWRVIP-05, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (Reference 1).

2. The NDE technique and evaluation that detected and characterized the flaw as originating from material manufacture or vessel fabrication is documented in a flaw evaluation report, and

3. The vessel containing the flaw is acceptable for continued service in accordance with ASME Section XI, IWB-3600, "Analytical Evaluation of Flaws," and the flaw is demonstrated acceptable for the intended service life of the vessel.

For ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, "Reactor Pressure Vessel Shell Longitudinal Welds," all flaws shall be re-inspected at successive intervals consistent with ASME Code and regulatory requirements.

c. Additional Examinations of Flaws

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.11, Reactor Pressure Vessel Shell Circumferential Welds, at the intersection with longitudinal welds, additional requirements per ASME Section XI, IWB-2430, Additional Examinations, are not required for flaws provided the following conditions are met:

1. If the flaw is characterized as subsurface in accordance with BWRVIP-05 then no additional examinations are required.

2. If the flaw is not characterized as subsurface in accordance with BWRVIP-05 then an engineering evaluation shall be performed, addressing the following as a minimum:
A determination of the root cause of the flaw,
An evaluation of any potential failure mechanisms,
An evaluation of service conditions which could cause subsequent failure,
An evaluation per ASME Section XI, IWB-3600 demonstrating that the vessel is acceptable for continued service.

3. If the flaw meets the criteria of ASME Section XI, IWB-3600 for intended service life of the vessel, then additional examinations may be limited to those welds subject to the same root cause conditions and failure mechanisms, up to the number of examinations required by ASME Section XI, IWB-2430(a). If the engineering evaluation determines that there are no additional welds subject to the same root cause conditions or no failure mechanism exists, then no additional examinations are required.

For ASME Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.12, additional examinations for flaws shall be in accordance with ASME Section XI, IWB-2430, "Additional Examinations." All flaws in RPV shell longitudinal shell welds shall require additional examinations consistent with the ASME Section XI Code and regulatory requirements. Examination of the circumferential shell welds shall be performed if longitudinal (axial) weld examinations reveal an active, mechanistic mode of degradation.

Basis for Relief

Augmented Exam

A September 8, 1992 revision to 10 CFR 50.55a(g)(6)(ii)(A) contains an augmented examination requirement to perform a one time volumetric examination of essentially 100% (>90%) of all circumferential and axial reactor pressure vessel (RPV) shell assembly welds. This rule revoked previously granted relief requests regarding the extent of volumetric examination on ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item No. B1.10 circumferential and longitudinal reactor pressure vessel shell welds. 10 CFR 50.55a(g)(6)(ii)(A) required the augmented

examinations to be performed as specified in the ASME Code Section XI (1995 Edition with the 1996 Addenda).

During refueling outage (RFO) 14 in 1996, the DAEC performed the augmented weld examination of the reactor vessel using the General Electric GERIS 2000 ultrasonic examination system. At the DAEC, the volumetric examinations of the reactor pressure vessel shell circumferential welds were performed from the vessel outside diameter using a composite of automated and supplemental manual Ultrasonic (UT) examination techniques; no reportable indications were found.

Complete examination of the subject welds was not obtained due to scanning limitation and access restrictions from various reactor pressure vessel appurtenances and containment structures. For circumferential weld VCB-B001, an examination coverage of 96.5% was obtained, for VCB-A002, an examination coverage of 96.7% was obtained, for VCB-B003, 96.7% was obtained and for VCB-B004, 86.91% was obtained. The examination coverage for VCB-B004 (the Course 3 to Course 4 circumferential weld) was limited due to the presence of vessel stabilizers and an insulation support ring. The insulation support ring is located 18" from the weld. The bottom of the stabilizer brackets are located on the weld. By letter dated October 18, 1999, the NRC granted relief from the requirement to perform an examination of essentially 100% of the weld length for VCB-B004.

GL 98-05

The technical justification for this request for inspection relief is documented in the report BWR Vessel and Internals Project, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995 (Reference 1). The NRC evaluated this report and responses to Requests for Additional Information, and issued Safety Evaluations to the BWRVIP (References 2 and 3).

On November 10, 1998, the NRC issued Generic Letter (GL) 98-05, Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds, (Reference 4). This GL stated that BWR licensees may request permanent (i.e., for the remaining term of operation under the existing, initial, license) relief from the inservice inspection requirements of 10 CFR 50.55a(g) for the volumetric examination of circumferential reactor pressure vessel welds (ASME Code Section XI, Table IWB-2500-1, Examination Category B-A, Item 1.11, Circumferential Shell Welds) by demonstrating that: (1) at the expiration of their license, the circumferential welds will continue to satisfy the limiting conditional failure probability for circumferential welds in the Staff's July 30, 1998, safety evaluation, and (2) licensees have implemented operator training and established procedures that limit the frequency of cold over-pressure events to the amount specified in the staff's July 30, 1998, safety evaluation. Licensees will still need to perform their required inspections of "essentially 100 percent" of all axial welds.

Although BWRVIP-05 provides the technical basis supporting the relief request, the following information is provided to show the conservatism of the NRC analysis relative to the DAEC reactor pressure vessel.

<u>Criterion 1, Demonstrate that at the expiration of the license, the RPV shell circumferential welds will continue to satisfy the limiting conditional failure probability for RPV shell circumferential welds that is established in the July 30, 1998 Safety Evaluation.</u>

The NRC evaluation of BWRVIP-05 utilized a probabilistic fracture mechanics (PFM) analysis to estimate the RPV shell weld failure probabilities. Three key assumptions of the PFM analysis are: (1) the neutron fluence used was the estimated end-of-life mean fluence; (2) the chemistry values are mean values based on vessel types; and (3) the potential for beyond-design-basis events is considered.

The following table illustrates that the DAEC reactor pressure vessel has additional conservatism in comparison to Table 2.6-4 for the Limiting Plant-Specific Analyses (32 effective full power years (EFPY)) of the NRC's evaluation of BWRVIP-05.

Effects of Irradiation on RPV Circumferential Weld Properties
Duane Arnold Energy Center

Parameter Description	DAEC Comparative Parameters At 32 EFPY for the Bounding Circumferential Weld Wire Heat/Lot 07L669 Lot K004A27A	USNRC Limiting Plant Specific Analysis Parameters at 32 EFPY SER Table 2.6-4***
Copper (Cu), wt%	0.03	0.10
Nickel (Ni), wt%	1.02	0.99
Chemistry Factor (CF)	41	134.9*
End of Life (EOL) Inside Diameter (ID) Fluence, $x10^{19}$ n/cm^2	0.355**	0.51
Initial (unirradiated) Reference Temperature $RT_{NDT(U)}$, °F	-50	-65
Increase in Reference Temperature ΔRT_{NDT}, °F	26.4	109.5
Mean (irradiated) Reference Temperature $RT_{NDT(U)} + \Delta RT_{NDT}$, °F	-23.6	44.5

*Revised value from the NRC SE Supplement (Reference 3).

**By Amendment 253 (Reference 5), the NRC approved revised Reactor Coolant System Pressure-Temperature curves for the DAEC. As discussed in the Safety Evaluation that accompanied the Amendment, the replacement curves were generated using an NRC-approved methodology (General Electric Report NEDC-32983PA, Revision 1, "Licensing Topical Report, General Electric Methodology for Reactor Pressure Vessel Fast Neutron Fluence Evaluations," December 2001) for determining the neutron fluence on the Reactor Pressure Vessel (RPV). The methodology used for the RPV fluence calculation is described in GE Report NEDC-32983PA. This methodology follows the guidance in RG 1.190 and has been approved by the NRC staff by letter dated September 14, 2001.

***The DAEC RPV was supplied and erected by the Chicago Bridge and Iron Company.

As shown in the Table, the nickel content for the DAEC bounding weld is slightly higher than the value used in the NRC analysis, however, the values for DAEC copper content and chemistry factor are considerably lower than the values used in the NRC analysis. The unirradiated reference temperature is higher than that used in the NRC analysis. The calculated 32 EFPY fluence for the DAEC is lower than the NRC estimated values. The overall result for the DAEC is a lower calculated mean reference temperature than the NRC analysis mean reference temperature value.

Since the mean (irradiated) reference temperature value for the DAEC RPV shell weld is less than the mean (irradiated) reference temperature value for its corresponding limiting plant reference case study (as shown in the Table), the shell weld is considered to have less embrittlement than the corresponding weld in the case study, and therefore to have a conditional probability of failure less than or equal to that calculated for the reference case study. The RPV shell circumferential weld failure probabilities are bounded by the conditional failure probability, P(FIE), in Table 2.6-4 of the NRC Safety Evaluation through the initial end of license.

This demonstrates that at expiration of the existing license, the circumferential welds of the DAEC RPV will continue to satisfy the limiting conditional failure probability for circumferential welds in the Staff's SE dated July 30, 1998.

Criterion 2, Licensees have implemented operator training and established procedures that limit the frequency of cold overpressure events to the amount specified in the July 30, 1998 Safety Evaluation.

As discussed below, the DAEC has procedures in place that guide operators in controlling and monitoring reactor pressure during all phases of operation. Use of the guidance provided in the operating procedures will prevent a Low Temperature Over-Pressurization (LTOP) event. Also, these procedures are reinforced through operator training, and system design features provide additional insurance against an LTOP event.

High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC)

Both HPCI and RCIC are high-pressure, steam driven systems. These systems use steam-turbine driven pumps to deliver emergency coolant to the RPV. The steam that is used to drive the turbines and actuate the pumps is delivered through the turbine steam supply line, which discharges from the main steam lines of the plant. Since the reactor does not deliver steam to the main steam lines during cold shutdown, the HPCI and RCIC systems will not cause a cold-overpressurization event while DAEC is in the cold shutdown operating mode.

Feedwater/Condensate Systems

The feedwater/condensate system is a potential source of high-pressure injection into the reactor vessel. The condensate pumps are a source of water to the reactor feed pumps. The reactor feed pumps provide water to the vessel. A system design feature of the reactor feed pumps is an automatic trip of all feed pumps on high vessel water level (+211 inches).

With respect to injection by an inadvertent start of a feedwater pump, injection of feedwater with vessel water level greater than +211 inches is controlled by a high water interlock. This interlock prevents operation or starts of the feedwater pumps when water level in the vessel is equal to or greater than +211 inches. Defeating this interlock is procedurally and administratively controlled to prevent inadvertently injecting feedwater into the vessel. The DAEC has high reactor water level and high reactor pressure alarms in the control room. These provide further assurance that an LTOP event will not occur.

The condensate and feed water pumps are used to control vessel level during startup. The startup procedure requires monitoring of reactor vessel temperatures and pressures. The reactor head vents are not closed until the coolant temperature is greater than 212°F. This administrative action for head vent closure serves as a mechanism to reduce the likelihood of over pressurization at low temperature. Monitoring of reactor temperature, pressure, and cool down rates, are prescribed in procedures and Technical Specifications.

A low temperature over pressurization event due to injection by the feedwater/condensate systems is very unlikely since strict controls on temperature and pressure are imposed by procedures. An unexpected change in reactor water level would allow for operator action. Therefore, these systems do not present a significant potential for over pressurization.

Standby Liquid Control (SBLC)

SBLC is another high-pressure water source to the reactor pressure vessel. SBLC is designed with two redundant trains of SBLC piping, each designed with an associated key-lock switch, piston-driven delivery pump and explosive squib-type discharge valve, each delivering to a common header to the RPV. No automatic starts are associated with this system; operator action is needed to manually start the system by a key-lock switch; therefore, inadvertent manual initiation of SBLC is an unlikely event.

Procedures have been developed for operation of the SBLC system and operators are trained on the system operation. The injection rate of one SBLC pump is approximately 26.2 gpm; the injection rate of two SBLC pumps is approximately 52.4 gpm. These low flow rates would provide DAEC operators ample time to control reactor pressure in the case of an inadvertent injection of SBLC. Therefore, this system does not present a significant potential for over-pressurization.

Residual Heat Removal (RHR) System, Low Pressure Coolant Injection (LPCI), Core Spray (CS)

The shutoff head for the DAEC Core Spray pumps is about 330 psig, and for the Residual Heat Removal pumps is about 260 psig. An inadvertent injection of LPCI or CS would be detected by operations and the injection would be terminated, based on observation and alarm of reactor vessel level. In addition, during cold shutdown when the reactor head is tensioned, a cold overpressure event is prevented by the operating shutdown procedure, which requires the operator to place the RPV head vent valves in an open position when reactor coolant temperatures are below 212°F. A Core Spray pump may be used for reactor vessel and cavity-fill during refueling outages. Under these conditions the reactor vessel head is removed which will prevent over pressurization.

A condensate, CS, or RHR pump may be used to inject into the RPV in the event of a loss of shutdown cooling. Abnormal Operating Procedure (AOP) 149, Loss of Decay Heat Removal, includes guidance on performing a feed and bleed to the torus via a safety relief valve (SRV). The handswitch for an SRV is placed in the open position. A condensate, CS or RHR pump is used to inject water into the RPV until a safety relief valve (SRV) is open and RPV pressure is about 50 psig above Torus pressure, but as low as practical. Coolant then exits the reactor vessel and flows to the torus via the SRV discharge line. In this situation, the open SRV prevents an overpressure event.

Control Rod Drive (CRD) and Reactor Water Cleanup (RWCU)

The CRD and RWCU systems are used to control RPV water level and pressure during cold shutdown conditions using a feed and bleed process. The low flow rate of these pumps allows sufficient time for operator action to react to unanticipated level changes and thus pressure changes. Therefore, these systems do not present a significant potential for over pressurization.

The CRD and RWCU systems are also used in the performance of RPV pressure and hydrotests. The pressure test procedures for the DAEC contains additional requirements to aid in the prevention of a low temperature over-pressurization event. The Class 1 System Leakage Test is performed at the conclusion of each outage, while the Hydrostatic Pressure Test (based on Code Case N498-1) is performed once every ten years. The leakage and hydrotests are considered to be infrequently performed, complex tasks and a requirement is included in them for a briefing with essential personnel. This briefing details the anticipated testing evolution with special emphasis on conservative decision making, plant safety awareness, lessons learned from similar in-house or industry operating experiences, the importance of open communications, and finally, the process in which the test would be aborted if plant systems responded in an adverse manner. Vessel temperature and pressure are required to be monitored throughout these tests to ensure compliance with the Technical Specification Pressure-Temperature (P-T) Curve.

As discussed in the NRC SE of the BWRVIP-05, the risk of cold over pressurization due to CRD injection may be higher if a loss of station power occurs during the pressure test, since the RWCU and CRD pumps would lose their power. If the operator restarts the CRD pumps but does not restore the RWCU, cold CRD flow would accumulate in the lower head region and, without further operator action, the pressure will increase. The beltline region would, nonetheless, stay near the original 200-degree level, maintaining the beltline P-T limits. To preclude this from occurring, special precautions are included in the surveillance test procedure for the DAEC pressure test. One precaution states that in the event of an interruption of offsite power, open CV-2729 (Cleanup System Drain Header Control Valve)

and allow the system to depressurize. This will preclude RPV over-pressurization as a result of closure of CV-2729 should control air pressure be lost. Another precaution instructs the operators to immediately trip the CRD pump if RWCU isolates. These actions provide additional protection against an LTOP event.

Reactor Operator Training

Simulator training is conducted on start-up and shut down scenarios in accordance with approved procedures, providing opportunities for the operators to perform RPV pressure and level control. Procedural controls for reactor temperature, water level, and pressure are an integral part of Operator training. Specifically, operators are trained in methods of controlling RPV water level within specified limits, as well as responding to abnormal RPV water level conditions outside the established limits. Plant-specific procedures have been developed to provide guidance to the operators regarding compliance with the Technical Specification requirements on pressure-temperature limits.

Work Control Process

During plant outages, work control procedures require that the outage schedule and changes to the schedule receive a risk assessment review commensurate with their safety significance. Senior Operations personnel provide input to the outage schedule to avoid conditions that could adversely impact reactor water level, pressure, or temperature. Schedules are issued listing the work activities to be performed.

During refueling outages, work is coordinated through the Outage Control Center. In the Control Room, the Shift Manager is required, by procedure, to maintain cognizance of any activity that could potentially affect reactor water level or decay heat removal. The Control Room Operator is required to provide positive control of reactor water level and pressure within the specified bands, and promptly report when operating outside the specified band, including restoration actions being taken. Cognizant individuals involved in the work activity attend pre-job briefings. Expected plant responses and contingency actions to address unexpected conditions, or responses that may be encountered, are included in the briefing discussion.

<u>**Conclusion**</u>

In summary, DAEC has reviewed the methodology used in BWRVIP-05 (Reference 1), and considered DAEC-specific materials properties and fluence, operational practices, the provisions of the NRC Safety Evaluation Report (Reference 2), and GL 98-05. DAEC's operational and procedural controls provide sufficient assurance that it is unlikely that a cold overpressure transient will occur at the DAEC. The probabilistic failure analysis of the circumferential welds in the DAEC RPV, when taken in conjunction with DAEC's operational and procedural controls to prevent cold-overpressurization events, provides an acceptable level of quality and safety in lieu of actually performing the volumetric inspections of the circumferential welds as required by ASME Boiler and Pressure Vessel Code, Section XI, Examination Category B-A, Inspection Item B1.11.

<u>**Duration of Proposed Alternative**</u>

This Relief Request was approved for the remaining term of the current DAEC operating license (TAC NO. MC2181).

Precedent

LaSalle County Station, Units 1 and 2 - Relief Request CR-38, Shell Weld Inspection (TAC Nos. MB9755 AND MB9756), from A. Mendiola (NRC) to J. Skolds (Exelon Nuclear) dated January 28, 2004, Docket Nos.: 50-373 and 50-374

References

1. EPRI Report TR-105697, BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations (BWRVIP-05), dated September 1995.

2. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "BWR Reactor Pressure Vessel Shell Weld Inspection Recommendations, BWRVIP-5" (TAC No. M93925), July 28, 1998.

3. NRC Safety Evaluation Report of Topical Report by the Boiling Water Reactor Vessel and Internals Project: "Supplement to Final Safety Evaluation of the BWR Vessel and Internals Project BWRVIP-5 Report (TAC No. MA3395)," March 7, 2000.

4. Generic Letter 98-05, Boiling Water Reactor Licensees Use of the BWRVIP-05 Report to Request Relief from Augmented Examination Requirements on Reactor Pressure Vessel Circumferential Shell Welds, dated November 10, 1998.

5. Amendment 253 to DAEC Technical Specifications Regarding Pressure and Temperature Limit Curves (TAC No. MB8750), by letter dated August 25, 2003, D. Hood (NRC) to M. Peifer (NMC).

CURRENT STATUS

Relief Request was submitted in letter NG-04-0103. Relief was authorized in SER dated January 6, 2005 for the remaining term of the operating license for the DAEC.

(4) Relief Request NDE-R004

COMPONENT IDENTIFICATION

Code Class:	Vessel Welded Attachments
References:	Table IWB-2500-1, Examination Category B-K, Footnote 4; IWC-2500-1, Examination Category C-C, Footnote 4; and Table IWD-2500-1, Examination Category D-A, Footnote 3 of the 2001 Edition, 2003 Addenda of ASME Section XI
Examination Categories:	B-K, C-C, D-A
Item Number:	B10.10, C3.10, D1.10
Description:	Relief is requested to use the alternatives of Code Case N-700 for the selection of Class 1, 2, and 3 vessel welded attachments for examination.

CODE REQUIREMENT

ASME Code, Section XI, Examination Category B-K, Footnote 4 and Examination Category C-C, Footnote 4 state that for multiple vessels of similar design, function, and service, only one of the multiple vessels shall be selected for a surface examination.

ASME Code, Section XI, Examination Category D-A, Footnote 3 states that selected samples of welded attachments shall be examined each inspection interval. All welded attachments selected for examination shall be those most subject to corrosion, as determined by the Owner, such as the welded attachments of the Service Water or Emergency Service Water systems. For multiple vessels of similar design, function, and service, the welded attachments of only one of the multiple vessels shall be selected for examination. For welded attachments of piping, pumps, and valves, a 10% sample shall be selected for examination. This percentage sample shall be proportional to the total number of nonexempt welded attachments connected to the piping, pumps, and valves in each system subject to these examinations.

Therefore, relief is being requested from the following requirements of ASME Code, Section XI:

- Table IWB-2500-1, Examination Category B-K, Footnote 4;
- Table IWC-2500-1, Examination Category C-C, Footnote 4; and
- Table IWD-2500-1, Examination Category D-A, Footnote 3

BASIS FOR RELIEF

Code Case N-509, "Alternative Rules for the Selection and Examination of Class 1, 2, and 3 Integrally Welded Attachments, Section XI, Division 1," was incorporated in the 1995 Edition, 1995 Addenda of ASME Section XI. The technical basis for development of Code Case N-509 concluded that operational transients/water hammers to be the major potential for welded attachment failures (possibility exists for corrosion related failures). The technical basis of Code Case N-509 also concluded that welded attachment failures have been identified as a result of connected support member deformation and had not been identified by the Section XI examinations. This is the basis for Code Case N-509 and the 1995 Addenda, and later addenda, which require welded attachments to be examined whenever component support deformation is identified. In addition, a sampling plan for welded attachments was maintained. Code Case N-509 and the 2001 Edition through 2003 Addenda require in Examination Categories B-K

and C-C that "For multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessels shall be selected for examination." There is no criterion for selection of the one welded attachment that must be examined. Code Case N-509 and the 2001 Edition through 2003 Addenda do not specifically address selection criteria for single vessels.

Code Case N-700 utilizes the basis for development of Code Case N-509 to provide criteria for selection of Class 1, 2, and 3 vessel welded attachments for examination. Code Case N-700 requires that for multiple vessels of similar design, function and service, only one welded attachment of only one of the multiple vessels shall be selected for examination. The code case requires that only one welded attachment on a single vessel be examined. However, the case also requires that the attachment selected for examination on one of the multiple vessels or the single vessel, as applicable, be an attachment under continuous load during normal system operation if such an attachment exists or an attachment subject to a potential intermittent load during normal system operation if an attachment under continuous load does not exist.

ALTERNATIVE EXAMINATION

In lieu of the requirements specified in the ASME Section XI, Code case N-700 will be used for selection of Class 1, 2, and 3 vessel welded attachments for examination. Code case N-700 was approved by the ASME Code Committee on November 18, 2003.

JUSTIFICATION FOR GRANTING RELIEF

Code Case N-509 was incorporated in the 1995 Edition, 1995 Addenda of ASME Section XI. The technical basis for development of Code Case N-509 concluded that operational transients/water hammers are the major potential for welded attachment failures (possibility exists for corrosion related failures). The technical basis of Code Case N-509 also concluded that welded attachment failures have been identified as a result of connected support member deformation and have not been identified by the code at that time. This is the basis for Code Case N-509 and the 1995 Addenda, and later addenda, which require welded attachments to be examined whenever component support deformation is identified. In addition, a sampling plan for welded attachments was maintained. For Class 1 and 2 multiple vessels this sampling plan requires only one welded attachment of only one of the multiple vessels shall be selected for examination. There is no criterion for selection of the one welded attachment that must be examined. Code Case N-509 and the 2003 Addenda do not specifically address selection criteria for single vessels.

Code Case N-700 utilizes the basis for development of Code Case N-509 to provide criteria for selection of Class 1, 2, and 3 welded attachments for examination. Code Case N-700 requires that for multiple vessels of similar design, function, and service, only one welded attachment of only one of the multiple vessels shall be selected for examination. The case requires that only one welded attachment on a single vessel be examined. However, the case also requires that the attachment selected for examination on one of the multiple vessels or the single vessel, as applicable, be an attachment under continuous load during normal system operation if such an attachment exists or an attachment subject to a potential intermittent load during normal system operation if an attachment under continuous load does not exist. Pursuant to 10 CFR 50.55a(a)(3)(i), approval is requested to use the proposed alternatives described above in lieu of the requirements specified in the Code. Compliance with the proposed alternatives will provide an

adequate level of quality and safety for selection of the Class 1, 2, and 3 vessel welded attachments for examination.

IMPLEMENTATION SCHEDULE

The alternative will be used for DAEC until the end of the fourth inspection interval.

PRECEDENCE

The NRC approved Wolf Creek Nuclear Station Relief Request RR I3R-02 on May 10, 2006 to allow the use of Code Case N-700 for their third 10-year Inservice Inspection Interval. (TAC NO. MD0298)

(5) Relief Request NDE-R005

COMPONENT IDENTIFICATION

Class:	1 and 2
References:	IWB-2500-1
Examination Category:	Class 1 B-F & B-J, and Class 2 C-F-2 Welds
Item Numbers:	B5.10, B5.20, B5.30
	B9.11, B9.21, B9.31, B9.32, B9.40, C5.51, C5.81
Component Numbers:	Various

CODE REQUIREMENT

ASME Code Section XI 2001 Edition with 2003 Addenda, IWB-2500-1 requires in part that for each successive 10-Year ISI Interval, 100% of Category B-F welds for the ASME Class 1 piping 4" NPS and greater be selected for volumetric and surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 100% of Category B-F welds for the ASME Class 1 piping less than 4" NPS be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 100% of Category B-F socket welds for the ASME Class 1 piping be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year ISI Interval, 25% of Category B-J welds for the ASME Class 1 piping 4" NPS and greater be selected for volumetric and surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 25% of Category B-J welds for the ASME Class 1 piping less than 4" NPS be selected for surface examination. IWB-2500-1 requires in part that for each successive 10-Year Interval, 25% of Category B-J socket welds for the ASME Class 1 piping be selected for surface examination. IWC-2500-1 requires in part that for each successive 10-Year Interval, 7.5% of C-F-2 welds be examined for ASME Class 2 piping greater than 4" NPS and 3/8" or greater nominal wall thickness for volumetric and surface examination. IWC-2500-1 requires in part that for each successive 10-Year Interval, 7.5% of C-F-2 welds be examined for ASME Class 2 piping 2" NPS or less for surface examination.

REASON FOR RELIEF REQUEST

Section XI, Examination Categories B-F and B-J currently contain the requirements for the non-destructive examination (NDE) of Class 1 piping components. Section XI, Examination Category C-F-2 currently contains the requirements for the NDE of Class 2 piping components. The previously approved Risk Informed Inservice Inspection (RI-ISI) Program (Reference 1) will be substituted for Class 1 and Class 2 piping (Examination Categories B-F, B-J, and C-F-2) in accordance with 10 CFR 50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. For example, existing pressure testing requirements remain unchanged.

BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS

Pursuant to 10 CFR 50.55a(a)(3)(i), NRC approval of the DAEC RI-ISI as an alternative to the current 2001 Edition through 2003 Addenda, ASME Section XI inspection requirements for Class 1 and Class 2 Code Examination Category B-F, B-J, and C-F-2 piping welds is requested. This request is to extend the relief previously granted to include the Fourth Interval.

The DAEC RI-ISI Program has been developed in accordance with the Electric Power Research Institute (EPRI) methodology contained in EPRI Topical Report TR-112657 Revision B-A, "Risk-Informed Inservice Inspection Evaluation Procedure'" (Reference 2). It was approved for use at DAEC during the 2nd and 3rd Periods of the 3rd Inspection Interval and is requested to be applicable for the 4th Inspection Interval. The DAEC specific RI-ISI program is summarized in Table 1. This Table reflects the recommended approach as provided in the Nuclear Energy Institute (NEI) 04-05 "Living Program Guidance To Maintain Risk-Informed Inservice Inspection Programs For Nuclear Piping Systems" (April 2004) for requesting relief to continue the RI-ISI program into the next inspection interval. This Table shows the final consequence ranking has not changed for individual line segments, and therefore the change in risk assessment for the new inspection interval as compared to the original RI-ISI submittal meets the acceptance criteria of the original RI-ISI submittal. The RI-ISI program was updated after a rigorous review of inputs and technical elements of the original submittal consistent with the intent of NEI-04-05 (Reference 3) and continues to meet EPRI TR-112657 and Reg. Guide 1.174 risk acceptance criteria. The current Class 1 and 2 piping weld scope is consistent with the submitted scope approved for the 3rd Interval ISI Program as described in Reference 1. The original list DAEC intended to credit for Class 1 or 2 RI-ISI piping weld exams has been substituted on specific occasions with similar welds due to accessibility issues that would have resulted in reduced exam volumes. DAEC chooses welds for examination that are classified within the same risk matrix classification segment, using the same treatment criteria as those originally selected in the first submittal. Socket welds that are chosen by the RI-ISI program for exam will be subjected to VT-2 exams as described by Code Case N-578-1. Welds chosen based on risk consequence alone will be volumetrically examined per ASME Section XI Code 2001Edition through the 2003 Addenda requirements for B-F, B-J, or C-F-2 welds depending on weld type.

The 3rd Interval RI-ISI program required DAEC to complete 38.7 % of the Section XI exams in the 1st Period and the remaining 61.3% of the RI-ISI program welds were to be completed by the end of the 3rd Inspection Interval. This Relief Request is to align the RI-ISI Interval and Code Year with the 4th Interval ISI Program. Therefore, 100% of the RI-ISI Program weld examinations will be completed in the 4th Inspection Interval.

All PRA inputs reported in the RI-ISI relief are derived from the Revision 5B PRA model, which was completed in February of 2005. The base core damage frequency value from this model, excluding internal flooding initiated sequences, is 1.10E-05 per year. This same Revision 5B PRA model was used as input to the Mitigating Systems Performance Index (MSPI).

Because of its on-going use as a decision-making tool, the DAEC PRA has been through a peer review as part of the BWR Owners' Group PRA certification program. The peer review team concluded that all of the graded elements are of sufficient detail and quality to support a risk significance determination supported by deterministic insights. The review team also commented on the DAEC's excellent PRA documentation and very consistent level of quality across all elements of the certification. Key PRA parameters, including train and component PRA importance parameters calculated for the MSPI, have been subjected to a cross-comparison study performed by the Integrated Risk Informed Regulation (IRIR) Committee of the BWR Owners' Group (Reference: NEDO-33215, GE Nuclear Energy, "BWR Owners' Group MSPI Cross Comparison Preliminary Results," September 2005.) None of the DAEC systems scoped for MSPI are identified as candidate outliers. This provides a reasonable level of confidence that

the DAEC PRA model is adequate for use in the MSPI application. Since there is an overlap between systems evaluated in the Risk-Informed Inservice Inspection application and those monitored in the MSPI program, the PRA cross comparison effort, although performed specifically for the MSPI application, provides confidence that the PRA model is of sufficient quality that it may be used for the Risk-Informed Inservice Inspection application. The final step in the cross-comparison process for the MSPI application was a high level screening of PRA metrics. Values reported for the DAEC in this step were consistent with calculated MSPI parameters reported for comparison purposes in previous steps, again indicating the accuracy of the PRA model for other applications and in particular for use on the Risk-Informed Inservice Inspection application.

IMPLEMENTATION SCHEDULE

Relief is requested for extension into the Fourth Ten-Year Interval of the DAEC Inservice Inspection Program.

PRECEDENTS

USNRC previously approved the DAEC RI-ISI program via Reference 1.

ATTACHMENTS

1. Table 1, "Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657, Rev B-A by Risk Category."
2. Table 2, "System Selection and Segment/Element Definition"

REFERENCES

1. USNRC Letter dated January 17, 2003 " Duane Arnold Energy Center – Risk Informed Inservice Inspection Program" (TAC No. MB4751).
2. Revised Risk-Informed Inservice Inspection Evaluation Procedure, EPRI, Palo Alto, CA: 1999. TR-112657, Rev B-A.
3. NEI-04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems," dated April 2004.

Table 1
Inspection Location Selection Comparison Between ASME Section XI Code And EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RPV	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-F	2	0		0	
						B-J	2	0		0	
RPV	6	Low	Medium	None	Low	B-F	6	0		0	
						B-J	21	0		0	
RCR	2 (2)	High (High)	High	TT (IGSCC)[15]	Medium (Medium)	B-F	8	2[4]		2[4]	
RCR	2 (2)	High (High)	High	TT (IGSCC)	Medium (Medium)	B-J	69	18[4]		18[4]	
RCR	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	2	0	2[5]	0	2[5]
						B-J	32	4[6]		4[6]	
RCR	5	Medium	Medium	TASCS	Medium	B-J	5	1		1	
RCR	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	26	0		0	
RCR	6	Low	Medium	None	Low	B-J	43	0		0	
RCR	7	Low	Low	None	Low	B-J	4	0		0	
RWCU	4(2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-J	1	1[7]		1[7]	
RWCU	6(5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-F	1	0		0	
						B-J	22	0		0	
RWCU	6	Low	Medium	None	Low	B-F	1	0		0	
						B-J	27	0		0	
RWCU	7	Low	Low	None	Low	B-J	2	0		0	
RCIC	6	Low	Medium	None	Low	B-J	22	0		0	
						C-F-2	7	0		0	
RCIC	6[16]	Medium[16]	Medium	None	Low	B-J	5	0		0	
RCIC	7	Low	Low	None	Low	C-F-2	7	0		0	
RHR	2(2)	High (High)	High	TT (IGSCC)	Medium (Medium)	B-F	2	1[8]		1[8]	
						B-J	2	0		0	
RHR	2	High	High	TT	Medium	B-J	8	2		2	

Table 1
Inspection Location Selection Comparison Between ASME Section XI Code And EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
RHR	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	1	1[9]		1[9]	
						B-J	1	0		0	
RHR	4	Medium	High	None	Low	B-J	7	1		1	
RHR	6 (5)	Low (Medium)	Medium	None (IGSCC)	Low (Medium)	B-J	1	0		0	
RHR	6	Low	Medium	None	Low	B-J	31	0		0	
						C-F-2	433	0		0	
CS	2(2)	High (High)	High	(IGSCC)[15]	Low[15](Medium)	B-F	2	1[10]		1[10]	
CS	4 (2)	Medium (High)	High	None (IGSCC)	Low (Medium)	B-F	4	0	4[11]	0	4[11]
						B-J	2	1[12]	1[13]	1[12]	1[13]
CS	4	Medium	High	None	Low	B-J	16	2		2	
CS	6	Low	Medium	None	Low	B-J	22	0		0	
						C-F-2	136	0		0	
HPCI	4	Medium	High	None	Low	B-J	3	3		3	
						C-F-2	49	3		3	
HPCI	6	Low	Medium	None	Low	B-J	7	0		0	
						C-F-2	91	0		0	
HPCI	6[16]	Medium[16]	Medium	None	Low	B-J	9	0		0	
HPCI	7	Low	Low	None	Low	C-F-2	12	0		0	
MS	4	Medium	High	None	Low	B-J	60	6		6	
MS	6(3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	7	0		0	
MS	6	Low	Medium	None	Low	B-J	38	0		0	
						C-F-2	147[17]	0		0	
FW	2(1)	High (High)	High	TASCS, TT (FAC)	Medium (High)	B-J	8	2		2	
FW	2(1)	High (High)	High	TASCS, CC (FAC)	Medium (High)	B-J	8	2	3[14]	2	3[14]
FW	2(1)	High (High)	High	TASCS (FAC)	Medium (High)	B-J	3	1		1	
FW	4(1)	Medium (High)	High	None (FAC)	Low (High)	B-J	49	5		5	
FW	5(3)	Medium (High)	Medium	TASCS (FAC)	Medium (High)	B-J	4	1		1	
FW	6(3)	Low (High)	Medium	None (FAC)	Low (High)	B-J	5	0		0	

Table 1
Inspection Location Selection Comparison Between ASME Section XI Code And EPRI TR-112657 by Risk Category

System[1]	Risk		Consequence Rank	Failure Potential		Code Category	Weld Count	First Approved Interval		New Interval	
	Category	Rank		DMs	Rank			RI-ISI	Other[2]	RI-ISI	Other[2]
CRD	4	Medium	High	None	Low	B-J	2	1		1	
CRD	6	Low	Medium	None	Low	B-F	2	0		0	
						B-J	31	0		0	
						C-F-2	27	2		0	
SLC	4	Medium	High	None	Low	B-J	6	1		1	
SLC	6	Low	Medium	None	Low	B-F	1	0		0	
						B-J	26	0		0	

Table Notes:

1) Systems are described in Table 2.

2) The column labeled "Other" is used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. DAEC added ten welds as examination selections to bring the overall percentage of Class 1 selections to 10%.

3) Not Used.

4) These twenty welds were selected for examination by both the IGSCC Program and the RI-ISI Program. Thermal Transients were identified along with IGSCC, as a potential damage mechanism for these welds. In order to be credited toward both the IGSCC Program and the RI-ISI Program the IGSCC examinations will include the requirements identified in EPRI TR-112657 for thermal transient examinations.

5) These two welds were selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

6) These four welds were selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

7) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs..

8) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Thermal transients were identified along with IGSCC as a potential damage mechanism for this weld. In order to be credited toward both the IGSCC Program and the RI-ISI Program, the IGSCC examination will include the requirements identified in EPRI TR-112657 for thermal transient examinations.

9) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism for this weld, the IGSCC examination will be credited to both programs.

10) This one weld was selected for examination by both the IGSCC Program and by the RI-ISI Program. For this weld, IGSCC was identified as the potential damage mechanism.

11) These four welds were selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. Since IGSCC was the only potential damage mechanism identified for these welds, the IGSCC examinations will be credited toward both programs.

12) This one weld was selected for examination by both the IGSCC Program and the RI-ISI Program. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examionation will be credited toward both programs.

13) This one weld was selected for examination by the IGSCC Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld sections to 10%. Since IGSCC was the only potential damage mechanism identified for this weld, the IGSCC examination will be credited toward both programs.

14) These three welds were selected for examination by the NUREG-0619 Program and by the RI-ISI Program to bring the overall percentage of Class 1 weld selections to 10%. For these welds, TASCS and crevice corrosion were identified as potential damage mechanisms. Although, the NUREG-0619 examinations are included in the RI-ISI Program, they are not credited as risk-informed examinations in the

risk impact analysis. As such, the NUREG-0619 examinations by themselves could be credited toward both programs. However, to ensure that all potential damage mechanisms are investigated, DAEC has elected to supplement the NUREG-0619 examinations for these three welds with the requirements identified in EPRI TR-112657 for TASCS and crevice corrosion examinations.

15) Recirculation riser safe-end and Core Spray injection safe-end welds are not considered to be subject to crevice corrosion degradation per the "Enhanced Crevice Corrosion Criteria in RI-ISI Evaluations," EPRI Technical Update 1011945, November 2005. The failure potential ranking for Core Spray was moved from medium to low because of the elimination of the degradation mechanism (crevice corrosion).

16) The risk rank was increased from low to medium as an effect from the updated PSA with higher probabilities of failure for human performance events. The results are not significant enough to change exam selections.

17) One new weld in Main Steam due to modifications of Main Steam Reheat System adding one weld.

Table 2
System Selection and Segment /Element Definition

System Description	Number of Segments	Number of Elements
RPV-Reactor Pressure Vessel	11	31
RCR-Reactor Coolant Recirculation	56	189
RWCU-Reactor Water Clean-Up	14	54
RCIC-Reactor Core Isolation Cooling	7	41
RHR-Residual Heat Removal	53	486
CS-Core Spray	29	182
HPCI – High Pressure Coolant Injection	21	171
MS- Main Steam	48	252
FW-Feedwater	20	77
CRD-Control Rod Drive	8	62
SLC – Standby Liquid Control	6	33
Totals	273	1578

6) Relief Request NDE-R006

COMPONENT IDENTIFICATION

Code Classes:	1, 2, and 3
Examination Categories:	B-G-1, B-G-2, B-L-2, B-M-2, B-N-1, B-N-2, B-P, C-B, C-H, D-A, D-B, F-A
Item Numbers:	B6.10, B6.50, B6.190, B6.200, B7.10, B7.50, B7.60, B7.70, B7.80, B12.20, B12.50, B13.10, B13.20, B13.30, B13.40, B15.10, C7.10, D1.20, D2.10, F1.10, F1.20, F1.30, F1.40
Component Numbers:	Various

CODE REQUIREMENT

ASME Code Section XI, 2001 Edition through 2003 Addenda IWA-2210 through IWA-2213 and Table IWA-2210-1

IWA-2210, "VISUAL EXAMINATIONS," requires:
Visual examinations shall be conducted in accordance with Section V, Article 9, Table IWA-2210-1, and the following.

(a) A written procedure and report of examination results is required.
(b) For procedure demonstration, a test chart containing text with some lower case characters without an ascender or descender (e.g., a, c, e, o) meeting Table IWA-2210-1 is required. Measurements of the test chart shall be made once before initial use with an optical comparator (10X or greater) or other suitable instrument to verify that the height of a representative lower case character without an ascender or descender, for the selected type size, meets the requirements of Table IWA-2210-1.
(c) Remote examination may be substituted for direct examination. The remote examination procedure shall be demonstrated to resolve the selected test chart characters.
(d) Alternatives to the direct visual examination distance requirements of Section V may be used as specified in Table IWA-2210-1.
(e) It is not necessary to measure illumination levels on each examination surface when the same portable light source or similar installed lighting equipment is demonstrated to provide the illumination specified in Table IWA-2210-1 at the maximum examination distance.
(f) The adequacy of the illumination levels from battery powered portable lights shall be checked before and after each examination or series of examinations, not to exceed 4 hours between checks. In lieu of using a light meter, these checks may be made by verifying that the illumination is adequate (i.e., no discernable degradation in the visual examination resolution of the procedure demonstration test chart characters).

IWA-2211, "VT-1 Examination," requires:
VT-1 examinations are conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.

IWA-2212, "VT-2 Examination," requires:
(a) VT-2 examinations are conducted to detect evidence of leakage from pressure retaining components, with or without leakage collection systems, as required during the conduct of system pressure test.

(b) VT-2 examinations shall be conducted in accordance with IWA-5000. For direct examination, the Table IWA-2210-1 maximum examination distance shall apply to the distance from the eye to the surfaces being examined.

IWA-2213, "VT-3 Examination," requires:
VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearances, settings, and physical displacements; and to detect discontinuities and imperfections, such as loss of integrity at bolted or welded connections, loose or missing parts, debris, corrosion, wear, or erosion. VT-3 includes examinations for conditions that could affect operability or functional adequacy of snubbers and constant load and spring-type supports.

Table IWA-2210-1:

Visual Examination	Minimum Illumination,[1] *fc*	Maximum Direct Examination Distance, *ft* (mm)	Maximum Procedure Demonstration Lower Case Character Height, in. (mm)
VT-1	50	2 (609.6)	0.044 (1.1)
VT-2	15	6 (1829)	0.158 (4)
VT-3	50	4 (1219)	0.105 (2.7)

NOTE:
(1) Resolution of the specified characters can be used in lieu of illumination measurement to verify illumination adequacy.

REASON FOR REQUEST

Pursuant to 10 CFR 50.55a(a)(3)(ii), the Duane Arnold Energy Center requests authorization to use ASME Code Case N-686, "Alternate Requirements for Visual Examinations, VT-1, VT-2, and VT-3, Section XI, Division 1," approved by ASME on February 14, 2003, in lieu of the requirements of ASME Code Section XI, IWA-2210 through IWA-2213 and Table IWA-2210-1, when performing VT-1, VT-2, and VT-3 visual examinations.

In order to meet the distance requirements and to gain access to all areas to complete VT-2 and VT-3 visual examinations in accordance with IWA-2210 through IWA-2213 and Table IWA-2210-1, remote visual equipment would have to be used or scaffolding would have to be erected and removed for some locations. This effort would cause additional radiation exposure. This requirement will cause a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

BASIS FOR RELIEF AND ALTERNATIVE EXAMINATIONS

As the proposed alternative, Duane Arnold Energy Center (DAEC) will use the provisions in Code Case N-686, without exception, in lieu of IWA-2210 through IWA-2213 and Table IWA-2210-1 when performing VT-1, VT-2, and VT-3 visual examinations. Specifically, Code Case N-686 states that VT-2 examination shall be conducted in accordance with IWA-5000, and that for VT-3 examination, there are no direct visual examination distance requirements, provided the examiner can resolve the characters in accordance with Table 1 (shown below). The only difference in the VT-1 examination is that the metric system for distance has been rounded off (slightly different numbers) in Code Case N-686.

Code Case N-686, Table 1:

Visual Examination	Minimum Illumination, *fc* [Note (1)]	Maximum Direct Examination Distance, *ft* (mm)	Maximum Height, in. (mm) for Procedure Demonstration Characters [Note (2)]
VT-1	50	2 (600)	0.044 (1.0)
VT-3	50	N/A	0.105 (3.0)

NOTES:
(1) Resolution of the specified characters can be used in lieu of illumination measurement to verify illumination adequacy.
(2) For procedure demonstration, a test chart or card containing text with some lower case characters, without an ascender or descender (e.g., a, c, e, o), that meet the specified height requirements is required. Measurement of the test chart or card shall be made once before its initial use with an optical comparator (10X or greater) or other suitable instrument to verify that the height of the lower case characters without an ascender or descender meets the specified requirements.

DAEC will perform VT-2 and VT-3 examinations without direct visual examination distance requirements in accordance with Code Case N-686.

The basis for use is as follows:

The different visual examination techniques have evolved over the years from a single technique (VT-1) to the separate techniques of VT-1, VT-2, and VT-3 with examination requirements commensurate with their application.

ASME Section XI, 1974 Edition, Summer 1975 Addenda, contained only one visual examination:

IWA-2210, "VISUAL EXAMINATION"
(a) A visual examination is employed to provide a report of the general condition of the part, component, or surface to be examined, including such conditions as scratches, wear, cracks, corrosion, or erosion on the surfaces; misalignment or movement of the part or component; or evidence of leaking.
(b) Visual examination shall be conducted in accordance with Article 9 of Section V, except that lighting shall be sufficient to resolve the 1/32-in. line.

The requirements of the corresponding edition of Section V, Article 9, "Visual Examination," are summarily stated as: Direct visual examination may usually be made when access is sufficient to place the eye within 24 in. of the surface to be examined and at an angle not less than 30 deg. to the surface to

be examined. Mirrors may be used to improve the angle of vision.... Remote visual examination may use visual aids.... Such systems shall have a resolution capability at least equivalent to that obtainable by direct visual observation.

This one visual examination contained requirements for physical damage (e.g., scratches, wear, cracks, corrosion, erosion), physical displacement (e.g., misalignment, movement), and evidence of leaking and applied it to all visual examinations required by Section XI, including pressure retaining welds, pressure retaining bolting, vessel cladding, vessel interior, component supports, and leakage tests.

In the 1970s the visual examinations were split into multiple examinations. For example, VT-1 for physical damage, with defined prerequisites; VT-2 for pressure boundary leakage, with fewer defined prerequisites; VT-3 for physical displacement, also with fewer defined prerequisites; and VT-4 for functional adequacy. The reason the visual examinations were separated into multiple methods with appropriate requirements was to apply a level of visual examination commensurate with the application.

The visual VT-2 examination performed during the Class 1 system leakage test is typically performed after a refueling outage when the unit is at reactor pressure and temperature. Table IWA-2210-1 requires the examiner to be within six feet of the surfaces being examined or use remote examination equipment that provides demonstrated equivalent resolution. For an examiner to be within six feet of the surfaces being examined would require the erection of scaffolding to perform a system pressure test because the piping runs for certain systems may be 20 to 30 feet above the floor. The plant personnel erecting and taking down the scaffolding or the additional plant personnel required to perform remote examinations (for example, personnel to install or hold a light source if the examiner used binoculars) would receive unnecessary radiation exposure. However, ASME Code Case N-686 allows the examiner to conduct VT-2 examinations to detect evidence of leakage from pressure retaining components without a distance limitation and prescribes examinations in accordance with IWA-5000. Paragraph IWA-5241, "Insulated and Noninsulated Components," allows the examiner to perform examinations for leakage "... *by examining the accessible external exposed surfaces of pressure retaining components... For components whose external surfaces are inaccessible for direct VT-2 visual examination, only the examination of the surrounding area (including floor areas or equipment surfaces located underneath the components) for evidence of leakage shall be required.*"

Table IWA-2210-1 also requires a minimum illumination level of 15 footcandles for a VT-2 examination. In order to meet this illumination level, temporary light may have to be provided which, again, involves more plant personnel and causes additional radiation exposure. Experience has shown, however, that there are other effective techniques and tools for locating leakage. For example, when water is illuminated with a flashlight it has a "mirror effect" or shiny reflective area, allowing leaks to be located from distances greater than six feet. Therefore, a VT-2 examination using a flashlight provides a level of quality equivalent to performing the examination with general illumination of 15 footcandles.

A VT-3 examination is conducted to determine the general mechanical and structural condition of a component or a component support. Table IWA-2210-1 requires the examiner to be within four feet of the surfaces being examined or use remote examination equipment that provides demonstrated equivalent resolution. Again, the piping runs for certain systems may be 20 to 30 feet above the floor. This would require the erection of scaffolding to perform a visual examination of a component support. In addition, as discussed above, the use of remote examination equipment involves more plant personnel.

The industry has over thirty years of experience performing visual examinations to the less prescriptive requirements for proximity and illumination, and examiners are fully qualified in accordance with IWA-2300, "Qualifications of Nondestructive Examination Personnel." Experience, training, and qualifications of visual examiners provide reasonable assurance that they will apply the appropriate illumination and distance requirements required to perform quality examinations.

The specific requirements of IWA-2210 through IWA-2213 and Table IWA-2210-1 will cause a hardship or unusual difficulty without a compensating increase in the level of quality and safety due to ALARA considerations. Thirty years of industry experience performing system pressure tests demonstrates that an equivalent level of quality and safety can be achieved by performing VT-2 examinations at distances well in excess of six feet and VT-3 examinations at distances well in excess of four feet. These time-proven methods for conducting visual examinations will continue to provide reasonable assurance of structural integrity while preventing plant personnel from receiving excessive radiation exposure.

The 1989 Edition of ASME Section XI, which was the applicable ASME Code for the DAEC third ten-year interval, did not specify distance and illumination requirements for VT examinations; however, per an Erratum, VT examinations now include distance and illumination requirements. ASME Code Case N-686 was prematurely incorporated into ASME Section XI 2001 Edition, 2003 Addenda (Sections IWA-2210 through 2213, including Table IWA 2211-1). An Erratum was issued in December 2003 which restored it back to the 2002 Addenda version, which specifies distance and illumination requirements. Subsequently, Code Case N-686 was incorporated into the 2004 Edition 2005 Addenda of ASME Section XI. However, the applicable code edition and addenda for DAEC is ASME Code Section XI, 2001 Edition, 2003 Addenda.

Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), DAEC requests authorization to use ASME Code Case N-686 in lieu of ASME Code IWA-2210 through IWA-2213 and Table IWA-2210-1 requirements.

DURATION OF PROPOSED ALTERNATIVE

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for DAEC.

PRECEDENT

None

7) Relief Request NDE-R007

COMPONENT IDENTIFICATION

Code Class:	3
Examination Category:	D-B
Item Number:	D2.10
Component Numbers:	Various

CODE REQUIREMENT

ASME Section XI, 2001 Edition through the 2003 Addenda, IWA-5244 (b) states that for buried components where a VT-2 visual examination cannot be performed, the examination requirement is satisfied by the following:

1) The system pressure test for buried components that are isolable by means of valves shall consist of a test that determines the rate of pressure loss. Alternatively, the test may determine the change in flow between the ends of the buried components. The acceptable rate of pressure loss or flow shall be established by the Owner.
2) The system pressure test for nonisolable buried components shall consist of a test to confirm that flow during operation is not impaired.

Reason for Request

IWA-5244(b)(l) requires either a pressure loss test or a test that determines the change in flow between the ends of the buried components for isolable sections of buried piping. The acceptable rate of pressure loss or flow shall be established by the Owner. Sections of River Water Supply, Emergency Service Water (ESW), and Residual Heat Removal Service Water (RHRSW) System buried piping were not designed with consideration for isolation valves adequate for performing a pressure loss type test or do not contain instrumentation adequate for measuring changes in flow between the ends of the buried piping.

The River Water Supply System contains large diameter buried piping (24 inch diameter) that runs from the River Intake Structure to the Pump House and is greater than 1500 feet in length. The ESW System and the RHRSW System contain large diameter buried piping (16 inch diameter for RHRSW and 8 inch and 6 inch diameter for ESW) that runs from the Pump House to the Turbine Building and is greater than 500 feet in length. The subject piping design for these systems did not provide for isolation valves that are capable of supporting a pressure loss type test considering the volume of the piping and the available capacity of test pumps. The system isolation valves were only intended to provide isolation for maintenance activities with only static system pressure.

River Water Supply and ESW were designed with a single flow element per train located in the Pump House. ESW has some additional flow instrumentation on some downstream components, but not for every branch on a train. RHRSW was designed with a single flow element per train located in the Reactor Building before the Residual Heat Removal System Heat Exchanger. Therefore, the installed instrumentation is inadequate for measuring the flow difference at each end of the buried piping. The use of ultrasonic flow instrumentation was considered, but the piping configurations do not provide for the straight runs of piping required for accurate flow measurement.

Both the River Water Supply and RHRSW systems include four pumps each with two pumps designated to each of two independent trains. The River Water Supply pumps and RHRSW pumps have installed excess capacity. Therefore, each of the independent trains of both the River Water and RHRSW systems can accommodate a leak and still satisfy the accident analysis requirements. ESW has one pump per train. The ESW system supplies various plant heat exchangers, which have flow margin due to heat transfer requirements.

PROPOSED ALTERNATIVE AND BASES FOR USE

IWA-5244(b)(l) requires the Owner to establish the acceptance criteria for the buried piping test. Since there is no industry guidance for acceptance criteria, DAEC considered that the allowable ASME OM Code 2001 Edition subsection ISTA instrument accuracy requirements for pump Inservice Testing should be adequate. The Subsection ISTA requires flow instruments with a calibration accuracy of ±2%. Each of the River Water Supply, ESW and RHRSW pumps are tested in accordance with the DAEC IST Program on a quarterly frequency. Each pump test requires approximately thirty-minutes to perform. Previously a leak was discovered in non-safety related buried piping. The leak was the size of a dime, which demonstrates that small indications are readily identified by visual observation of the surrounding ground surface area on operating systems.

At least one River Water pump is required to be in operation at all times during normal plant power operation. At least one RHRSW pump and ESW pump are required to be in operation for extended periods of time at the beginning and end of each refueling outage. Therefore, both systems are inservice for extended periods of time and leaks like those discussed above would be readily identified by plant personnel performing routine inspections during rounds.

DAEC proposes to perform visual examination of the ground surface area immediately above each buried section of River Water Supply, ESW and RHRSW on a refuel cycle bases in lieu of performing the test required by IWA-5244(b)(l). The visual examinations will be performed only after the subject piping has been in operation at nominal operating conditions for a minimum of 24-hours. The ASME Section XI code only requires a pressure test once each period (Every 3 to 4 years).

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested on the basis that the proposed alternative would provide an acceptable level of quality and safety

DURATION OF PROPOSED ALTERNATIVE

The proposed alternative will be used for the entire fourth ten-year interval of the Inservice Inspection Program for DAEC.

Precedents

None

8) Relief Request NDE-R008

COMPONENT IDENTIFICATION

Code Class:	1
Examination Categories:	B-A
Item Number:	B1.30, B1.40
Component Number:	Reactor Pressure Vessel (RPV) shell-to-flange weld and head-to-flange weld

CODE REQUIREMENT

The Applicable Code Edition for requested relief is Section XI, 2001 Edition through the 2003 Addenda.

ASME Section XI, Appendix I, I-2110(b) currently requires that ultrasonic (UT) examination, which includes personnel qualification, procedures, scanning and examination requirements of the subject welds be conducted in accordance with Article 4 of Section V for the Reactor Pressure Vessel (RPV) shell-to-flange weld and head-to-flange weld.

Fourth interval examinations will be performed per the requirements of ASME Section XI, 2001 Edition through the 2003 Addenda, as amended by 10 CFR 50.55a. Per 10 CFR 50.55a(b)(2)(xxiv), the use of Appendix VIII and supplements to Appendix VIII of Section XI of the 2002 Addenda through the 2003 Addenda is prohibited. Therefore, for Appendix VIII and supplements to Appendix VIII the 2001 Edition of Section XI (no addenda) will be used.

REASON FOR REQUEST

10 CFR 50.55a required that ASME Section XI, Appendix VIII, Supplement 4, "Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel," and Supplement 6, "Qualification Requirements For Reactor Vessel Welds Other Than Clad/Base Metal Interface," be implemented for most of the RPV welds by November 22, 2000. However, the RPV shell-to-flange weld and head-to-flange weld examinations were not included in this requirement. For these welds, ASME Section XI, Appendix I, I-2110(b) currently requires that ultrasonic (UT) examination which includes personnel qualification, procedures, scanning and examination requirements of the subject welds be conducted in accordance with Article 4 of Section V.

PROPOSED ALTERNATIVE AND BASES FOR USE

The use of this alternative will allow the use of Performance Demonstration Initiative (PDI) qualified procedures to perform the examination of these welds in lieu of Article 4 of Section V requirements. During the upcoming Fourth Interval examinations, DAEC proposes to perform examinations using, personnel qualification, procedures, scanning, and equipment that are demonstrated and qualified in accordance with ASME Section XI, 2001 Edition (no addenda), Appendix VIII, Supplements 4 and 6 as amended by 10 CFR 50.55a for the RPV shell-to-flange weld and RPV head-to-flange weld. The examination will be performed manually or automated, as qualified in accordance with ASME Section XI, 2001 Edition (no addenda), Appendix VIII, Supplements 4 and 6 as amended by 10 CFR 50.55a and

the PDI demonstration process. Since the examination is performed from a single side due to the weld configuration, all procedures, personnel and equipment will be qualified for single side access for scanning of both welds.

Appendix VIII requirements were developed to ensure the effectiveness of UT examinations within the nuclear industry by means of a rigorous, item-specific performance demonstration. The performance demonstration (through PDI) was conducted on RPV mockups containing flaws of various size and allocations. The demonstration established the capability of equipment, procedures, and personnel to find flaws that could be detrimental to the integrity of the RPV. The performance demonstration showed that for the detection of flaws in RPV welds, the UT techniques were equal to or surpassed the requirements of Section V, Article 4 of the ASME Code. Additionally, the PDI qualified sizing techniques are considered to be more accurate than the techniques used in Article 4 of Section V.

Although Appendix VIII is not required for the RPV shell-to-flange weld and RPV head-to-flange weld, the use of Appendix VIII Supplement 4 and 6 criteria for detection and sizing of flaws in these welds will be equal to or will exceed the requirements established by Article 4 of Section V. Therefore, the use of this proposed alternative will continue to provide an acceptable level of quality and safety, and approval is requested pursuant to 10 CFR 50.55a(a)(3)(i).

DURATION OF PROPOSED ALTERNATIVE

The proposed alternative is applicable for the 4th Inservice Inspection Interval.

Precedents

The NRC approved Edwin Hatch Nuclear Plant Unit Nos 1 and 2 Relief Request ISI-ALT-1 on January 3, 2006, to allow the use of ASME Section XI Appendix VIII, Suplements 4 and 6, qualified procedures and personnel for examination of the RPV shell-to-flange and the RPV head-to-flange welds techniques for the fourth 10-year Inservice Inspection Interval. (TAC NO. MC6528 and MC6529).

1.0 COMPONENT EXAMINATION SUMMARY LISTING

All components and component supports potentially subject to inservice NDE examination under the 2001 Edition thru 2003 Addenda of Section XI are contained in Table I.

The table identifies the number of components and component supports selected for examination during the fourth inspection interval and provides a schedule by period, for the applicable required examination to be performed.

Table I is broken into three sections for Code Class and sorted by, Code Category/Item No., Examination Description, System Identification, required NDE examination method, Scheduled Period, and Comments. DAEC will maintain, on site, a controlled comprehensive ISI examination plan and schedule and will be made available for review.

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled			Comments
					1	2	3	
B-A	B1.11	Circumferential Shell Welds	Reactor Vessel	Volumetric			0	NDE-R003
	B1.12	Longitudinal Shell Welds	Reactor Vessel	Volumetric	2		6	
	B1.21	Circumferential Head Welds	Reactor Vessel	Volumetric			2	
	B1.22	Meridional Head Welds	Reactor Vessel	Volumetric			15	
	B1.30	Shell-to-Flange Weld	Reactor Vessel	Volumetric	1/2		1/2	NDE-R008
	B1.40	Head-to-Flange Weld	Reactor Vessel	Volumetric & Surface			1	NDE-R008 (Partial Exams during 1st & 2nd Periods)
B-D	B3.90	Nozzle-to-Vessel Welds in Reactor Vessel	Reactor Vessel	Volumetric	16	8	9	(1) Exempt by IWB 1220(c)
	B3.100	Nozzle Inside Radius Section in Reactor Vessel	Reactor Vessel	Volumetric	16	8	9	(1) Exempt by IWB 1220(c)
B-F	B5.10	Dissimilar Metal Nozzle-to-safe End Butt Welds NPS 4 or Larger	Various Class 1	Volumetric & Surface	0	0	0	NDE-R005
	B5.20	Dissimilar Metal Nozzle-to-safe End Butt Welds NPS 4 or Smaller	Various Class 1	Surface	0	0	0	NDE-R005
	B5.30	Reactor Vessel Nozzle-to-Safe End Socket Welds	Various Class 1	Surface	0	0	0	NDE-R005

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
B-G-1	B6.10	Reactor Vessel Closure Head Nuts	Reactor Vessel	Visual, VT-1			60	NDE-R006
	B6.20	Reactor Vessel Closure Studs, in Place	Reactor Vessel	Volumetric			60	
	B6.40	Threads in Reactor Vessel Flange	Reactor Vessel	Volumetric			60	
	B6.50	Reactor Vessel Closure Washers, Bushings	Reactor Vessel	Visual, VT-1			60	NDE-R006
	B6.180	Bolts & Studs in Pumps	All Class 1	Volumetric				Inspected only when disassembled
	B6.190	Flange Surface, When Connection Disassembled, in Pumps	All Class 1	Visual, VT-1				Inspected only when disassembled NDE-R006
	B6.200	Nuts, Bushings, & Washers in Pumps	All Class 1	Visual, VT-1				Inspected only when disassembled NDE-R006
B-G-2	B7.10	Bolts, Studs, & Nuts in Reactor Vessel	Various Class 1	Visual, VT-1	1	1	1	NDE-R006
	B7.50	Bolts, Studs, & Nuts in Piping	Various Class 1	Visual, VT-1				Inspected only when disassembled NDE-R006
B-G-2	B7.60	Bolts, Studs, & Nuts in Pumps	Various Class 1	Visual, VT-1				Inspected only when disassembled NDE-R006
	B7.70	Bolts, Studs, & Nuts in Valves	Various Class 1	Visual, VT-1		1		Inspected only when disassembled NDE-R006

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
10 CFR 50.55a Requirement	B7.80	Bolts, Studs, & Nuts in CRD Housings	Reactor Vessel	Visual, VT-1	13			Inspected only when disassembled and bolting reused NDE-R006
B-J	B9.11	Circumferential Welds in Piping NPS 4 or Larger	Various Class 1	Volumetric & Surface	0	0	0	NDE-R005
	B9.21	Circumferential Welds in Piping Less Than NPS 4	Various Class 1	Surface	0	0	0	NDE-R005
	B9.31	Branch Pipe Connection Welds NPS 4 or Larger	Various Class 1	Volumetric & Surface	0	0	0	NDE-R005
	B9.32	Branch Pipe Connection Welds Less Than NPS 4	Various Class 1	Surface	0	0	0	NDE-R005
	B9.40	Socket Welds	Various Class 1	Surface	0	0	0	NDE-R005
B-K	B10.10	Integrally Welded Attachments to Reactor Vesse	Various Class 1	Surface			1	NDE-R004
	B10.20	Integrally Welded Attachments to Piping	Various Class 1	Surface	2	1	1	
	B10.30	Integrally Welded Attachments to Pumps	Various Class 1	Surface			1	
B-L-2	B12.20	Pump Casings	Various Class 1	Visual, VT-3				2 pumps Inspected only when disassembled NDE-R006
B-M-2	B12.50	Valve Bodies, Exceeding NPS 4	Various Class 1	Visual, VT-3				Selected valves Inspected only when disassembled NDE-R006

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
B-N-1	B13.10	Vessel Interior	Various Class 1	Visual, VT-3	11	11	11	Accessible 11 components – period NDE-R006
B-N-2	B13.20	Interior Attachments within Beltline Region in Reactor Vessel	Reactor Vessel	Visual, VT-1			11	NDE-R006
	B13.30	Interior Attachments beyond Beltline Region in Reactor Vessel	Reactor Vessel	Visual, VT-3			23	Some attachments are normally not accessible (bottom head area) NDE-R006
	B13.40	Core Support Structure in Reactor Vessel	Reactor Vessel	Visual, VT-3				NDE-R006 (Accessible Surfaces)
B-O	B14.10	Welds in CRD Housing, Peripheral CRDs	Reactor Vessel	Volumetric or Surface			3	
B-P	B15.10	System Leakage Test	Reactor Vessel	Visual, VT-2	2	1	1	NDE-R006
C-A	C1.10	Circumferential Shell Welds	RHR	Volumetric	1		1	
	C1.20	Circumferential Head Welds	RHR	Volumetric		1		
C-B	C2.21	Nozzle-to-Shell (or Head) Weld without Reinforcing Plates in Vessels > 1/2" Nominal Thickness	RHR	Volumetric & Surface	--	1	1	
	C2.22	Nozzle Inner Radius		Volumetric	--	1	1	
C-C	C3.10	Integrally Welded Attachments to Pressure Vessels	Various Class 2	Surface			1	NDE-R004

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
	C3.20	Integrally Welded Attachments to Piping	Various Class 2	Surface	2	2	3	
C-F-2	C5.51	Circumferential Welds in Carbon or Low Alloy Steel Piping > or = 3/8" Nominal Wall Thickness For Piping > NPS 4	Various Class 2	Volumetric & Surface	0	0	0	NDE-R005
	C5.81	Circumferential Welds in Carbon or Low Alloy Steel Branch Connections Piping For Piping > NPS 4 (Reference Table IWC-2500-1, Note 1).	Various Class 2	Surface	0	0	0	NDE-R005
C-H	C7.10	System Leakage Test of Pressure Retaining Components	Various Class 2	Visual, VT-2	7	7	7	NDE-R006
D-A	D1.20	Integral Attachments - Piping	Various Class 3	Visual, VT-1	2	2	3	NDE-R006
D-B	D2.10	System Leakage Test	Various Class 3	Visual, VT-2	3	3	3	NDE-R006 NDE-R007
F-A	F1.10	Class I Component Supports	Various Class 1	Visual, VT-3	14	13	16	NDE-R006
F-A	F1.20	Class II Component Supports	Various Class 2	Visual, VT-3	14	16	22	NDE-R006
	F1.30	Class III Component Supports	Various Class 3	Visual, VT 3	6	7	12	NDE-R006
	F1.40	Supports Other Than Piping Supports (Class 1, 2, and 3)		Visual, VT-3	9	10	9	NDE-R006

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
R-A*		* Risk-Informed weld selection for Categories B-F, B-J, and C-F-2.	Various Class 1 & Class 2	Volumetric	18	31	21	NDE-R005
	R1.10	No Degradation Mode						
	R1.11	Thermal Fatigue						
	R1.14	Corrosion Cracking						
	R1.16	Intergranular Stress Corrosion Cracking						
	R1.18	Flow-Acclerated Corrosion						