August 31, 2011

NG-11-0320
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

Reference: Letter, Gary Van Middlesworth (FPL Energy Duane Arnold, LLC) to Document Control Desk (USNRC), Fourth 10-Year Inservice Inspection Plan, dated June 30, 2006, NG-06-0439 (ML061870230)

The Duane Arnold Energy Center (DAEC) began its fourth inservice inspection (ISI) interval on November 1, 2006. Per the referenced letter, this interval was to end on February 21, 2014 to coincide with the expiration of the DAEC operating license. However, on December 16, 2010, the operating license for the DAEC was renewed and its expiration date was extended to February 21, 2034. Therefore, NextEra Energy Duane Arnold, LLC (hereafter NextEra Energy Duane Arnold) herewith submits the DAEC ISI plan for the current fourth full 10 year interval as extended to October 31, 2016.

The attached plan was submitted prior to the beginning of the interval and is being used for inspections during the fourth ten year interval. By extending the interval to end in 2016, some minor changes to relief requests were needed. The relief requests are contained in Section H of the Enclosure. DAEC is requesting approval of relief requests NDE-R003, NDE-R010, NDE-R011, and NDE-R014. The only change to these four relief requests is extending their duration to the end of the current fourth full 10 year interval now ending on October 31, 2016. Note that NDE-R014 was submitted to the NRC for approval on November 10, 2010 (ML103160155). NextEra Energy Duane Arnold requests approval of the interval extension for these relief requests by February 21, 2014. Additionally, DAEC also requests approval of relief request NDE-R015 (submitted to the NRC on December 4, 2010 (ML103400074).

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NRR

NextEra Energy Duane Arnold, LLC, 3277 DAEC Road, Palo, IA 52324

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

Peter Wells
Vice President, Duane Arnold Energy Center
NextEra Energy Duane Arnold, LLC

Enclosure: 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

TECHNICAL REVIEW

Reviewed by:	(See NAMS for review/approval information) ISI Program Owner	Date:	
Reviewed by:	(See NAMS for review/approval information) NDE Level III	Date:	
Reviewed by:	(See NAMS for review/approval information) Inspection and Material Supervisor	Date:	

PROCEDURE APPROVAL

Approved by Procedure Owner:	(See NAMS for review/approval information) Program Engineering Manager	Date:	
Reviewed by:	(See NAMS for review/approval information) Authorized Nuclear Inservice Inspector	Date:	

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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 interval started November 1, 1996 and ended October 31, 2006 . An extension of the License for 20 years was granted in December 2010, as such, the Fourth Ten Year interval was extended to a full 10 years.

(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, 2013 (48 Months)
Third Period: November 1, 2013 - October 31, 2016 (36 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 15, 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 I*	*RG 1.147 Rev. 15*	*None*
N-504-3	*Alternative Rules for Repair of Class 1, 2 and 3 Austenitic Stainless Steel Piping, Section XI Division I*	*RG 1.147 Rev. 15*	*The provisions of Section XI, Nonmandatory Appendix Q, "Weld Overlay Repair of Class 1, 2 and 3 Austenitic Stainless Steel Piping Weldments," must also be met.*
N-513-2	*Evaluation Criteria or Temporary. Acceptance of Flaws in Class 2 or 3 Piping Section XI Division I*	*RG 1.147 Rev. 15*	*None*
N-526	*Alternative Requirements for Successive Inspections of Class 1 and 2 Vessels, Section XI, Division I*	*RG 1.147 Rev. 15*	*None*
N-532-4	*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. 15*	*None*

N-552	*Alternative Methods – Qualification for Nozzle Inside Radius Section from the Outside Surface, Section XI, Division I*	*RG 1.147 Rev. 15*	*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-566-2	*Corrective Action for Leakage Identified at Bolted Connections Section XI, Division 1*	*RG 1.147 Rev 15*	*None*
N-586-1	*Alternative Additional Examination Requirements for Classes 1, 2, and 3 Piping, Components, and Supports, Section XI, Division 1.*	*RG 1.147 Rev 15*	*None*

N-597-2	*Requirements for Analytical Evaluation of Pipe Wall Thinning, Section XI, Division 1*	*RG 1.147 Rev. 15*	*1) Code Case must be supplemented by the provisions of EPRI Nuclear Safety Analysis Center Report 202L-R2, "Recommendation for an Effective Flow Accelerated Corrosion Program," (Ref 6), April 1999, 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" is to be applied as"shall" (i.e., a requirement)..* *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, "Rules for Construction of Nuclear Power Plant Components" of the ASME Code (Ref. 7) prior to the value of t_p reaching the allowable minimum wall thickness t_{min}, as specified in -3622.1(a)(1) of this Code Case. Alternatively, use of the Code Case is subject to NRC review and approval per 10CFR50.55(a)(3).* *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 per 10CFR50.55(a)(3).* *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 t_{min}.* *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 for BWR CRD Housing/Stub Tube Repairs", Section XI, Division 1."*	*RG 1.147 Rev. 15*	*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	*Ultrasonic Examination of Penetration Nozzles in Vessels, Examination Category B-D, Item No's. B3.10 and B3.90, Reactor Nozzle-To-Vessel Welds, Figs. IWB-2500-7(a), (b), and (c) Section XI, Division 1.*	*RG 1.147 Rev. 15*	*None*
N-624	*Successive Inspections, Section XI, Division I*	*RG 1.147 Rev. 15*	*None*
N-629	*Use of Fracture Toughness Test Data to Establish Reference Temperature for Pressure Retaining Materials, Section XI, Division I*	*RG 1.147 Rev. 15*	*None*
N-638-1	*Similar and Dissimilar Metal Welding using Ambient Temperature Machine GTAW Temper Bead Technique, Section XI, Division 1.*	*RG 1.147 Rev. 15*	*UT volumetric examinations shall be performed with personnel and procedures qualified for the repaired volume and qualified by demonstration using representative samples which contain construction type flaws. The acceptance criteria of NB-5330 in the 1998 Edition through 2000 Addenda of Section III (Ref. 7) apply to all flaws identified within the repaired volume.*

N-639	*Alternative Calibration Block Material, Section XI, Division 1*	*RG 1.147 Rev. 15*	*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-661	*Alternate Requirements for Wall Thickness Restoration of Classes 2 and 3 Carbon Steel Piping for Raw Water Service, Section XI, Division 1.*	*RG 1.147 Rev. 15*	*(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-685	*Lighting Requirements for Surface Examination, Section XI, Division 1.*	*RG 1.147 Rev. 15*	*None* *Ref. NG-09-0911 for lighting demonstration.*
N-686	*Alternate requirements for Visual Examinations, VT-1, VT-2, and VT-3 Section XI.*	*Relief Request NDE-R006*	*Shall use the 2001 Edition of ASME Section V for T-941 and T-990 (ref. CAP062676)*
N-695	*Qualification Requirements for Dissimilar Metal Piping Welds, Section XI, Division 1*	*RG 1.147 Rev. 15*	*None*
N-700	*Alternative Rules for Selection of Classes 1, 2, and 3 Vessel Welded Attachments for Examination, Section XI, Division 1.*	*RG 1.147 Rev. 15*	*None*

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 <u>function</u> 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 the operating license for DAEC was applied and the Final Safety Analysis Report (FSAR) for the facility was submitted 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 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) NRC IE Bulletin 80-13, Cracking in Core Spray Spargers. These examinations have been upgraded to implement the recommended inspections contained in BWRVIP-18.

(f) 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 Relief Request 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

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:

<u>Class 1</u>

B2.10, B2.11, B2.12, 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

<u>Class 2</u>

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

<u>Class 3</u>

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:

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

$$D_s = 2 \ D_w$$

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.

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

Ds = 2 x 1.12 = 2.24" 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.

(c) 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

(3) 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	Buried Piping Intake Structure to Pumphouse	3.1-39
N/A	Buried Piping Pumphouse to HPCI/RCIC Building	3.1-40
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 - 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 - 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 - 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 - 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 - 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 - 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	NDE-R013	
	B3.100	Nozzle Inside Radius Section in Reactor Vessel	34	Volumetric	NDE-R013	
B-F	B5.10	Reactor Vessel Nozzle-to-Safe End Butt Welds NPS 4 or Larger	N/A[(1)]	Volumetric & Surface	NDE-R005	
	B5.20	Reactor Vessel Nozzle-to-Safe End Butt Welds Less Than NPS 4	N/A[(1)]	Surface	NDE-R005	
	B5.30	Reactor Vessel Nozzle-to-Safe End Socket Welds	N/A[(1)]	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	48 valve locations	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(1)	Volumetric & Surface	NDE-R005	
	B9.21	Circumferential Piping Welds	N/A(1)	Surface	NDE-R005	
	B9.31	Branch Pipe Connection Welds NPS 4 or Larger	N/A(1)	Volumetric & Surface	NDE-R005	
	B9.32	Branch Pipe Connection Welds Less Than NPS 4	N/A(1)	Surface	NDE-R005	
	B9.40	Socket Welds	N/A(1)	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		
	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	N/A(2)	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 locations	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	

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
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		
	C2.22	Nozzle Inner Radius	2 (one vessel)	Volumetric		
C-C	C3.10	Integrally Welded Attachments to Pressure Vessels	5 (one vessel)	Surface		
	C3.20	Integrally Welded Attachments to Piping	63	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(1)	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(1)	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	6	Visual, VT-2	NDE-R006 NDE-R007	

Examination Category	Item Number	Description	Total Number of Components	Exam Requirements	Relief Request	Technical Approach & Position
F-A	F1.10	Class I Component Supports	164	Visual, VT-3	NDE-R006	
	F1.20	Class II Component Supports	340	Visual, VT-3	NDE-R006	
	F1.30	Class III Component Supports	238	Visual, VT 3	NDE-R006	
	F1.40	Supports Other Than Piping Supports (Class 1, 2, and 3)	31	Visual, VT-3	NDE-R006	
R-A(1, 3)	R1.10	No Degradation Mode	1264	Volumetric	NDE-R005	
	R1.11	Thermal Fatigue	77	Volumetric	NDE-R005	
	R1.14	Corrosion Cracking	0	Volumetric	NDE-R005	
	R1.16	Intergranular Stress Corrosion Cracking	162	Volumetric	NDE-R005	
	R1.18	Flow-Accelerated Corrosion	61	Volumetric	NDE-R005	

Notes:

(1) Examination Categories B-F, B-J, and C-F-2 are evaluated as part of Risk Informed ISI based on EPRI methodology and included in Examination Category R-A.

(2) Examination in the spaces above and below the reactor core for loose or missing parts and debris.

(3) Risk Informed components may have multiple methods of degradation. Numbers reflect the primary degradation method identified for each component.

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.
TAP-I004	1	DAEC	Examination of Reactor Vessel Internals
TAP-I005	0	DAEC	Examination of Reactor Vessel Nozzle-to-Shell 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.

CONCLUSION

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.

(4) TECHNICAL APPROACH AND POSITION NUMBER: TAP-I004

COMPONENT IDENTIFICATION

Code Classes:	1
References:	Reactor Vessel Internals
Examination Categories:	B-N-1 and B-N-2
Item Numbers:	B13.10, B13.20, B13.30, and B13.40

EXAMINATION REQUIREMENTS

ASME Section XI (2001 Edition with the 2003 Addenda), Table IWB-2500-1, Examination Categories B-N-1 and B-N-2 requires examination of the reactor vessel internals.

Item Number B13.10 requires a visual examination (VT-3) of the accessible vessel interior spaces above and below the reactor core (B-N-1) each inspection period.

Item Number B13.20 requires a visual examination (VT-1) of the interior attachment welds within the beltline region (B-N-2) each inspection interval.

Item Number B13.30 requires a visual examination (VT-3) of the interior attachment welds beyond the beltline region (B-N-2) each inspection interval.

Item Number B13.40 requires a visual examination (VT-3) of the core support structure surfaces (B-N-2) each inspection interval.

POSITION

This Technical Approach and Position is to define the different reactor vessel interior components that will be examined under each of the above ASME Section XI requirements. Each Inspection Item Number will be addressed.

Background

A paper titled "Development of In-Service Inspection Safety Philosophy for U.S.A Nuclear Power Plants" by S.H. Bush and R.R. MacCary was reviewed which defined the philosophy behind the development of the Inspection Category N in the 1971 Edition of the ASME Section XI Code.

This Paper stated:

> *The special examination category N covers the examination of the interior surfaces and internal components of the reactor vessel; it is considered one of the most critical examination requirements in the A.S.M.E Section XI Code. Among the consideration contributing to the development of this examination category were the reported experiences and difficulties encountered in operating facilities. These interior examination areas should assure:*
>
> a. *Inspection of all internal support attachments welded to the reactor vessel whose failure could result in reactor core disarrangement.*
>
> b. *Discovery of any loose parts which might have accumulated at the bottom of the reactor vessel during service.*
>
> c. *Detection of undue wear as a result of flow-induced vibrations of components of the reactor core structure.*
>
> d. *Verification of the overall structural integrity of the core structure, including supplementary internal components such as moisture separators, material surveillance capsules, instrumentation, and reactor control rod assembly guides.*

Item (a) above is interpreted to be addressed by the VT-1 and VT-3 for Examination Category B-N-2, Item Numbers B13.20 and B13.30 "Interior Attachments within and beyond the beltline region".

Item (b) above is interpreted to be addressed by VT-3 for Examination Category B-N-1, Item Number B13.10, "Vessel Interior" covering "the spaces above and below the reactor core."

Item (c) above is interpreted to be addressed by the VT-3 for Examination Category B-N-2, Item Number B13.40 "Core Support Structure".

Item (d) above is also interpreted to be addressed by the VT-3 for Examination Category B-N-2, Item Number B13.40 "Core Support Structure" except that the supplementary components were not included when the Examination Categories B-N-1, B-N-2, B-N-3, or B-I-1 replaced the former Examination Category N in the 1974 edition. The reactor vessel interior surfaces referred to in the former Examination Category N were addressed by Examination Category B-I-1 in the 1974 Edition as sample "clad patches." However, this examination category was eliminated in the 1977 Edition.

The following table provides the different components that make up the reactor vessel internals (those that are **bolded** are those components that are normally accessible during refueling outages) taken from System Description "Reactor Vessel and Internal Components" SD-262 Rev. 5:

Component	Description
Jet Pump Assemblies	Eight jet pump assemblies are located in two semi-circular groups in the downcomer annulus, between the core shroud and the reactor vessel wall. Each jet pump assembly consists of two jet pumps and a common inlet header. Each stainless steel jet pump consists of a driving nozzle, suction inlet, throat or mixing section, and diffuser. Each driving nozzle, suction inlet, and throat are joined together as a removable unit. Each diffuser is welded to the shroud support ring (baffle plate) between the shroud and the vessel wall. High-pressure water from the recirculation pumps is supplied to each pair of jet pumps through a riser pipe welded to the recirculation inlet nozzle. A riser brace is welded to cantilever beams extending from pads on the reactor vessel wall
Top Guide (Core Grid)	The top guide (core grid) is a circular structure formed by a series of stainless steel beams joined at right angles to form rectangular cells. These cells provide lateral support and guidance for the fuel assemblies. The square cell openings in the center of the grid accommodate the typical control cells which consist of four fuel assemblies and a control rod. The cell openings at the periphery of each quadrant are partitioned by braces and brackets to accept one peripheral fuel assembly which is not next to the control rod. Detent sockets are provided beneath the top guide to anchor neutron sources, Local Power Range Monitor (LPRM) in-core detectors and the Source Range Monitor/Intermediate Range Monitor (SRM/IRM) dry tubes. The top guide is held in radial and rotational position within the shroud by fitted wedge blocks. Vertical movement of the top guide is prevented by hold downs.
In-Core Flux Monitor Assemblies	Four (4) Source Range, six (6) Intermediate Range, and twenty (20) Local Power Range detector assemblies are positioned in the reactor core. The Source and Intermediate Range detectors are located within dry tubes that serve as pressure boundaries. These detectors enter the vessel from below, pass through guide tubes in the lower plenum, and terminate in a spring plunger assembly at the core grid. Each neutron detector is inside of a shuttle tube, which is moveable within the dry tube, thus allowing axial positioning of the detectors within the core. The 20 Local Power Range detector assemblies (strings) enter the core from above, pass through the core plate and seat with a mating flange on the lower vessel head. Each Local Power Range detector assembly tube contains four detectors. The upper end of each assembly is attached to the core grid by a spring plunger assembly, similar to the Source and Intermediate Range detectors. Lateral alignment, below the core plate, is provided by stainless steel guide tubes that originate at the lower detector flange and terminate at the core plate. The guide tubes of all of the neutron detectors are secured by a lattice-work of clamps, tie bars and spacers. All bolts and clamps are welded after assembly to prevent loosening during reactor operation.

Component	Description
Core Differential Pressure and Standby Liquid Control Line	The differential pressure and Standby Liquid Control line serves a dual function within the reactor vessel. The line is used to inject the Standby Liquid Control solution into the coolant stream and to sense the differential pressure across the core support plate. The line enters the reactor vessel at a point below the core shroud as two concentric pipes. In the lower plenum, the two pipes separate. The inner pipe terminates near the lower shroud with a perforated length below the core support plate. This section is used to sense the pressure below the core support plate during normal operation and to inject the Standby Liquid Control solution, if required. The outer pipe terminates immediately above the core plate and senses the pressure above the core plate as part of the CRD system (drive and cooling water differential pressure), the jet pump differential pressure, and the core spray sparger break detection instrumentation system.
Surveillance Sample Holders	Vessel sample specimens are mounted inside the reactor vessel. These samples, which contain metal typical of the vessel wall, weld material and weld/wall interface, are occasionally removed and tested to verify NDTT calculations. The surveillance sample holders are welded baskets containing impact and tensile specimen capsules. The baskets hang from brackets on the inside wall of the reactor vessel near the midplane of the active core. The radial positions are chosen to expose the specimens to the same environment and maximum neutron flux experienced by the reactor vessel itself.
Core Shroud (portions of OD are accessible)	The core shroud is a stainless steel cylindrical assembly that provides a partition to separate the upward flow of coolant through the core from the downcomer annulus flow. This partition separates the core region from the downcomer annulus, thus providing a floodable region following a hypothetical recirculation line break. The volume enclosed by the core shroud is characterized by three regions, each with a different shroud diameter. The upper shroud has the largest diameter and surrounds the core discharge plenum, which is bounded by the shroud head on top and the top guide below. The central portion of the shroud surrounds the active fuel and forms the longest section of the shroud. This section has the intermediate diameter and is bounded at the bottom by the core support plate boundary. The lower shroud, surrounding part of the lower plenum, has the smallest diameter and is welded at the bottom to the reactor vessel shroud support ring (also called the baffle plate) and the shroud support legs.
Shroud Support Ring (portions of top are accessible)	The shroud support ring is an annular plate welded to the inner surface of the reactor vessel, just above the lower head. It, along with the shroud support legs, supports the weight of the core shroud, the steam separators, the jet pumps, and the core support plate, and forms the lower of the downcomer annulus. Penetration through the shroud support ring is provided for the discharge of each of the sixteen jet pumps.

Component	Description
Fuel Support Pieces (certain locations are accessible depending on fuel shuffle)	The fuel support pieces are of two basic types: peripheral and four-lobed. The peripheral fuel support pieces, which are welded to the core support plate, are located at the outer edge of the active core, and are not adjacent to control rods. Each peripheral fuel support piece will support one fuel assembly and contains a replaceable orifice assembly designed to assure proper coolant flow to the fuel assembly. The four-lobed support pieces will each support four fuel assemblies and are provided with replaceable orifice plates to assure proper coolant flow distribution to each fuel assembly. The four-lobed support pieces rest within and on top of the control rod guide tubes, which are supported laterally by the core support plate. A slotted alignment lug on the four-lobed fuel support piece fits over a corresponding alignment pin in the core support plate.
Shroud Head and Steam Separator Assembly (portions are accessible depending on fuel shuffle)	The shroud head and steam separator assembly consists of an array of axial flow steam separators mounted on a dished closure head. The steam separator is the first of two stages of moisture removal in the path of the steam. The steam-water mixture exiting the core is of very poor quality (high moisture content). Prior to admitting this steam to the high-pressure turbine, almost all of its moisture must be removed. The entraining of moisture in a steam line is termed "carryover" and will cause severe damage to a turbine.
Feedwater Spargers	The feedwater spargers are perforated stainless steel headers, located in the mixing plenum above the downcomer annulus. A separate sparger is fitted onto each of the four 10-inch diameter feedwater nozzles and is shaped to conform to the curve of the vessel wall. Sparger end brackets are attached to vessel brackets to support the weight of the spargers. Wedge blocks are used to position the spargers away from the vessel wall. Feedwater flow enters the center of the spargers and is discharged radially inward and downcomer flow from the steam separators and dryers before it contacts the vessel wall.
Core Support Plate	The core support plate consists of a circular stainless steel plate stiffened with a rim and beam structure. Perforations in the plate provide lateral support and guidance for the control rod guide tubes, in-core flux monitor guide tubes, peripheral fuel support pieces and startup neutron sources. The core support plate also supports the peripheral fuel support pieces and startup neutron sources vertically. The entire assemble is bolted to a support ledge between the central and lower portions of the core shroud. Proper positioning of the core support plate is assured by alignment pins, which bear against the shroud.

Component	Description
Control Rod Guide Tubes and Housings	The control rod drive housings are inserted through the control rod drive penetrations in the reactor vessel bottom head and are welded to stub tubes extending into the reactor vessel. Each housing transmits a number of loads to the bottom head of the reactor. These loads include the weight of a control rod and control rod drive, which is bolted to the housing from below, a control rod guide tube, one four-lobed fuel support piece, and the four fuel assemblies which rest on the top of the fuel support piece. The control rod guide tubes, located inside the vessel, extend from the top of the control rod drive housing up through holes in the core support plate. Each tube is designed as the lateral guide for a control rod, and as the vertical support for a four-lobed fuel support piece and the four fuel assemblies surrounding the control rod drive housing which in turn transmits the weight of the guide tube, fuel support piece, and fuel assemblies to the reactor vessel bottom head via a stub tube.
Steam Dryer	The steam dryers remove additional moisture from the wet steam exiting the steam separators. The wet steam flows across the dryer vanes where rapid changes in direction separate the water from the steam. The separated water flows down through collecting troughs and tubes to the area above the downcomer annulus. A dryer seal skirt extends down into the water around the moisture separators and forms a seal between the wet steam plenum and the dry steam flowing out of the top of the dryers.
Core Spray Lines and Spargers	Two 100% capacity core spray lines each have an 8-inch penetration through the reactor vessel, located approximately at the same elevation as the steam separator standpipes. The penetrations enter the reactor vessel 180 degrees apart. After entering the vessel, the lines immediately divide and are routed to opposite sides of the reactor vessel. Clamps attached to the vessel wall support the lines. The lines are then routed downward into the downcomer annulus and pass through the upper shroud, immediately below the flange. The flow again divides as it enters the center of each semicircular header, which is routed halfway around the inside of the upper shroud. This arrangement forms two nearly complete circular core spray spragers, one directly above the other, both being located directly above the top guide grid perimeter. The ends of the two headers are supported by brackets to accommodate thermal expansion. The line routing and supports are designed to accommodate differential movement between the shroud and the vessel. Both core spray lines are essentially identical, except that the spargers are at a slightly different elevation in the shroud. The correct spray distribution pattern is provided by a combination of distribution nozzles pointed radially inward and downward from the headers.

B13.10 "Vessel Interior"

Footnote 1 of Table IWB-2500-1 states "Areas to be examined shall include the spaces above and below the reactor core that are made accessible for examination by removal of components during normal refueling outages". A number of Code interpretations have been issued to clarify the scope of both B-N-1 and B-N-2 and are shown below. The components other than core support structures that are within those spaces are not examined as clarified by question two of Interpretation XI-1-95-27, and the core support structures themselves are explicitly covered by Examination Category B-N-2. As clarified by question one of Interpretation XI-1-95-27, Examination Category B-N-1 requires a VT-3 examination in the spaces above and below the reactor core for loose or missing parts and debris as defined in IWA-2213, which is further clarified by acceptance standard IWB-3520.2(c) as "foreign materials or acculmulation of corrosion products in the interior of the reactor vessel that could interfere with control rod motion or could result in blockage of coolant flow through fuel."

B13.20 "Interior Attachments within Beltline Region" and

B13.30 "Interior Attachments beyond the Beltline Region"

The table below provides all interior attachments and the associated Item Number that has been assigned.

ASME Section XI Item Number	Interior Attachment	Location
B13.20	Jet Pump Riser Brace	Within Beltline Region
B13.20	Surveillance Sample Holder Brackets	Within Beltline Region
B13.30	Core Spray Piping Brackets	Beyond Beltline Region
B13.30	Steam Dryer Support Brackets	Beyond Beltline Region
B13.30	Feedwater Sparger Brackets	Beyond Beltline Region
B13.30	Guide Rod Brackets	Beyond Beltline Region
B13.30	Steam Dryer Holddown Brackets	Beyond Beltline Region

B13.40 "Core Support Structure"

As clarified by Interpretation XI-1-95-28, examination in spaces below the core is required only when they are made accessible by removal of components that are normally removed during refueling outages. Normal refueling outage activities consist of removal of the vessel head, steam dryer, steam separator and fuel. In addition, control rod blades are swapped out during a refueling outage. Control Rod Drives are swapped out during refueling outages, however this activity is based on the age of the drive and is not scheduled each refueling outage therefore not considered a normal activity.

The DAEC Updated Final Safety Analysis Report (UFSAR) includes the following in 4.1.2 "Reactor Internal Components".

> *The major reactor internal components are the core (fuel, channels, control blades, and incore instrumentation), core support structure (including the shroud, top guide, and core plate), shroud head and steam separator assembly, steam dryer assembly, feedwater spargers, core spray spargers, and jet pumps. Except for the Zircaloy in the reactor core, these reactor internals are made of stainless steel or other corrosion-resistant alloys. Of the preceding components, the fuel assemblies (including fuel rods and channel), control blades, incore instrumentation, shroud head and steam separator assembly, and steam dryers are removable when the reactor vessel is open.*

Therefore, the UFSAR defines the core support structure components as the shroud, top guide, and core plate (including those fuel support castings that are made available between fuel shuffles). These are the components that will be included in the examination item number B13.40.

Code Interpretations

Interpretation: *XI-1-95-27*
Subject: *Section XI, IWA-2213 and Table IWB-2500-1: Examination of Spaces Within the Reactor Vessel (1977 Edition, and Later Editions and Addenda Through 1995 Edition)*
Date Issued: *June 26, 1995*
File: *IN94-006A*

Question (1): Does the VT-3 examination required by Table IWB-2500-1, Examination Category B-N-1, Note 1, for spaces within the reactor vessel, include examination for loose or missing parts and debris as required by IWA-2213?

Reply (1): Yes.

Question (2): If in-vessel items, other than core support structures, addressed by Table IWB-2500-1, Examination Category B-N-1, are within the accessible spaces described by Note 1, are those in-vessel items themselves included in Examination Category B-N-1?

Reply (2): No.

Interpretation: *XI-1-95-28*
Subject: *Section XI, Table IWB-2500-1; Visual Examination - Below the Core (1977 Edition with Summer 1978 Addenda, and Later Editions and Addenda Through 1995 Addenda)*
Date Issued: *June 26, 1995*
File: *IN94-006B*

Question: Is it a requirement of Table IWB-2500-1, Examination Category B-N-1, or Examination Category B-N-2, and Note 1, to perform VT-3 visual examinations in spaces below the core when they are made accessible by removal of components even if those components are not normally removed during refueling outages?

Reply: No.

Interpretation: *XI-81-12*
Subject: *Section XI, Division 1, IWB-2500-1, Category B-N-1, Preservice Examination Requirements for Reactor Vessel Interior, 1980 Edition With Winter 1980 Addenda*
Date Issued: *December 18, 1981*
File: *BC81-593*

Question (1): Is it the intent of Section XI, Division 1, 1980 Edition with Addenda through Winter 1980, that the extent of the visual examination VT-3 required in Table IWB-2500-1, Examination Category B-N-1, item B13.10, is limited to those areas of the interior of the reactor vessel shell and bottom head that are made accessible for examination by removal of components which are normally removed during a normal refueling outage?

Reply (1): Yes.

Question (2): Is the interior of the vessel closure head considered a part of item B13.10, Examination Category B-N-1, Table IWB-2500-1?

Reply (2): No.

Interpretation: *XI-80-01*
Subject: *Section XI, Division 1, Table IS-251, Vessel Examination Interior Surfaces*
Date Issued: *March 7, 1980*
File: *BC80-151*

Question: Does Table IS-251, Examination Category N, of the 1971 Edition, including Addenda through Summer 1973, require visual examination of the interior surfaces beyond that made accessible by the normal refueling operation?

Reply: No; it requires the performance of a visual examination of the interior surfaces of the vessel, the internal components, and the space below the core, of only those areas made available by normal refueling operations until the end of the inspection interval is reached, at which time a visual examination is required on essentially all the surfaces

CONCLUSION

Based on the above discussion the following components will be examined under Categories B-N-1, B-N-2 and B-N-3.

Examination Category	Examination Item No.	Component	Comments
B-N-1	B13.10	N/A	Covers the spaces above and below the reactor core made accessible during normal refueling outages
B-N-2	B13.20	Jet Pump Riser Brace	Vessel Pad to Vessel Weld
B-N-2	B13.20	Surveillance Sample Holder Brackets	
B-N-2	B13.30	Core Spray Piping Brackets	
B-N-2	B13.30	Steam Dryer Support Brackets	
B-N-2	B13.30	Feedwater Sparger Brackets	

Examination Category	Examination Item No.	Component	Comments
B-N-2	B13.30	Guide Rod Brackets	
B-N-2	B13.30	Steam Dryer Holddown Brackets	
B-N-2	B13.40	Core Shroud	Accessible Surfaces defined as the area between Jet Pumps 16 & 1 and 8 & 9.
B-N-2	B13.40	Top Guide	
B-N-2	B13.40	Core Plate	Includes Accessible Fuel Support Castings

REFERENCES

1. "Development of In-Service Inspection Safety Philosophy for U.S.A Nuclear Power Plant" by S.H. Bush and R.R MacCary. Published in Periodic Inspection of Pressure Vessel Institution of Mechanical Engineers, London, 1972.
2. Duane Arnold Energy Center System Description "Reactor Vessel and Internal Components" SD-262 Rev. 5
3. ASME Section XI interpretations XI-1-95-27, XI-1-95-28, XI-81-12, and XI-80-01.

(5) TECHNICAL APPROACH AND POSITION NUMBER: TAP-I005

COMPONENT IDENTIFICATION

Code Classes:	1
References:	Reactor Vessel Nozzle-to-Shell Welds
Examination Categories:	B-D
Item Numbers:	B3.90

EXAMINATION REQUIREMENTS

ASME Section XI (2001 Edition with the 2003 Addenda), Table IWB-2500-1, Examination Category B-D, Item Number B3.90 requires a volumetric examination of the reactor vessel nozzle to vessel welds.

Footnote 4 states "The examination volumes shall apply to the applicable figure shown in Figs. IWB-2500-7(a) through (d)." The DAEC nozzles are of the configuration specified in Fig. IWB-2500-7(b) "Examination Zones in Flange Type Nozzles Joined by Full Penetration Butt Welds". This figure requires the examination of the entire weld volume and the adjacent base metal on each side of the weld for a distance of ½ t ($t_s/2$).

POSITION

The nozzle configuration does not allow examination from the nozzle side to obtain the required coverage of $t_s/2$. Examination can be performed on the nozzle side in the circumferential direction but not in the axial direction. Examination in both directions (circumferential and axial) can be completed from the vessel side. During the 3rd 10 year Interval relief requests were developed and submitted to the Nuclear Regulatory Commission. The relief requests were approved for that interval.

A review of the Ultrasonic Examination Procedures (ACP 1211.27 and ACP 1211.44) was completed.

ACP 1211.27 "Manual Ultrasonic Examination of Reactor Pressure Vessel Welds includes the following requirements for Nozzle-to-Shell Welds (paragraph 3.7(1)(c):

> *"For examination of nozzle-to-shell welds the clad to base metal interface, including a minimum of 15% t_s shall be examined in at least one radial direction using personnel qualified in accordance with Appendix VIII, Supplement 4. The outer 85% shall be examined in at least two orthogonal directions (i.e., one radial and one tangential within +/- 10°) using personnel qualified for single side access examination in accordance with Appendix VIII, Supplement 6. This procedure is not qualified for examination of the nozzle-to-shell weld, clad to base metal interface and the adjacent metal to a depth of 15%* t_s *(measured from the clad to base metal interface), when scanning in the circumferential direction."*

ACP 1211.44 "Ultrasonic Detection and Sizing of Reactor Pressure Vessel Nozzle to Shell Welds and Nozzle Inner Radius" includes the following requirements for Nozzle-to-Shell Welds (paragraph 6.1.2):

"This procedure is qualified for examination of the nozzle-to-shell weld when scanning in the circumferential direction. The examination volume shall include the clad to base metal interface, including a minimum of 15% t_s *and the adjacent base material on each side of the weld for a distance of ½ t,* (t_s*/2) from the toe of the weld (reference Attachment 1, Figure IWB-2500-7(a) or 7(b), as appropriate). This procedure may also be used when the volume has been modified by alternative requirements, e.g., Code Case N-613-1 "Ultrasonic Examination of Full Penetration Nozzles in Vessels, Examination Category B-D, Items Nos. B3.10 and B3.90, Reactor Nozzle-To-Vessel Welds, Figs. (IWB-2500-7(a), (b), (c), Section XI, Division 1, which allows .50" adjacent base material each side of the weld."*

The above procedures are written to exclude the examination in the radial direction on the nozzle side.

The Nuclear Regulatory Commission (NRC) recognized this and included in 10CFR50.55a(b)(2)(xv)(K)(3) the following:

50.55a(b)(2)(xv)(K)(3)	For examination of reactor pressure vessel nozzle-to-shell welds conducted from the outside of the vessel
50.55a(b)(2)(xv)(K)(3)(i)	The clad to base metal interface and the adjacent metal to a depth of 15 percent T, (measured from the clad to base metal interface) must be examined from one radial and two opposing circumferential directions using a procedure and personnel qualified in accordance with Supplement 4 to Appendix VIII, as modified by §§50.55a(b)(2)(xv)(B) and 50.55a(b)(2)(xv)(C), for examinations performed in the radial direction and Supplement 5 to Appendix 5 to Appendix VIII, as modified by §50.55a(b)(2)(xv)(J), for examinations performed in the circumferential direction.
50.55a(b)(2)(xv)(K)(3)(ii)	The examination volume not addressed by §50.55a(b)(2)(xv)(K)(3)(i) must be examined in a minimum of one radial direction using a procedure and personnel qualified for single sided examination in accordance with Supplement 6 Appendix VIII, as modified by§§50.55a(b)(2)(xv)(D), 50.55a(b)(2)(xv)(E), 50.55a(b)(2)(xv)(F), and 50.55a(b)(2)(xv)(G).

The NRC further clarified the situation in a letter dated December 5, 2003 to the PDI Chair. This letter addressed the outside diameter OD examination as follows:

"Examinations from the Outside Surface of the Vessel

For ultrasonic examinations performed from the outside surface of the nozzle-to-vessel weld, the nozzle itself may obstruct the examination in four directions on the inner 15% through-wall. The staff recognized the obvious nozzle restriction in 10CFR50.55a(b)(2)(xv)(K)(3)(i) which establishes a minimum three direction inspection requirement: one radial and two circumferential directions. The requirement is based on an assumption that nozzles would interfere with examinations performed from the nozzle side of the weld, thus creating an impractical condition. The application of paragraph (K)(3) is to establish minimum acceptable criteria for examinations that a licensee determined to be impractical from four directions.

10CFR50.55a(b)(2)(xv)(K)(3)(ii) provides the same flexibility for examination of the outer 85% from the outside surface as from the inside surface. Examinations are to be performed according to Supplement 6 as modified by 10CFR50.55a(b)(2)(xv)(G)(3). If restrictions inhibit an examination in two perpendicular directions, the impacted area must be examined in at least one radial direction."

The NRC concluded their letter by stating:

"In summary, the staff concludes that 10CFR50.55a(b)(2(xv)(K) requires the examination of nozzle-to-vessel welds to be performed in accordance with 10CFR50.55a(b)(2)(xv)(G) where achievable. The staff recognizes that restrictions often hinder nozzle-to-vessel weld examinations in four directions. When such restrictions prevent complete coverage, the maximum achievable coverage is acceptable provided the minimum coverage requirements in 10CFR50.55a(b)(2)(xv)(K) are satisfied."

A review of 10CFR50.55a(b)(2)(xv)(G) is needed to confirm that the ultrasonic procedures used at the DAEC are in compliance with the regulations. 10CFR50.55a(b)(2)(xv)(G) states:

50.55a(b)(2)(xv)(G)	When applying Supplement 4 to Appendix VIII, Supplement 6 to Appendix VIII, or combined Supplement 4 and Supplement 6, the following additional provisions must be used, and examination coverage must include:
50.55a(b)(2)(xv)(G)(1)	The clad to base metal interface, including a minimum of 15 percent T (measured from the clad to base metal interface), shall be examined from four orthogonal directions using procedures and personnel qualified in accordance with Supplement 4 to Appendix VIII.
50.55a(b)(2)(xv)(G)(2)	If the clad-to-base-metal-interface procedure demonstrated detectability of flaws with a tilt angle relative to the weld centerline of at least 45 degrees, the remainder of the examination volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction. This must be accomplished using a procedure and personnel qualified for single-side examination in accordance with Supplement 6. Subsequent examinations of this volume may be performed using examination techniques qualified for a tilt angle of at least 10 degrees.
50.55a(b)(2)(xv)(G)(3)	The examination volume not addressed by §50.55a(b)(2)(xv)(G)(1) is considered fully examined if coverage is obtained in one parallel and one perpendicular direction, using a procedure and personnel qualified for single sided examination when the provisions of §50.55a(b) (2)(xv)(G)(2) are met.

ACP1211.27 is qualified for examination of nozzle-to-vessel weld when scanning in the radial direction and for examination of the outer 85% of nozzle-to-vessel welds when scanning in the circumferential direction. Note that this procedure is also qualified for single side access in accordance with Appendix VIII Supplements 4, 6 and 10CFR50.55a.

ACP1211.44 is qualified for examination of nozzle-to-vessel welds when scanning in the circumferential direction of the examination in the inner 15% of the examination volume. Note that personnel are required to be qualified to Appendix VIII Supplement 4 single sided qualification.

CONCLUSION

Based on the discussion above the two procedures utilized (ACP1211.27 and ACP1211.44) meet the requirements of 10CFR50.55a for nozzle-to-vessel weld examinations performed at the DAEC and there is no need to submit a relief request to the NRC.

REFERENCES

1. 10CFR50.55a "Code and Standards"

2. NRC letter dated December 5, 2003 "Nozzle-To-Reactor Pressure Vessel Weld Coverage Issues"

3. ACP 1211.27 "Manual Ultrasonic Examination of Reactor Pressure Vessel Welds" Revision 1

4. ACP 1211.44 "Ultrasonic Detection and Sizing of Reactor Pressure Vessel Nozzle to Shell Welds and Nozzle Inner Radius" Revision 2.

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	Approved 01/31/07	0	Use of the DAEC Technical Requirements Manual for Snubber Visual Examination & Testing
NDE-R002	Approved 01/31/07	0	Approved use of PDI for Overlays in lieu of Supplement 11 to Appendix VIII
NDE-R003	Submitted	1	Requested to use BWRVIP-05 recommendations for reduced circumferential vessel weld exams
NDE-R004	No longer required as Code Case N-700 has been incorporated into RG 1.147		
NDE-R005	Approved 01/31/07	0	Risk Informed ISI for Class 1 B-F & B-J Welds and Class 2 C-F-2 Welds
NDE-R006	Approved 01/31/07	0	Request use of Code Case N-686 "Alternate Requirements for Visual Examinations"
NDE-R007	Approved 06/12/07	0	Alternative for pressure testing and visual examination of buried piping and components
NDE-R008	Approved 01/31/07	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
NDE-R009	Approved 10/13/08	0	Request to Allow use of the Provisions of IWA-4132 for stock rotation of Recirculation Pump Seal Flange Assemblies
NDE-R010	Submitted	1	Limited Examination on Welds HCC-C001, CUA-J024, RMA-J004 (ref NG-09-0539)
NDE-R011	Submitted	1	Request to use Code Cases N-504-2 and N-638-1 for Weld Overlay Repairs
NDE-R012	Approved 07/02/08	0	Request to allow use of Post Qualifying Seal Weld Procedures for Target Rock Valves
NDE-R013	Approved 08/29/08	0	Request to use Code Case N-702 for Nozzle-to-Vessel Welds and Inner Radius Sections
NDE-R014	Submitted	1	Relief Request for Alternative to ASME Section XI Requirements to Use Structural Weld Overlay Repairs as an Alternative Repair Technique at the Duane Arnold Energy Center (NG-10-0567)
NDE-R015	Submitted	0	Request for Authorization of Alternative Regarding Pressure Test Requirements (NG-10-0597)

(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.

Criteria	ASME OM Code 2003 Addenda (Subsection ISTD) Requirements	DAEC TRM Section T3.7.2 Requirements
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.
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 ±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.

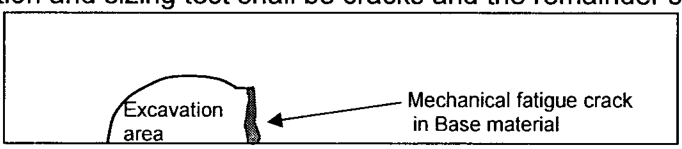
Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.1 inches be reported as being intrusions into the overlay material.

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

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**
(d) *Flaw Conditions*		
(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.

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)(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.**
(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.

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)(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		
(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.

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
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**
(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.

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**
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		
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).

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) 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.

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.

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.

Conclusion

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.

Duration of Proposed Alternative

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

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

"THIS RELIEF REQUEST IS NO LONGER REQUIRED AS CODE CASE N-700 HAS BEEN INCORPORATED INTO REGULATORY GUIDE (RG) 1.147"

(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	
RHR	4 (2)		High			B-F	1	1[9]		1[9]	

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]
		Medium (High)		None (IGSCC)	Low (Medium)	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	
CRD	4	Medium	High	None	Low	B-J	2	1		1	
						B-F	2	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	6	Low	Medium	None	Low	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 examination 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).

<u>DURATION OF PROPOSED ALTERNATIVE</u>

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

<u>Precedents</u>

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, Supplements 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).

9) Relief Request NDE-R009

Relief Request NDE-R009
Request To Allow Use Of The Provisions Of IWA-4132 For The Remainder Of The Fourth Ten-Year Inservice Inspection Interval

1. ASME Code Component(s) Affected

Code Class:	1
References:	IWA-4000 IWA-4132
Examination Category:	N/A
Item Number:	N/A
Description:	Relief to use the provisions of IWA-4132 for the stock rotation of Recirculation Pump Seal Flange Assemblies
Component Number	N/A

2. Applicable Code Edition and Addenda

American Society of Mechanical Engineers (ASME) Section XI, 2001 Edition with the 2003 Addenda.

3. Applicable Code Requirement

The Duane Arnold Energy Center (DAEC) fourth ten-year interval uses the ASME Section XI, 2001 Edition with the 2003 Addenda.

IWA-4132 "Items Rotated From Stock" states:

> "For snubbers and pressure relief valves rotated from stock and installed on components (including piping systems), the following requirements may be used in lieu of all other requirements of IWA-4000, provided the rotation is only for testing the removed items."
>
> (a) Items being removed and installed shall be of the same design and construction.
> (b) Items being removed shall have no evidence of failure at the time of removal.
> (c) Items being rotated shall be removed and installed only by mechanical means.
> (d) Items being installed shall previously have been in service.
> (e) Preservice inspections shall be performed as required by IWA-4500.
> (f) The Owner shall track the items to ensure traceability of inservice inspection and testing records.
> (g) Use of an Inspector and an NIS-2 form are not required.
> (h) Testing of removed snubbers, including required sample expansions, shall be performed in accordance with Subsection IWF.

4. Reason for Request

IWA-4132 of the 2001 Edition with the 2003 Addenda of ASME Section XI provides specific requirements for items rotated from stock for the purposes of testing. The items specifically mentioned are snubbers and relief valves. NextEra Energy Duane Arnold believes the criteria established in IWA-4132 for snubbers and pressure relief valves for testing can also be applied to the stock rotation of Recirculation Pump Seal Flange Assemblies for preventative maintenance, given that all the cited stock rotations involve mechanical joints on pressure retaining items and that none of these stock rotations involve a repair/replacement activity.

Non-mandatory Appendix J of the ASME Section XI code provides guidance to help the users of the code determine the applicability of IWA-4000. This appendix establishes that repair/replacement activities are separate from maintenance activities. The examples that are given for repair/replacement activities are:

1. removing weld or material defects;
2. reducing the size of defects to a size acceptable to the applicable flaw evaluation criteria;
3. performing welding or brazing;
4. adding items;
5. system changes, such as rerouting of piping;
6. modifying items;
7. rerating.

The stock rotation of seal flange assemblies does not fall under these examples. The disassembly and re-assembly of the mechanical connection for the stock rotation of spare Recirculation Pump Seal Flange assemblies is considered a maintenance activity and thus, the use of IWA-4132 for these activities provides an acceptable level of quality and safety, per 10 CFR 50.55a.

The requirements in IWA-4100 would not apply unless there was a repair/replacement activity performed on the spare Recirculation Pump Seal Flange Assembly prior to installation.

5. Proposed Alternative and Basis for Use

The Recirculation Pump Seal Flange Assembly consists of the seal cartridge and the seal flange. The seal flange is a pressure boundary component that supports the seal cartridge. The entire assembly (seal flange and seal cartridge) is bolted to the pump casing. During maintenance on the seal cartridge the complete assembly (seal cartridge and flange) is rotated as a single unit.

The normal activity of testing components under the criteria of IWA-4132 involves their removal from their installed location by mechanical means (i.e., disassembly and re-assembly of mechanical joints); this activity constitutes the basis for IWA-4132. IWA-4132 recognizes that, rather than re-install the same component tested, a similar component from stock can be rotated into service in its place. NextEra Energy Duane Arnold believes the criteria established in IWA-4132 for stock rotation of snubbers and pressure relief valves for testing can also be applied to the stock rotation of Recirculation Pump Seal Flange Assemblies for preventative maintenance.

Per IWA-4132:

(a) Items being removed and installed shall be of the same design and construction.
(b) Items being removed shall have no evidence of failure at the time of removal.
(c) Items being rotated shall be removed and installed only by mechanical means.
(d) Items being installed shall previously have been in service.
(e) Preservice inspections shall be performed as required by IWA-4500.
(f) The Owner shall track the items to ensure traceability of inservice inspection and testing records.
(g) Use of an Inspector and an NIS-2 form are not required.
(h) Testing of removed snubbers, including required sample expansions, shall be performed in accordance with Subsection IWF.

The above criteria are planned to be adopted by NextEra Energy Duane Arnold for stock rotation of Recirculation Pump Seal Flange Assemblies, as described below:

1. The seal flange assembly being installed will be a like-for-like replacement (same design) and built to the same construction code.
2. The seal flange assembly will have no evidence of failure (failure being defined as the pressure boundary failure).
3. The seal flange assembly will be removed and installed by mechanical means.
4. The seal flange assembly will have been previously installed.
5. Preservice and inservice inspections will be completed (i.e. Visual VT-1 of the bolting).
6. As the seal flange assembly is installed, the work will be controlled under the work order process. Unique identification for each item will be tracked and controlled.
7. An Inspector and NIS-2 form will not be used unless the item being installed has been repaired/replaced* in accordance with IWA-4000.
8. Testing of the seal flange assembly will be completed as needed to determine acceptability.

*If there is a repair/replacement activity performed on the item to be installed, NextEra Energy Duane Arnold plans to follow IWA-4000, including the use of an Inspector, NIS-2 form, and pressure testing.

NextEra Energy Duane Arnold has one spare Recirculation Pump Seal Flange Assembly for the DAEC, which is used as stock rotation for preventive maintenance. The Recirculation Pump Seal Flange Assembly is rotated from stock when the mechanical seal starts to show signs of wear prior to its failure. This spare Recirculation Pump Seal Flange Assembly has been previously installed and subsequently refurbished by replacing seal components which are not pressure retaining. NextEra Energy Duane Arnold considers this stock rotation to be a maintenance function (disassembly and re-assembly of mechanical joints) and not a repair/replacement activity, per IWA-4000.

Since IWA-4132 does not specifically include the stock rotation of Recirculation Pump Seal Flange Assemblies, NextEra Energy Duane Arnold requests the use of the criteria stated above for the stock rotation of Recirculation Pump Seal Flange Assemblies as an alternative to the requirements of IWA-4000 for the fourth ten-year interval.

It is important to note that IWA-4132 of the 2004 Edition of the ASME Section XI code includes preventative maintenance in the rotation of stock items.

Because the basic activity of disassembly and re-assembly of the mechanical joint is the same between the requested activity and that currently permitted, the proposed alternative would provide an acceptable level of quality and safety, and would not adversely impact the health and safety of the public.

6. Duration of Proposed Alternative

This alternative will be used for the remainder of the DAEC fourth ten-year inspection interval.

10) Relief Request NDE-R010

Component Identification

Code Class:	Class 1
References:	ASME Code, Section XI, Subarticle IWB-2500 Table IWB-2500-1
Examination Categories:	B-A, R-A
Item Numbers:	B1.40, and R1.16
Description:	Reactor Vessel Head-to-Flange Weld, and Piping Welds.
Component Numbers:	See Table A for Component Identification

Applicable Code Edition and Addenda

ASME Section XI 2001 Edition, 2003 Addenda

Code Requirement

Section XI (2001 Edition with the 2003 Addenda), Subarticle IWB-2500 states in part "Components shall be examined and tested as specified in Table IWB-2500-1." Table IWB-2500-1, Category B-A, Item B1.40, requires a volumetric examination of applicable Class 1 pressure retaining welds, which includes essentially 100% of weld length once during the ten year interval.

Relief Request NDE-R005 was approved on January 31, 2007 allowing the use of Risked Informed (RI) Inservice Inspection (ISI) for Class 1 and 2 welds. This relief request states that the original list of intended credited Class 1 welds may have been substituted on specific occasions with similar welds due to accessibility issues that would have resulted in reduced exam volumes. NextEra Energy Duane Arnold chooses welds for examination that are classified within the same risk matrix classification segment, using the same treatment criteria as those originally selected.

Reason for Request

The Duane Arnold Energy Center (DAEC) construction permit was issued in 1970 and the operating license was issued in 1974. The reactor vessel was designed and installed to ASME Section III, 1965 Edition, with the Summer 1967 Addenda. The parameters for accessibility for ISI were not requirements at that time and therefore were not necessarily factored into component and system configurations, thereby creating conditions where ASME Section XI Code required examination coverage of Class 1 welds cannot be obtained.

10 CFR 50.55a recognizes the limitations to in-service inspection of components in accordance with Section XI of the ASME Code that are imposed due to early plants' design and construction, as follows:

> 10 CFR 50.55a(g)(1): For a boiling or pressurized water-cooled nuclear power facility whose construction permit was issued prior to January 1, 1971, components (including supports) must meet the requirements of paragraphs(g)(4) and (5) of this section to the extent practical.
>
> 10 CFR 50.55a(g)(4): Throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) which are classified as ASME Code Class 1, Class 2, and Class 3 must meet the requirements, except design and access provisions and pre-service examination requirements, set forth in Section XI of editions of the ASME Boiler and Pressure Vessel Code... to the extent practical within the limitation of design, geometry and materials of construction of the components.
>
> 10 CFR 50.55a(g)(5)(iii): If the licensee has determined that conformance with certain code requirements is impractical for its facility, the licensee shall notify the Commission and submit, as specified in § 50.4, information to support the determinations.

Table A - Limited Exam Weld List

Exam Category	Item Number	Component Number	Period Examined	Code Coverage	Description
B-A	B1.40	HCC-C001	1	76.05%	Reactor Vessel Head-to-Flange Weld
R-A	R1.16	CUA-J024	1	50%	Reactor Water Cleanup Piping Weld
		RMA-J004	1	50%	Recirculation Manifold to Riser Piping Weld

Reactor Vessel Head to Flange Weld – HCC-C001

This weld is the Head to Flange Weld which can only be examined from the head surface. The examination is limited to approximately 76.05% and is limited due to the configuration of the weld. Note that this weld examination is divided up into thirds, with each third being examined each period. The examination completed was from Stud Holes 60 to 20. There is no feasible option in order to examine the remaining 23.95%.

The Nondestructive Examination (NDE) procedure used for this examination incorporates the examination techniques qualified under Appendix VIII of the ASME

Section XI Code by the Performance Demonstration Initiative (PDI). That procedure was approved under Relief Request NDE-R008 on January 31, 2007.

Reactor Water Cleanup Class 1 Weld CUA-J024

This weld is between a containment penetration and a motor operated valve (MO-2701). The valve side of the weld is not accessible for scanning due to geometry. This weld is the only weld in the risk segment (CU-007) so there is no other weld to select. The consequence evaluation determined this weld to be high because it is the weld between the containment and the isolation valve (MO-2701). The material for the items welded together is A182 F316 forging penetration to an A351 CF8M cast valve. Per Generic Letter 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping," "Inspection Schedules" position (2), this weld would be considered in the Intergranular Stress Corrosion Cracking (IGSCC) Category A. The A182 F316 is solution heat treated which is an acceptable process for resistance to IGSCC. The staff position states:

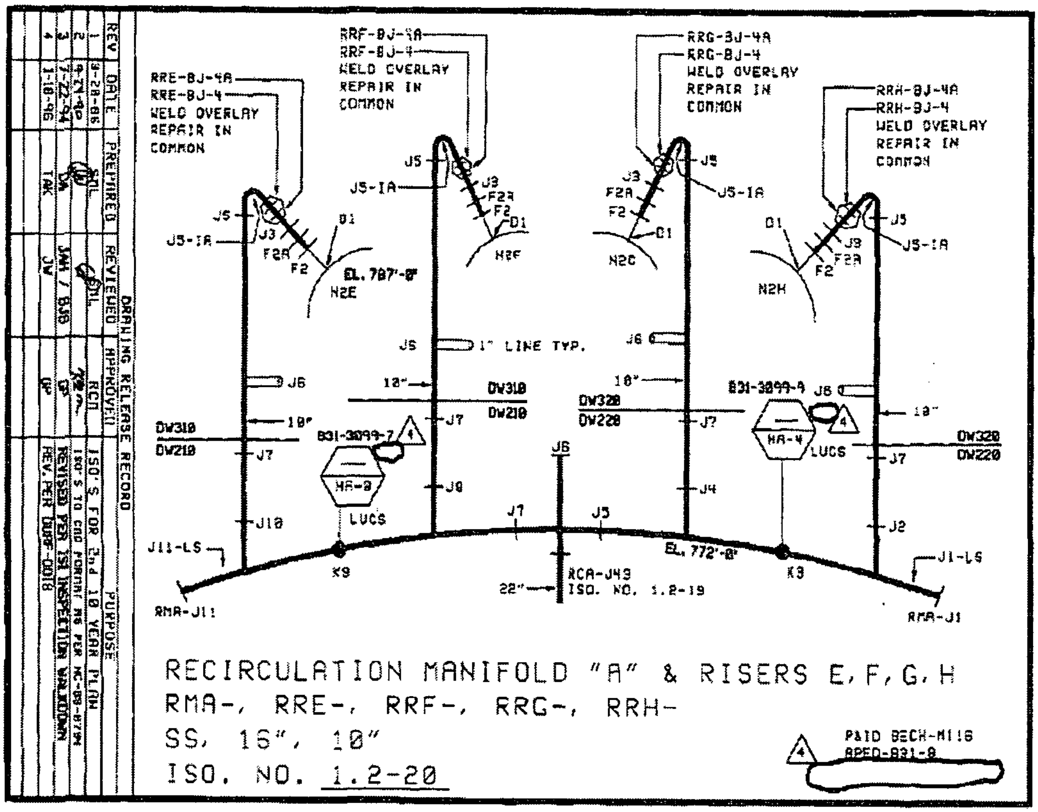
> *"Although castings with higher carbon content than 0.035% are not considered to be resistant to sensitization, welds joining such castings (in the form of pump and valve bodies) to piping have been relatively free of IGSCC. This may be attributed to a favorable residual stress distribution, as calculations have indicated. For this reason, weld joining resistant material to pumps and valves will be considered to be resistant welds, and included in IGSCC Category A. If extensive weld repairs were performed the residual stress may be unfavorable, in which case such welds should be included in Category D."*

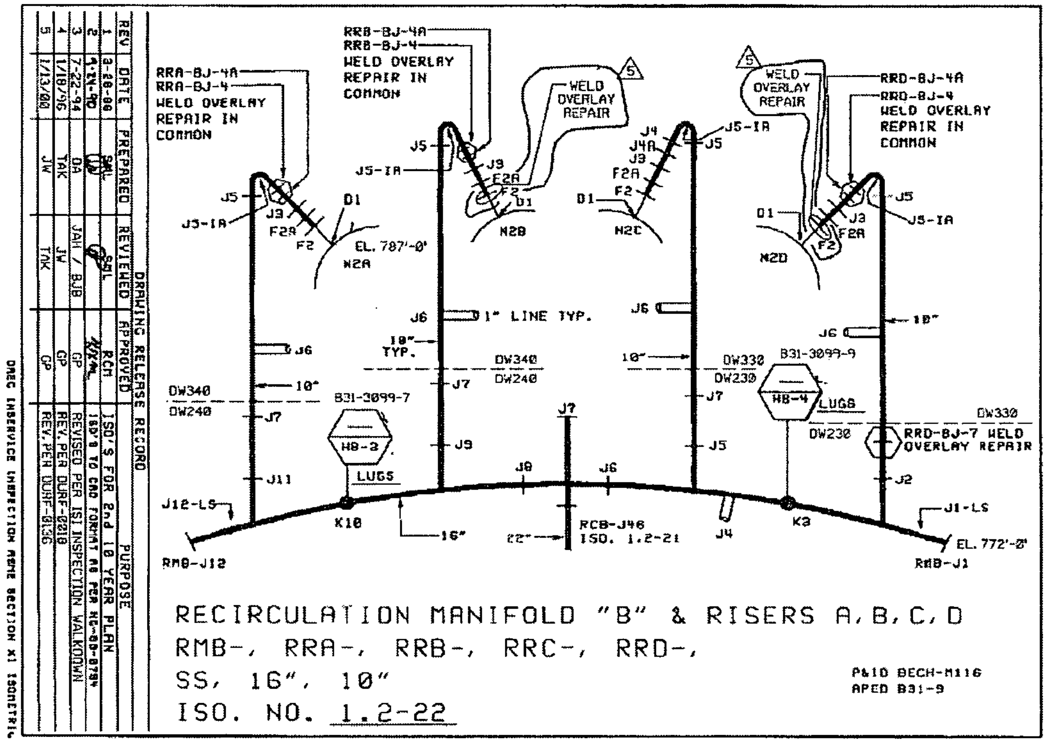
Therefore, this weld is considered resistant to IGSCC. The volume examined was from the A182 F316 penetration side of the weld. The weld was not examined from the A351 CF8M cast valve side due to the configuration of the valve (see photo that follows).

Since this weld is resistant to the degradation mechanism of IGSCC and there is no other degradation mechanism that has been identified for this weld, the volume examined is acceptable.

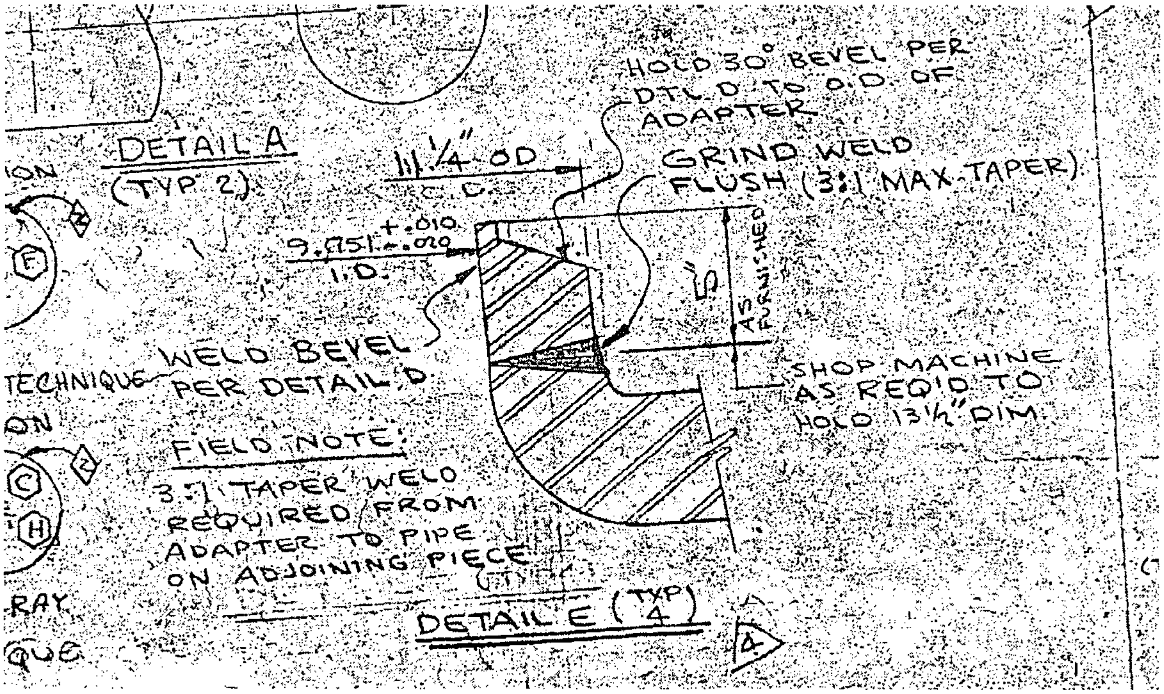
Recirculation System Class 1 Weld RMA-J004

This weld is the branch connection of the recirculation manifold to the recirculation riser line. The manifold side is not accessible for scanning due to geometry. There are a total of eight welds with this configuration (8 recirculation risers coming off the two manifolds); see ISI Isometrics 1.2-20 and 1.2-22 that follow.


RECIRCULATION MANIFOLD "A" & RISERS E, F, G, H
RMA-, RRE-, RRF-, RRG-, RRH-
SS, 16", 10"
ISO. NO. 1.2-20
RRE-BJ-4A
RRE-BJ-4
WELD OVERLAY REPAIR IN COMMON
RRF-BJ-4A
RRF-BJ-4
WELD OVERLAY REPAIR IN COMMON
RRG-BJ-4A
RRG-BJ-4
WELD OVERLAY REPAIR IN COMMON
RRH-BJ-4A
RRH-BJ-4
WELD OVERLAY REPAIR IN COMMON
1" LINE TYP.
RCA-J43
ISO. NO. 1.2-19
EL. 787'-0"
EL. 772'-0"
LUCS
DRAWING RELEASE RECORD
REV
DATE
PREPARED
REVIEWED
APPROVED
PURPOSE
ISO'S FOR 2nd 10 YEAR PLAN
REVISED PER ISI INSPECTION WALKDOWN
REV. PER DWR-0018



The detailed configuration of the branch connection weld provided below is taken from the construction drawing (B31-G001-RD-N-B3). As the configuration shows it does not allow an examination to be performed from the manifold side.



All eight manifold to riser welds were solution annealed which is an acceptable process for resistance to IGSCC. Each weld was identified in its own risk category so there is no other weld that could be selected to obtain a higher examination coverage. These welds were determined to have Thermal Transients as a degradation mechanism under the RI-ISI Program. The Degradation Mechanism Evaluation Document (M453-047) states:

For the most part, the RCR System heats up and cools down slowly and uniformly so it is not susceptible to the thermal shocks that result in TT degradation mechanism. However, at the beginning of shutdown cooling, the RCR System does receive a double-shock from the RHR System (cold fluid and then hot fluid into formerly hot lines). The double-shock was analyzed for the RCR main loop discharge, manifold, and riser piping. The results indicate that all of these pipe segments are potentially susceptible to the TT degradation mechanism.

NextEra Energy Duane Arnold proposes to perform the ultrasonic examination on one more of the eight welds (which would only receive 50% coverage) in lieu of performing the examination on just one weld. One weld will be examined from each loop.

Performing two of the eight welds (each receiving 50% coverage) is considered an acceptable alternative.

Proposed Alternative and Basis for Use

In accordance with 10 CFR 50.55a(g)(5)(iii), relief is requested for the components listed in Table A on the basis that the required examination coverage of "essentially 100 percent" is impractical due to physical obstructions and the limitations imposed by design, geometry and materials of construction. NextEra Energy Duane Arnold performed qualified examinations that achieved the maximum, practical amount of coverage obtainable within the limitations imposed by the design of the components.

Additionally, for the Class 1 examination Category B-P, a VT-2 examination is performed on the subject components of the Reactor Coolant Pressure Boundary during system pressure tests each refueling outage. The examination was completed during the 2009 refueling outage and no evidence of leakage was identified for these components.

Based on the above, with due consideration of the earlier plant design, the underlying objectives of the Code required volumetric examinations have been met. The examinations were completed to the extent practical and no evidence of unacceptable flaws was detected. VT-2 examinations performed on the subject Class 1 components during system pressure testing each refueling outage (in accordance with examination Category B-P) provide continued assurance that the structural integrity of the subject components is maintained. Additionally, the DAEC Water Chemistry Program and inerted primary containment environment provide added measures of protection for the component materials.

Duration of Proposed Alternative

Relief is requested for the Fourth Ten year Interval of the Inservice Inspection Program for the DAEC.

Precedents

1) NRC Letter dated May 1, 2008, “Safety Evaluation for Request for Relief from IWB-2500 and IWC-2500 Requirements to Allow Performance of Limited Examinations of Various Welds for the Third 10-Year Interval of the Inservice Inspection Program (TAC No. MD5669).”

2) NRC Letter dated October 18, 1999, “Safety Evaluation of Third 10-Year Interval Inservice Inspection Program Plan Requests for Relief for Duane Arnold Energy Center (TAC No. MA4151)” specifically for Relief Request NDE-R028.

3) NRC Letter dated March 23, 1998, “Evaluation of Third 10-Year Inservice Inspection Interval Program Plan Requests for Relief for the Duane Arnold

Energy Center (TAC No. M95412)" specifically for Relief Requests NDE-R006, NDE-R007, NDE-R008, NDE-R009, and NDE-R010.

References

None

11) Relief Request NDE-R011

Alternative to ASME Section XI Repair Requirements
to use Code Cases N-504-2 and N-638-1 for Weld Overlay Repairs
at the Duane Arnold Energy Center (NDE-R011)

1.0 ASME Code Component(s) Affected

Code Class:	1
References:	ASME Section XI, 2001 Edition, including and through the 2003 Addenda ASME Section XI, Case N-504-2 ASME Section XI, Case N-638-1 NUREG-0313 Rev 2 Generic Letter 88-01 BWRVIP-75 DAEC Fourth Ten Year ISI Plan – NRC Approved Relief Request NDE-R002, "Relief to use the PDI Program for Implementation of Appendix VIII, Supplement 11 requirements," and Relief Request NDE-R005 "Risked Informed ISI for Class 1 B-F & B-J Welds and Class 2 C-F-2 Welds (ML070090357)
Examination Categories:	R-A (B-F)
Item Number:	R1.16 (B5.10)
Description:	Alternative Repair for the RRC-F002 and RRF-F002 Recirculation Inlet Nozzle, Safe-end-to-Nozzle Welds
Component Numbers:	RRC-F002 Recirculation Inlet Nozzle Safe-end Weld RRF-F002 Recirculation Inlet Nozzle Safe-end Weld

2.0 Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, including Addenda through 2003.

3.0 Applicable Code Requirement

IWA-4421(a) and IWA-4611.1(a) require removal of the detected flaw.

IWA-4610(a) requires that the area to be welded shall be pre-heated to 300°F minimum for gas tungsten arc welding (GTAW).

IWA-4610(a) requires that thermocouples shall be used to monitor process temperatures.

IWA-4631(b) specifies that the surface of the completed weld on the ferritic steel shall not exceed 100 square inches.

IWA-4633.2(c) specifies that the first three layers of the weld shall be deposited with heat inputs within ±10% of that used in the procedure qualification test. Subsequent layers shall be deposited using heat input equal to or less than that used for layers beyond the third in the procedure qualification.

IWA-4633.2(c) also specifies that at least one layer of weld reinforcement shall be deposited and then this reinforcement shall be removed substantially flush with the surface surrounding the weld.

4.0 Reason for Request

The request is based on restoring the structural integrity of the RRC-F002 and RRF-F002 recirculation inlet nozzle safe-end-to-nozzle weld joints using technically sound welding practices and non-destructive examination (NDE), while limiting repair personnel exposure to the maximum extent practical. The following cited Code articles identify the actions that would be required if the repair were conducted in accordance with the Code without exception.

IWA-4421(a) and IWA-4611.1(a) require defect removal in this case. The repair cavity would extend through wall since outer diameter (OD) removal would be required. Internal diameter (ID) removal of the indication would be impractical since it would require the removal of the thermal sleeve and jet pump riser from the reactor interior.

IWA-4610(a) requires the area to be welded shall be pre-heated to 300°F minimum for GTAW. Since the nozzle will remain full of water, establishing the 300°F minimum pre-heat temperature cannot be achieved.

IWA-4610(a) also requires the use of thermocouples to monitor process temperatures. Due to the personnel exposure associated with the installation and removal of the thermocouples, the nozzle configuration, and because the nozzle will be full of water, a contact pyrometer will be used, in lieu of thermocouples, to verify pre-heat and interpass temperature limits are met.

IWA-4631(b) specifies the surface of the completed weld on the ferritic steel shall not exceed 100 square inches. Restoring the structural integrity of the safe-end-to-nozzle weld with the weld overlay will require welding on more than 100 square inches of surface on the low alloy steel base material.

IWA-4633.2(c) specifies the first three layers of the weld shall be deposited with heat inputs within ±10% of that used in the procedure qualification test. Subsequent layers shall be deposited using heat input equal to or less than that used for layers beyond the third in the procedure qualification. Code Case N-638-1 allows for layers beyond the third to exceed the heat input, provided it is in accordance with the procedure qualification records (PQRs).

IWA-4633.2(c) also specifies that at least one layer of weld reinforcement shall be deposited and then this reinforcement shall be removed substantially flush with the surface surrounding the weld. The weld reinforcement will not be removed flush to the surface.

5.0 Proposed Alternative and Basis for Use

A full structural weld overlay repair is proposed for the safe-end-to-nozzle weldments. The nozzle material is SA-508 Class 2 low alloy steel. The safe-end is Alloy 600 SB-166. The existing weld material is Alloy 82, with Alloy 182 buttering.

The weld overlay will be designed consistent with the requirements of NUREG-0313, Revision 2 (which was implemented by Generic Letter (GL) 88-01), Code Case N-504-2, "Alternative Rules for Repair of Classes 1, 2, and 3 Austenitic Stainless Steel Piping," Code Case N-638-1, "Similar and Dissimilar Metal Welding Using Ambient Temperature GTAW Temper Bead Technique," and IWB-3640, ASME Section XI 2001 Edition, including Addenda through 2003 with Appendix C.

Welder Qualification And Welding Procedures

All welders and welding operators will be qualified in accordance with ASME Section IX and any special requirements of ASME XI or applicable code cases. Qualified personnel under the vendor's (Welding Services Inc. (WSI)) welding program will perform the weld overlay repair.

Welding Procedure Specification (WPS) WPS 03-43-T-804-102967 (machine GTAW with cold wire feed) for welding SFA-5.14, ERNiCrFe-7A, UNS N06054, F-No. 43 (commercially known as Alloy 52M) will be used.

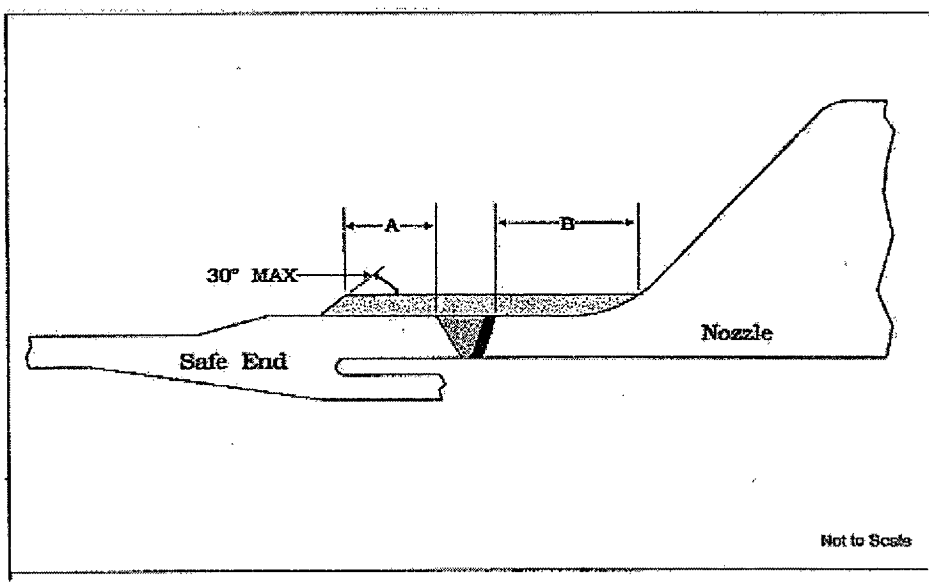
If repairs to the overlay are required, manual GTAW for welding SFA-5.14, ERNiCrFe-7A, UNS N06054, F-No. 43 (commercially known as Alloy 52M) will be used. In the unlikely event of a through-wall defect, UNS W86152, F No. 43 (commercially known as Alloy 152) will be used to seal any defect if it is greater than 0.125 inch from the P-3 nozzle material before beginning the structural weld overlay using GTAW.

Welding Wire and Electrodes

A consumable welding wire, highly resistant to stress corrosion cracking (SCC), was selected for the overlay material. Alloy 52M contains a nominal 30 wt% Cr that imparts excellent resistance to SCC. Where localized repairs are required, Alloy 52M will be used.

Weld Overlay Design

The weld overlay will extend around the full circumference of the safe-end-to-nozzle weldment location in accordance with NUREG-0313, Rev. 2, Code Case N-504-2, and GL 88-01. The overlay length will extend across the projected flaw intersection with the outer surface beyond the extreme axial boundaries of the flaw. The design thickness and length has been computed in accordance with the guidance provided in Code Case N-504-2 and ASME Section XI, IWB-3640, 2001 Edition including Addenda through 2003 and Appendix C. The overlay will completely cover the area of the flaw and other Alloy 182 susceptible material with the highly resistant Alloy 52M weld filler material.



Design Dimensions		
A	B	Thickness
2.0 inch	Overlay to be gently blended into nozzle to minimize stress concentration and to accommodate temper bead weld passes	0.500 inch

To provide the necessary weld overlay geometry, it will be necessary to weld on the low alloy steel nozzle base material. A temper bead welding approach will be used for this purpose following the guidance of ASME Section XI Code Case N-638-1, "Similar and Dissimilar Metal Welding Using Ambient Temperature Machine GTAW Temper Bead Technique." This Code Case provides for machine GTAW temper bead weld repairs to P-No. 3, Group No. 3, nozzle base material at ambient temperature. The temper bead approach was selected because temper bead welding supplants the requirement for post-weld heat treatment (PWHT) of the heat-affected zone (HAZ) in welds on low alloy steel material. Also, the temper bead welding technique produces excellent toughness and ductility as demonstrated by welding procedure qualification in the HAZ of welds on low alloy steel materials, and, in this case, results in compressive residual stresses on the inside surface, which assists in inhibiting SCC. This approach provides a comprehensive weld overlay repair and increases the volume under the overlay that can be examined.

The overlay length conforms to the guidance of Code Case N-504-2, which satisfies the stress requirements.

Examination Requirements

NUREG-0313, Rev. 2, and Code Case N-504-2, specify ultrasonic test (UT) using methods and personnel qualified in accordance with ASME Section XI, Appendix I. The UT techniques to be used for the final post-weld examination have been qualified through the Electric Power Research Institute (EPRI) NDE Center, which satisfies the requirements of ASME Section XI, Appendix I. Furthermore, NUREG-0313 states that the UT to be performed in accordance with the requirements of the applicable Edition and Addenda of ASME Section XI. ASME Section XI, 2001 Edition including Addenda through 2003 is the Code of record for the DAEC fourth 10-year Inservice Inspection Interval. Therefore, the acceptance criteria that will be used for the UT will be IWB-3130, "Inservice Volumetric and Surface Examinations," and ASME Section XI Non-mandatory Appendix Q, "Weld Overlay Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Weldments," as clarified under Exceptions to Code Case N-638-1 Paragraph 4.0(b). In addition, an NRC-approved relief request (NDE-R002) for the DAEC (ML070090357) allows the use of the Performance Demonstration Initiative (PDI) Program for implementation of Appendix VIII, Supplement 11 requirements for the examination of piping welds with overlays.

The examination requirements for the weld overlay repair are summarized in Table 1. No final post-weld examinations will be performed until 48 hours has elapsed after completion of welding. This is required to detect any possible hydrogen-induced cracking that may occur in the low alloy steel nozzle HAZ.

TABLE 1
Examination Requirements

Exam Description	Method	Technique	Reference
As Found Flaw Detection	Auto UT	PDI Qualified Implementing ASME Section XI Appendix VIII	IWB-3514.4
Pre-weld UT Thickness	Manual UT	0°	N-504-2
Surface Prior to Welding	PT	Color Contrast (Visible) Penetrant	IWA-4611.1(a) N-504-2(c) N-638-1,4.0(a)
Final Weld Overlay Surface	PT	Color Contrast (Visible) Penetrant	IWA-4634 N-504-2(j) N-638-1,4.0(b)
Final Weld Overlay for Thickness	UT	0°	IWA-4634 N-504-2(j) N-638-1,4.0(b)
Final Weld Overlay and Outer 25% of the Underlying Wall Thickness Volumetric Pre-service	UT	PDI Qualified Implementing ASME Section XI Appendix VIII RR-NDE-002	IWA-4634 IWB-3514.4 N-504-2(j) N-638-1,4.0(b) Appendix Q

Pressure Testing

The completed repair shall be given a system leakage test in accordance with ASME Section XI, IWA-5000, since the pressure boundary has not been penetrated (no leakage has occurred). In the event an unexpected through wall defect is identified, either before or during the repair, an additional exception from the hydrostatic pressure test requirements defined in Code Case N-504-2 will be needed. A system leakage test will be performed in accordance with ASME Section XI, IWA-5000 of the 2001 Edition with the 2003 Addenda. Precedence for use of a leak test at normal operating temperature and pressure, in lieu of a hydrostatic test, has been set with Code Case N-416-1, which has been incorporated in ASME Section XI starting with the 1998 Edition, 1999 Addenda.

Pre-heat and PWHT Requirements

Pre-heat and PWHT are typically required for welding on low alloy steel material. ASME Section III specifies PWHT on P-No. 3, Group No. 3, base materials unless temper bead welding is performed under limited restrictions (area and depth limits). ASME Section XI, 2001 Edition including Addenda through 2003, specifies 300°F minimum pre-heat be used for temper bead welding. PWHT cannot be performed and the pre-heat requirements would necessitate draining the reactor pressure vessel (RPV) and a portion of the recirculation system piping. This would result in unacceptable radiation dose rates. Therefore, consistent with ALARA practices and prudent utilization of outage personnel, the RPV will not be drained for this activity. The nozzle and connected piping will be full of water.

Alternatives to Code Case N-504-2

Code Case N-504-2 Applicability to Nickel Based Austenitic Steel

Code Case N-504-2 was prepared specifically for austenitic stainless steel material. An alternate application for nickel based austenitic materials (Alloy 52M) is needed due to the specific materials and configuration of the existing nickel based alloy weld and buttering (Alloy 82 and Alloy 182).

Exception to Code Case N-504-2, Requirement (b)

Code Case N-504-2, Requirement (b) requires the weld overlay shall be low carbon (0.035% maximum) austenitic stainless steel. A nickel-based filler is required and Alloy 52M has been selected to be used.

Exception to Code Case N-504-2, Requirement (e)

Code Case N-504-2, Requirement (e) requires the first two layers of the weld overlay to have a ferrite content of at least 7.5 FN (Ferrite Number). These measurements will not be performed for this overlay since the nickel alloy filler is a fully austenitic material.

Exception to Code Case N-504-2, Requirement (h)

Code Case N-504-2, Requirement (h) specifies that a system hydrostatic test shall be performed in accordance with IWA-5000 if the flaw penetrates the pressure boundary. In the event the flaw becomes through wall, leak testing only, in accordance with ASME Section XI, IWA-5000, will be performed.

Alternatives to Code Case N-638-1

Exception to Code Case N-638-1 Paragraph 1.0(a)

Code Case N-638-1 paragraph 1.0(a) specifies that the maximum weld area on the finished surface shall be 100 square inches. Restoring the structural integrity of the safe-end-to-nozzle weld with the weld overlay will require welding on more than 100 square inches of surface on the low alloy steel base material.

Exception to Code Case N-638-1 Paragraph 4.0(b)

Code Case N-638-1 paragraph 4.0(b) specifies that the final weld surface and the band around the area (1.5T width or 5 inches, whichever is less) shall be examined using surface and ultrasonic methods when the completed weld has been at ambient temperature for at least 48 hours. The UT shall be in accordance with ASME Section XI Appendix I. Full UT of the 1.5T band will not be performed.

Exception to Code Case N-638-1 Paragraph 4.0(c)

Code Case N-638-1 paragraph 4.0(c) specifies that the area from which weld-attached thermocouples have been removed shall be ground and examined using a surface examination method. Thermocouples will not be used. Calibrated pyrometers will be utilized to monitor pre-heat & interpass temperatures.

Basis For The Alternatives

IWA-4421(a) and IWA-4611.1(a) require defect removal in this case. The repair cavity would extend through wall since OD removal would be required. The ID is inaccessible due to the thermal sleeve. Therefore, the flaw will not be removed. Structural weld overlays covering flaws are permitted by Code Case N-504-2, provided the necessary weld overlay geometry is used. Therefore, this alternative provides an acceptable level of quality and safety.

IWA-4610(a) requires the area to be welded shall be pre-heated to 300°F minimum for GTAW. Since the nozzle will remain full of water, establishing the 300°F minimum pre-heat temperature cannot be achieved. Code Case N-638-1, paragraph 1.0(b) provides for machine GTAW temper bead weld repairs to P-No. 3, Group No. 3, nozzle base material at ambient temperature. The ambient temperature temper bead approach was selected because temper bead welding supplants the requirement for PWHT of the HAZ in welds on low alloy steel material. Also, the temper bead welding technique produces excellent toughness and ductility, as demonstrated by welding procedure qualification, in HAZ of welds on low alloy steel materials. Welding procedure qualifications have been successfully performed using Alloy 52M welds on P-No. 3, Group No. 3, base material using the ambient temperature temper bead technique. Therefore, this alternative provides an acceptable level of quality and safety.

IWA-4610(a) also requires the use of thermocouples to monitor process temperatures. Due to the personnel exposure associated with the installation and removal of the thermocouples, the nozzle configuration, and because the water in the line containing the nozzle will not be drained, thermocouples will not be used to verify that pre-heat and interpass temperature limits are met. In lieu of thermocouples, a contact pyrometer will be used to verify pre-heat temperature and interpass temperature compliance with the WPS requirements. The use of a contact pyrometer provides equivalent temperature monitoring capabilities and is recognized as acceptable calibrated measuring and test equipment (M&TE). Therefore, this alternative provides an acceptable level of quality and safety.

IWA-4631(b) specifies the surface of the completed weld on the ferritic steel shall not exceed 100 square inches. Restoring the structural integrity with the weld overlay of the safe-end-to-nozzle weld will require welding on more than 100 square inches of surface on the low alloy steel base material. If this limit were maintained, the length of weld overlay extension on the nozzle base material would be limited to approximately 2.25 inches, including the taper. This distance could be justified as sufficient to provide load redistribution from the weld overlay back into the nozzle without violating ASME III stress limits for primary local and bending stresses, and secondary and peak stresses. However, this length would not permit a complete UT of the outer 25% of the nozzle and safe-end thickness as specified by Code Case N-504-2. The overlay will extend to the transition taper of the low alloy steel nozzle so that qualified UT of the required volume can be performed. Therefore, this alternative provides an acceptable level of quality and safety.

Code Case N-432 has always allowed temper bead welding on low alloy steel nozzles without limiting the temper bead weld surface area. The two additional conditions required by N-432, that are not required by Code Case N-638, are that temper bead welds have pre-heat applied and that the procedure qualification be performed on the same specification, type, grade, and class of material. As previously discussed, elevated pre-heat necessitates draining of the RPV and a portion of the recirculation system piping. This would result in unacceptable radiation dose rates.

The ASME Code committees have recognized that the 100 square inches restriction on the surface area is unnecessarily limiting and Code Case N-638-3 has been issued to increase the surface area limit to 500 square inches. The code case attempts to combine the features of Code Case N-432 and N-638 into a single code case. The supporting analysis for the code case (EPRI Technical Report 1008454, "Proposed Code Case, Expansion of Temper Bead Repair") concluded that the residual stresses are not detrimentally changed by increasing the surface area of the repair and increasing the HAZ tempering is unaffected by the weld overlay application. Therefore, this alternative provides an acceptable level of quality and safety.

IWA-4633.2(c) specifies the first three layers of the weld shall be deposited with heat inputs within ±10% of that used in the procedure qualification test. Subsequent layers shall be deposited using heat input equal to or less than that used for layers beyond the third in the procedure qualification. Code Case N-638-1 allows for layers beyond the third to exceed the heat input provided it is in accordance with the PQRs. Therefore, this alternative provides an acceptable level of quality and safety.

IWA-4633.2(c) also specifies that at least one layer of weld reinforcement shall be deposited and then this reinforcement shall be removed, to be substantially flush with the surface surrounding the weld. The weld overlay is austenitic and thus, there is no need to remove the final layer. Also, overlays, by definition, cannot be substantially flush with the surrounding surface. Overlays are permitted per Code Case N-504-2. The toe of the weld on the low alloy steel nozzle shoulder will be indexed between layers such that proper HAZ tempering will result. Therefore, this alternative provides an acceptable level of quality and safety.

Code Case N-638-1 is approved (with one limitation) for generic use in Regulatory Guide (RG) 1.147, Revision 14, and was developed for both similar and dissimilar metal welding using ambient temperature machine GTAW temper bead technique. The welding methodology of Code Case N-638-1 will be followed for the overlay, whenever welding within the 0.125-inch minimum distance from the low alloy steel nozzle base material.

Code Case N-504-2 is approved (with one limitation) for generic use in RG 1.147, Revision 14, and was developed for welding on and using austenitic stainless steel material. An alternate application for nickel-based and low alloy steel materials is proposed due to the specific configuration of this weldment. The weld overlay proposed is austenitic material having a mechanical behavior similar to austenitic stainless steel. It is also compatible with the existing weld and base materials.

The methodology of Code Case N-504-2 is to be followed, except for the following:

<u>Exception to Code Case N-504-2, Requirement (b)</u>

Code Case N-504-2, Requirement (b) requires the weld overlay shall be low carbon (0.035% maximum) austenitic stainless steel.

A consumable welding wire highly resistant to SCC was selected for the overlay material. This material, designated as UNS N06054, F-No. 43, is a nickel based alloy weld filler material, commonly referred to as Alloy 52M and will be deposited using the machine GTAW process with cold wire feed. Alloy 52M contains about 30 wt% chromium, which imparts excellent corrosion resistance to the material. By comparison, Alloy 82 is identified as a SCC-resistant material in NUREG-0313 Revision 2 and contains nominally 20 wt% chromium, while Alloy 182 has a nominal chromium content of 15 wt%. With its higher chromium content than Alloy 82, Alloy 52M provides an even higher level of resistance to SCC consistent with the requirements of the Code Case. Therefore, this alternative provides an acceptable level of quality and safety.

Exception to Code Case N-504-2, Requirement (e)

Code Case N-504-2, Requirement (e) requires the first two layers of the weld overlay to have a ferrite content of at least 7.5 FN (Ferrite Number).

The composition of nickel-based Alloy 52M is such that delta ferrite does not form during welding, because Alloy 52M welds are 100% austenitic and contain no delta ferrite due to the high nickel composition (approximately 60 wt% nickel). Consequently, delta ferrite measurements will not be performed for this overlay. Therefore, this alternative provides an acceptable level of quality and safety.

Exception to Code Case N-504-2, Requirement (h)

Code Case N-504-2, Requirement (h) specifies that a system hydrostatic test shall be performed in accordance with IWA-5000 if the flaw penetrates the pressure boundary.

Leak testing in accordance with ASME Section XI (2001 Edition with the 2003 Addenda), IWA-5000, will be performed. Precedence for use of a leak test at normal operating temperature and pressure in lieu of a hydrostatic test has been set with Code Case N416-1 that has been incorporated in ASME Section XI beginning in the 1998 Edition with the 1999 Addenda. Therefore, this alternative provides an acceptable level of quality and safety.

Exception to Code Case N-638-1 Paragraph 1.0(a)

Code Case N-638-1 paragraph 1.0(a) specifies that the maximum weld area on the finished surface shall be 100 square inches. Restoring the structural integrity with the weld overlay of the safe-end-to-nozzle weld will require welding on more than 100 square inches of surface on the low alloy steel base material. The weld overlay will cover approximately 180 square inches of the low alloy steel nozzle.

Code Case N-432 allows temper bead welding on low alloy steel nozzles without limiting the temper bead weld surface area. The two additional conditions required by N-432, that are not required by Code Case N-638-1, are that temper bead welds have pre-heat applied and that the procedure qualification be performed on the same specification, type, grade and class of material. As previously discussed, elevated pre-heat necessitates draining of the RPV and a portion of the recirculation system piping. By removing the water in the pipe, nozzle area, and (in vessel) inlet riser a large amount of shielding is removed. The radiation dose rates at the weld overlay location would increase, thereby significantly increasing personnel dose.

The ASME Code committees have recognized that the 100 square inches restriction on the surface area is unnecessarily limiting and Code Case N-638-3 has been issued to increase the surface area limit to 500 square inches. The code case attempts to combine the features of Code Case N-432 and N-638 into a single code case. The supporting analysis for the code case is found in EPRI Technical Report 1008454, "Expansion of Temperbead Repair: Proposed Code Case," which concluded that the residual stresses are not detrimentally changed by increasing the surface area of the repair and increasing the HAZ tempering is unaffected by the weld overlay application. The technical basis that justifies exceeding 100 square inches of surface area for repair welds is found in EPRI Technical Report 1003616, "Additional Evaluations to Expand Repair Limits for Pressure Vessels and Nozzles." This technical report describes an ANSYS Finite Element Analysis (FEA) conducted on the Nine Mile Point - Unit 2 feedwater nozzle weld overlay repair. The analysis consisted of modeling the welding processes for both thermal and mechanical respects. The two overlays were modeled; one was 100 square inches, the other was extended to blend into the nozzle radius to achieve greater than 100 square inches surface area repair currently permitted by the ASME Code requirements. Comparison of the residual stresses of the two overlays showed that the effect of extending the overlay to the nozzle radius minimally impacted the residual stress profile and, in some cases, slightly increased the beneficial compressive stresses on the nozzle inner diameter. Therefore, this alternative provides an acceptable level of quality and safety.

<u>Exception to Code Case N-638-1 Paragraph. 4.0(b)</u>

Code Case N-638-1 Paragraph 4.0(b) specifies that the final weld surface and band area (1.5T width or 5 inches, whichever is less) shall be examined using surface and ultrasonic methods when the completed weld has been at ambient temperature for at least 48 hours. The UT shall be in accordance with ASME Section XI, Appendix I. Surface exams will be performed. IWA-4634 requires UT of the weld only. Any laminar flaws in the weld overlay will be evaluated in accordance with ASME Section XI Non-mandatory Appendix Q, Paragraph Q-4100, except, as allowed by IWB-3132.3, any flaws that exceed the acceptance standards of Table IWB-3410-1 are acceptable for continued service, without repair, if an analytical evaluation, performed in accordance with IWB-3600, meets the acceptance criteria of IWB-3600. Full UT of the 1.5T band will not be performed. The weld overlay will extend into the blend radius of the nozzle beyond the length required by Code case N-504-2 for structural reinforcement. This extension onto the blend radius eliminates a stress riser on the nozzle and provides additional OD surface area for UT examination of the defect area. UT examination on the nozzle beyond the overlay will not provide any information regarding the area of the defect that required repair. Additionally, such UT would likely be unsatisfactory when applied to the nozzle blend radius, where the toe of the weld overlay resides. The UT return signal would be difficult to obtain and to interpret. Alternatively, surface examination will assure that no defects have been created at the toe of the weld overlay. Therefore, this alternative provides an acceptable level of quality and safety.

Exception to Code Case N-638 Paragraph. 4.0(c)

Code Case N-638-1 paragraph 4.0(c) specifies that the area from which weld-attached thermocouples have been removed, shall be ground and examined using a surface examination method. Due to the personnel exposure associated with the installation and removal of the thermocouples, the nozzle configuration, and because the nozzle will be full of water, thermocouples will not be used to verify that the pre-heat and interpass temperature limits are met. In lieu of thermocouples, a contact pyrometer will be used to verify pre-heat temperature and interpass temperature compliance with the WPS requirements. Therefore, this alternative provides an acceptable level of quality and safety.

The use of overlay filler material that provides excellent resistance to SCC develops an effective barrier to flaw extension. Also, temper bead welding techniques produce excellent toughness and ductility in the weld HAZ low alloy steel materials, and in this case, results in compressive residual stresses on the inside surface that help to inhibit further SCC. The design of the overlay for the safe-end-to-nozzle weldment uses methods that are standard in the industry. There are no new or different approaches in this overlay design which would be considered either first-of-a-kind or inconsistent with previous approaches. The overlay will be designed as a full structural overlay in accordance with Code Case N-504-2. The temper bead welding technique that will be implemented in accordance with Code Case N-638-1 will produce a tough, ductile, corrosion-resistant overlay.

Use of Code Cases N-504-2 and N-638-1 has been accepted in RG 1.147, Revision 14, with the following limitations as providing an acceptable level of quality and safety.

Code Case N-504-2 Limitation

The provisions of Section XI, Non-mandatory Appendix Q, "Weld Overlay Repair of Class 1, 2, and 3 Austenitic Stainless Steel Piping Weldments," must also be met, as noted in RG 1.147.

The DAEC will meet the associated requirements contained in this non-mandatory Appendix Q.

Code Case N-638-1 Limitation

UT examinations shall be demonstrated for the repaired volume using representative samples which contain construction type flaws. The acceptance criteria of NB-5330 of Section III edition and addenda approved in 10CFR50.55a apply to all flaws identified in the repair volume

The DAEC will implement this limitation.

NextEra Energy Duane Arnold concludes that the alternative repair approach described above presents an acceptable level of quality and safety to satisfy the requirements of 10CFR50.55a(a)(3)(i).

6.0 Duration of Proposed Alternative

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

7.0 Precedents

The observed flaws at DAEC are consistent with the documented SCC observed at DAEC in 1999 on the safe-end-to-nozzle welds N2B and N2D. Similar flaws have been observed at other BWRs, including Perry, Nine Mile Point – Unit 2, Susquehanna – Unit 1, and more recently at Hope Creek Generating Station.

12) Relief Request NDE-R012

10 CFR 50.55a Request Number NDE-R012
Seal Weld Procedure Qualification

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(ii)
Hardship or Unusual Difficulty without Compensating Increase
in Level of Quality and Safety

ASME CODE COMPONENT(S) AFFECTED

Code Class: 1

Component Numbers: PSV-4400, PSV-4401, PSV-4402, PSV-4405, PSV-4406, PSV-4407, and six uninstalled spare Main Steam Relief Valves (MSRVs)

Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, 2003 Addenda

APPLICABLE CODE REQUIREMENT

IWA-4221, Construction Code and Owner's Requirements

IWA-4421, "Construction Code and Owner's Requirements," states that:

"(a) An item to be used for repair/replacement activities shall meet the Owner's Requirements. Owner's Requirements may be revised, provided they are reconciled in accordance with IWA-4222. Reconciliation documentation shall be prepared.
(b) An item to be used for repair/replacement activities shall meet the Construction Code specified in accordance with (1), (2), or (3) below.

(1) When replacing an existing item, the new item shall meet the Construction Code to which the original item was constructed.
(2) When adding a new component to an existing system, the Owner shall specify a Construction Code that is no earlier than the earliest Construction Code used for construction of the system or of any originally installed component in that system.
(3) When adding a new system, the Owner shall specify a Construction Code that is no earlier than the earliest Construction Code used for other systems that perform a similar function.

(c) As an alternative to (b) above, the item may meet all or portions of the requirements of different Editions and Addenda of the Construction Code, or Section III when the Construction Code was not Section III, provided the requirements of IWA-4222 through IWA-4226, as applicable, are met. Construction Code Cases may also be used. Reconciliations required by this Article shall be documented. All or portions of later different Construction Codes may be used as listed below:

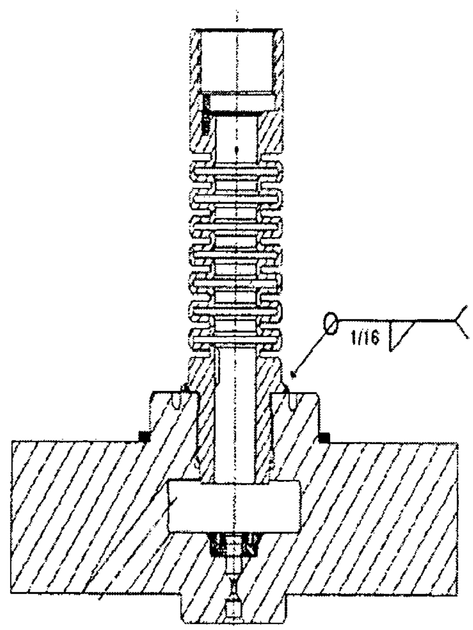
(1) Piping, piping subassemblies, and their supports: B31.1 to B31.7 to Section III.

(2) Pumps, valves, and their supports: from B31.1 to Draft Code for Pumps and Valves for Nuclear Power to Section III.
(3) Vessels and their supports: Section VIII to Section III.
(4) Atmospheric and 0-15 psig (0–103 kPa) storage tanks and their supports: Section VIII, API 620, or API 650 to Section III."

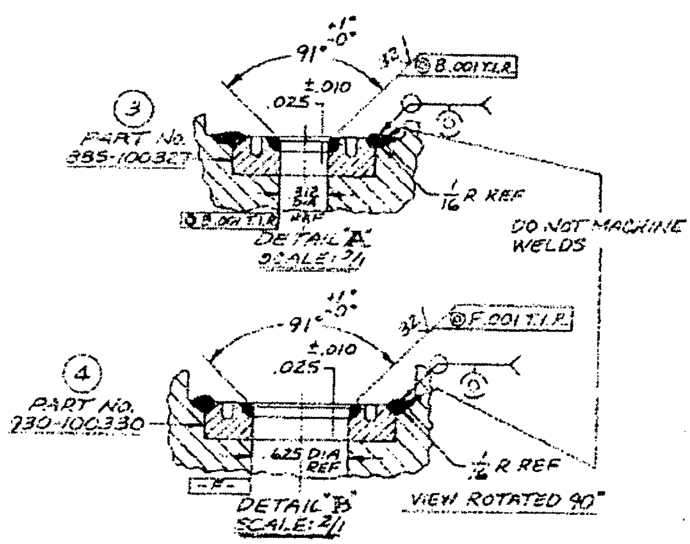
Installed and spare MSRVs were originally manufactured to the 1968 Edition, Winter 1968 Addenda of ASME, Section III, and General Electric Design Specification 21A9206 Rev. 6. All subsequent Repair/Replacement Activities were intended to meet the original construction code and design specification. All Repair/Replacement Activity seal welding was performed by the original Manufacturer (Target Rock) using the same Weld Procedure Specifications (WPSs) as used during original fabrication.

Reason for Request

Current weld procedure specifications for seal welding were qualified in accordance with the Manufacturer's standard rather than ASME requirements. The Manufacturer's standard included multiple surface Non-Destructive Examinations (NDE) and macro examinations of sectioned specimens. The three seal welds affected are the bellows-to-spacer plate seal weld, the pilot seat-to-body seal weld, and the second stage seat-to-body seal weld (see sketches; welds illustrated by black areas).



Bellows Seal Weld



Seat Seal Welds

The 1968 Edition, Winter 1968 Addenda of ASME Section III did not include fabrication requirements for valves or provide any requirements for seal welding.

The General Electric design specification required weld procedures to be qualified in accordance with ASME Section IX. However, the author of this design specification, and the original purchaser of the MSRVs (General Electric), believed that this requirement was never intended to be applicable to seal welding.

It is reasonable to assume that the 1968 Edition of ASME Section IX should have been used to qualify seal welding procedures, since the 1968 Edition of ASME Section IX, paragraph Q-10 (b), requires all welding to be qualified using reduced section tension specimens and guided bend specimens.

Pursuant to 10 CFR 50.55a, "Codes and Standards," Paragraph (a)(3)(ii), the Duane Arnold Energy Center requests authorization to post-qualify those seal welding procedures in accordance with ASME Section IX requirements.

To qualify the weld procedures at this time in accordance with ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality or safety, as further described below:

First, in order to meet the ASME Code requirements to qualify weld procedures prior to production welding and subsequent code stamping, all seal welds would have to be removed and re-welded using the same welding procedure that is now pre-qualified.

Second, removal of existing seal welds would require that the MSRV be completely disassembled, the seat rings replaced, and the reassembled valve tested. This unnecessary welding evolution could potentially degrade the carbon steel casting. Therefore, replacement of existing seal welds is considered a hardship, or unusual difficulty, without a compensating increase in the level of quality and safety.

Proposed Alternative and Basis for Use

Target Rock has completed three procedure qualification records (PQR's) using the same seal welding parameters as in the original seal welding procedures and weld coupons were tested in accordance with the 2004 Edition, 2006 Addenda, of ASME Section IX. All tensile and bend testing was found acceptable per ASME Section IX requirements, 2004 Edition, 2006 Addenda. All three seal weld WPSs have been revised to reference the new PQRs that were qualified via tensile and bend testing.

These post-qualifying PQRs verify that the seal welds made with the original seal welding WPSs meet all tensile and bend test requirements and justify continued use. The revised seal welding WPSs that now reference the new PQRs are planned to be used during future Repair/Replacement activities, if performed by Target Rock.

DURATION OF PROPOSED ALTERNATIVE

The proposed alternative will be used for the entire Fourth Ten-Year Interval of the Inservice Inspection Program for Duane Arnold Energy Center.

Precedents

None

13) Relief Request NDE-R013

10 CFR 50.55a Request Number NDE-R013
Alternative to Nozzle to Vessel Weld and Inner Radius Examinations

Proposed Alternative in Accordance with 10 CFR 50.55a(a)(3)(i) which provides for an acceptable level of quality and safety

ASME CODE COMPONENT(S) AFFECTED

Code Class: 1
Component Numbers: N1, N2, N-3, N-5, N6, N7, N8, N11, N12, and N16 Nozzles (see Attachment 1 for specific nozzle identifications)
Examination Category: B-D
Item Number: B3.90 and B3.100
Description: Alternative to Table IWB-2500-1 (Inspection Program B)

Applicable Code Edition and Addenda

The DAEC is currently in its fourth 10-year interval and is committed to the ASME Code Section XI, 2001 Edition, 2003 Addenda. Additionally, for ultrasonic examinations Section XI, Appendix VIII, "Performance Demonstration for Ultrasonic Examination Systems," of the 2001 Edition is implemented as required (and modified) by 10CFR50.55a(b)(2)(xv)..

APPLICABLE CODE REQUIREMENT

Table IWB-2500-1 "Examination Category B-D, Full Penetration Welded Nozzle in Vessels – Inspection Program B"

Class 1 nozzle-to-vessel weld and nozzle inner radii examination requirements are given in Item Number B3.90 "Nozzle-to-Vessel Welds" and B3.100 "Nozzle Inside Radius Section." The method of examination is volumetric. All nozzles with full penetration welds to the vessel shell (or head) and integrally cast nozzles are examined each interval. All of the nozzle assemblies identified in attachment 1 are full penetration welds.

Reason for Request

The identified nozzles (see attachment 1) are scheduled for examination prior to the end of the DAEC's current inspection interval. The proposed alternative provides an acceptable level of quality and safety, and the reduction in scope could provide a dose savings of as much as 25.2 Rem over the remainder of the interval.

Proposed Alternative and Basis for Use

Proposed Alternative:

Pursuant to 10CFR50.55a(a)(3)(i), relief is requested from performing the required examinations on 100% of the identified nozzle assemblies (see attachment 1). Alternatively, in accordance with Code Case N-702 a minimum of 25% of the nozzle-to-vessel welds and inner radius sections, including at least one nozzle from each system and nominal pipe size, would be examined. For the nozzle assemblies identified in attachment 1, this would mean one from each of the groups identified below:

Group	Total Number	Number to be examined	Comments
Recirculation Outlet (N1)	2	2	This group does not meet Criteria 4 of the NRC SER[1]
Recirculation Inlet (N2)	8	2	Four completed in RFO20 (2007)
Vessel Instrumentation (N11, N12, N16)	6	2	One scheduled in RFO22 (2010) and one scheduled in RFO23 (2012)
Core Spray (N5)	2	1	One completed in RFO20 (2007)
Nozzles on Vessel Top Head (N6, N7)	3	1	One scheduled in RFO21 (2009)
Jet Pump (N8)	2	1	One completed in RFO20 (2007)
Main Steam (N3)	4	1	One scheduled in RFO23 (2012)

Footnote 1 – the RPV wall thickness was taken from the Form N-1 Data Sheet.

Code Case N-702 stipulates that VT-1 examination may be used in lieu of the volumetric examination for the inner radii (Item No. B3.100). Note that the DAEC is not currently using Code Case N-648-1 and has no plans using the code case in the future. Volumetric examinations of all inner radii will be completed.

Basis for Use:

EPRI Technical Report 1003557, "BWRVIP-108: BWR Vessel and Internals Project Technical Basis for the Reduction of Inspection Requirements for the Boiling Water Reactor Nozzle-to-Vessel Shell Welds and Nozzle Blend Radii," provides the basis for Code Case N-702. The evaluation found that failure probabilities at the nozzle blend radius region and nozzle-to-vessel shell weld due to a Low Temperature Overpressure event are very low (i.e.$<1 \times 10^{6}$ for 40 years) with or without inservice inspection. The report concludes that inspection of 25% of each nozzle type is technically justified.

This report received an NRC Safety Evaluation (SER) dated December 19, 2007. In the SER, Section 5.0 "Plant Specific Applicability" requires each licensee who plan to request relief from

the ASME Code, Section XI requirements for RPV nozzle-to-vessel shell welds and nozzle inner radii sections may reference BWRVIP-108 report as the technical basis for the use of ASME Code Case N-702 as an alternative. However, each licensee should demonstrate the plant-specific applicability of the BWRVIP-108 report to their units in the relief request by showing that all the following general and nozzle-specific criteria are satisfied:

(1) the maximum RPV heatup/cooldown rate is limited to less than 115°F per hour. The DAEC surveillance that monitors reactor vessel heatup/cooldown (STP 3.4.9-01) limits the rate to less than 100°F for Curve B and less than 20°F for Curve A.
(2) For the Recirculation Inlet Nozzles the following criteria must be met:
   a. $(pr/t)/C_{RPV}<1/15$, the calculation for the DAEC N2 Nozzle results in 0.9748 which is less than 1.15
   b. $[p(ro2+ri2)/(ro2-ri2)]/C_{NOZZLE}<1.15$, the calculation for the DAEC N2 Nozzle results in 1.0923 which is less than 1.15.
(3) For the Recirculation Outlet Nozzles the following criteria must be met:
   a. $(pr/t)/C_{RPV}<1/15$, the calculation for the DAEC N1 Nozzle results in 1.17 which is higher than 1.15.
   b. $[p(ro2+ri2)/(ro2-ri2)]/C_{NOZZLE}<1.15$, the calculation for the DAEC N1 Nozzle results in 0.87 which is less than 1.15.

So based on the above information, the Recirculation Outlet Nozzle does not meet both criteria and Code Case N-702 would not be applied. See attachment 2 for details.

DURATION OF PROPOSED ALTERNATIVE

The proposed alternative will be used for the remaining portion of the fourth ten-year interval of the Inservice Inspection Program for Duane Arnold Energy Center.

Precedents

None

14) Relief Request NDE-R014

1.0 Component Identification

Code Classes: 1

References: ASME Code Section XI, 2001 Edition, including Addenda through 2003
ASME Section XI, Case N-504-4
NUREG-0313, Rev 2
Generic Letter 88-01
BWRVIP-75
DAEC Fourth Ten Year IS1 Plan – NRC Approved Relief Request NDE-R002, "Relief to use the PDI Program for Implementation of Appendix VIII, Supplement II requirements," (ML070090357)

Examination Category: R-A
Item Number: R1.16
Description: Alternative Repair for the RRA-F002A Recirculation Inlet Safe End to Safe End Extension Weld
Component Numbers: RRA-F002A Recirculation Inlet Safe End to Safe End Extension Weld and RRA-J003 Recirculation Safe End Extension to Pipe Weld

2.0 Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, including Addenda through 2003

3.0 Applicable Code Requirement

The applicable Code requirement for which relief is requested is ASME Code Section XI, 2001 Edition including Addenda through 2003, IWA-4410 and IWA-4611.

IWA-4410 states in part the following: "Welding, brazing, defect removal, metal removal by thermal methods, and installation shall be performed in accordance with the requirements of this Subarticle."

IWA-4611(a) states in part the following: "Defects shall be removed in accordance with IWA-4422.1."

4.0 Reason for Alternative

Dissimilar metal welds (DMWs), primarily consisting of Alloy 82/182 weld metal are frequently used in boiling water reactor (BWR) construction to connect stainless or Inconel safe ends to vessel and pipe nozzles, generally constructed of carbon or low alloy ferritic steel. These welds have shown a propensity for intergranular stress corrosion cracking (IGSCC) degradation in Boiling Water Reactor (BWR) environments.

This request is based on restoring the structural integrity of the RRA-F002A weld joint using technically sound welding practices and non-destructive examination (NDE), while limiting repair personnel exposure to the maximum extent practical. The following cited Code article identifies the actions that would be required if the repair were conducted in accordance with the Code without exception.

IWA-4421(a) requires defect removal in this case. The repair cavity would extend through wall since outer diameter (OD) removal would be required. Internal diameter (ID) removal of the indication would be impractical since it would require the removal of the thermal sleeve and jet pump riser from the reactor interior.

5.0 Proposed Alternative and Basis for Use

A full structural weld overlay repair is proposed for the safe-end-to-safe-end-extension weldment. The safe-end is Alloy 600 SB-166 austenitic nickel base Inconel forging (ASME Section II SB-166). The safe-end-extension is SA-336 Class F8 austenitic stainless steel forging (304 stainless steel). The full structural weld overlay will be extended beyond the safe end extension to stainless steel pipe weld to allow for ultrasonic examination of both of the welds, see Figure 1.

The weld overlay will be designed consistent with the requirements of NUREG-0313, Revision 2 (which was implemented by Generic Letter (GL) 88-01), the requirements specified in Attachment 1, and IWB-3640, ASME Section XI 2001 Edition, including Addenda through 2003 with Appendix C.

This proposed alternative (Attachment 1) is the result of industry's experience with weld overlay modifications for flaws suspected or confirmed to be caused by stress corrosion cracking and directly applies Alloy 52 or 52M weld material that is primarily being used for these weld overlays.

The ultrasonic examination of the completed overlay will be accomplished with personnel and procedures qualified in accordance with ASME Code, Section XI, 2001 Edition, for Appendix VIII, Supplement 11 (as approved by the NRC in Relief Request NDE-R002 (ML070090357)).

Structural Weld Overlay Design

The weld overlay satisfies all the structural design requirements of the pipe as specified in the Alternative Requirements shown in Attachment 1 for the original safe-end to safe-end extension welds. In particular, the design of the overlay will consider all the identified flaws, circumferential and axial, found during the initial UT examination. As shown in Figure 1, the weld overlay will completely cover the existing weld and will extend the Inconel safe-end and austenitic safe-end extension for the RRA-F002A. The weld overlay extends around the entire circumference from the vessel side of the safe end covering the safe-end extension and the safe end extension to stainless steel pipe weld and a distance onto the Type 304 stainless steel pipe, covering the RRA-J003 weldment. Alloy 52M filler metals are compatible with all the base materials and the dissimilar metal welds that will be covered by the overlay.

The weld overlay will be designed as one full structural overlay covering both welds, as illustrated in Figure 1. Postulated 100% through-wall flaws shall be assumed as specified in 2(b)(4) and (5), Attachment 1, for overlay length and thickness sizing per 2(b)(6) Attachment 1. Planar flaws detected during the acceptance examination will be characterized and flaw growth calculations performed using the flaw(s) detected plus the postulated 100% through-wall flaws.

To confirm the long-term condition of the safe-end area, a complete residual stress analysis of the Nozzle, safe end, safe end extension, and attached recirculation pipe assembly will be performed in which the weld overlay (WOL) repair will be modeled in the as-applied sequence. The final stress analysis report will provide the as-left stress distribution of the entire safe-end area, and include crack growth and fatigue assessments of the weld overlay repaired safe-end-to-safe-end extension joint. This analysis will provide NextEra Energy Duane Arnold the necessary information to properly classify these welds in the ISI program and to schedule future UT exams in accordance with BWRVIP-75.

The information from the final stress analysis report, and post-application UT examinations will be included in the Licensee Event Report (LER), or supplement thereto, that will be submitted after plant startup.

Welding

The welding will be performed in accordance with the approved weld procedure described in Attachment 1 using a machine gas tungsten-arc welding (GTAW) process for the RRA-F002A weld and adjacent RRA-J003 stainless steel weld with ERNiCrFe-7A (Alloy 52M) being used for the filler metal. In some instances of this process, flaws in the first layer have occurred in the portion of the overlay deposited on the austenitic stainless steel portions (safe ends, pipe etc.) of the assemblies.

The flaw characteristics previously observed above are indicative of hot cracking. This phenomenon has not been observed on austenitic stainless steel or Alloy 82/182 DMW portions of the assemblies when welding Alloy 52M thereon.

Studies have determined that this problem may be exacerbated when using Alloy 52M filler metal on austenitic stainless steel materials with higher sulfur content and high levels of silicon, as in the case of cast austenitic stainless steel.

Extensive test and field experience from WSI indicate that hot cracking can be a concern when the sulfur and silicon content in the diluted weld puddle equals or exceeds 0.014%. The impurity hot cracking threshold level is a function of the composition of the base material, weld filler materials, and the welding parameters that are used because these two factors control the dilution of the solidified weld deposit. This suggests that a combined sulfur plus silicon content of the base material of approximately 0.046% will represent a threshold for hot cracking with the weld parameters WSI will use at Duane Arnold. Duane Arnold will use a barrier layer (buffer layer) on all stainless steel. The barrier layer will use ER308L on the stainless steel and will incorporate Alloy 82 on the stainless steel near the DMW to stainless steel fusion zone only.

The barrier layer will not be used in the structural analysis. The inside diameter of the portion of the overlay over the barrier layer will be the outside diameter of the barrier layer that is applied

over the stainless steel material beneath the overlay. Since this barrier layer will not be considered as part of the structural overlay, a delta ferrite measurement is not required. See Attachment 2 for more information.

The Cr content of the 1st layer was verified by direct measurement of weld overlay deposits on ASTM A106 Grade B mockups. Welding was performed using a double up progression (starting at the bottom and welding upward to the top on each side) for 5G and 6G mockups and orbital progression for 2G mockups. The Cr content was measured at 90 degree intervals. All welding parameters were recorded and a 24% minimum Cr value, greater than that required in 1(e), Attachment 1 (20% minimum Cr for BWRs), was attained in all cases. The attainment of the minimum 24% Cr threshold level has been demonstrated many times on similar mockups using the same WSI welding parameters. The same heat of wire, or a wire heat with equal or greater chrome content than that used in qualification, will be used in situ for the first layer and the same welding parameters will be specified in the WPS as was used in the mockup for the first layer. It should be noted that the deposition of Alloy 52M over base material already having significant Cr content will result in a higher final deposited Cr content.

Examination

All examinations will meet the requirements of Attachment 1, excluding qualification of the ultrasonic examination for the completed overlay. The ultrasonic examination qualification will be in accordance with ASME Code Section XI, 2001 Edition (Reference 5), for Appendix VIII, Supplement 11 with the alternatives that are used in Relief Request NDE-R002.

The final ultrasonic examination report will be submitted to the NRC as part of the In-service Inspection (ISI) Report to be submitted after startup from the refuel outage, in accordance with IWA-6240. Any flaws detected that exceed the acceptance standards of Table IWB-3514-2 will be reported to the NRC as soon as possible. A discussion and reason for any overlay or base metal repairs will be provided.

The ultrasonic examination requirements specified in NRC Regulatory Guide 1.147, Revision 16, as conditional acceptance of Code Case N-504-4 will be applied to these overlays. In doing so, UT of the overlays will be performed in accordance with Section XI, Appendix VIII, Supplement 11 qualified procedures and personnel as modified by Relief Request NDE-R002. Supplement 11 was prepared to be specifically applicable to weld overlays. The ultrasonic examination requirements in Section 3, Attachment 1, are similar to the ultrasonic examination requirements provided in Appendix Q which have been developed specifically for austenitic weld overlays. The UT examination to be performed, in conjunction with the surface examinations to be performed, as specified in Section 3 Attachment 1 are based on the latest industry experience and practice and are completely satisfactory for the weld overlay application.

Conclusion

The proposed alternative shown in Attachment 1 has been developed to cover the most recent operating experience and NRC approved criteria that are associated with similar overlay applications. Similar NRC approved requests have been used to produce acceptable weld overlays when applied to dissimilar metal welds with Alloy 82/182 weld material. Therefore,

NextEra Energy considers the proposed alternative described in Attachment 1, with the inclusion of approved relief request NDE-R002, to provide an acceptable level of quality and safety, consistent with provision of 10CFR 50.55a(a)(3)(i).

Precedents

The proposed relief request is similar to that previously submitted by FPL Energy[1] titled "Alternative to ASME Section XI Repair Requirements to use Code Case N-504-2 and N-638-1 for Weld Overlay Repairs at the Duane Arnold Energy Center" which was approved by the Staff on June 12, 2007. (ML071110007)

In addition, the NextEra Energy Duane Arnold request is similar to other recent requests for dissimilar metal weld overlays, both Boiling Water Reactors (BWRs) (e.g., Pilgrim Nuclear Power Station – ML092370549) and Pressurized Water Reactors (PWRs) (e.g., Seabrook Station - ML081000008), with the noted exception that the NextEra Energy Duane Arnold request will not rely upon Code Case N-638-1. And while the enclosed request is not pre-emptive, many of the same requirements are common in the Staff's approval for the Materials Reliability Program (MRP) topical report (MRP-169), for full structural weld overlays (ML101660468).

Duration of Proposed Alternative

The alternative requirements of this request will be applied for the duration of up to and including the last refuel outage of the current 4th 10-year ISI interval, which includes inservice examination requirements of Attachment 1 for any applied weld overlays. Future inservice examination of weld overlay at Duane Arnold beyond this inspection interval will be as required by the NRC in the regulations.

References

1. ASME Code, Section XI, 2001 Edition, including Addenda through 2003
2. Duane Arnold Relief Request NDE-R002 approved January 31, 2007
3. ASME Code, Section XI 2001 Edition, for Appendix VIII, Supplement 11 examinations as modified by 10CFR50.55a(b)(2)(xxiv).

[1] By License Amendment No. 275, the licensee name for the Duane Arnold Energy Center was legally changed from FPL Energy Duane Arnold, LLC to NextEra Energy Duane Arnold, LLC.

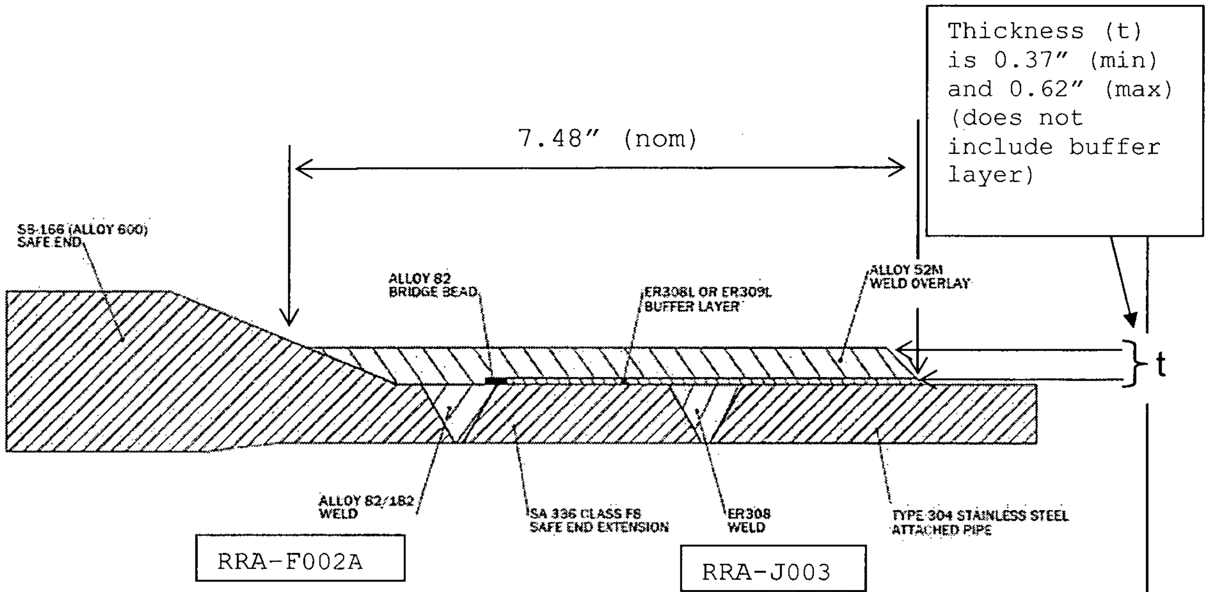


Figure 1.

Weld Overlay Repair for Safe End-to-Safe End Extension DMW (RRA-F002A) and Adjacent Stainless Steel Weld (RRA-J003).

Attachment 1
Alternative Requirements for Dissimilar Metal Weld Overlays

In lieu of the requirements of IWA-4410 and IWA-4611, a defect in austenitic stainless steel or austenitic nickel alloy piping, components, or associated welds may be reduced to a flaw of acceptable size in accordance with IWB-3640 by the addition of a repair weld overlay. The weld overlay shall be applied by deposition of weld reinforcement (weld overlay) on the outside surface of the piping, component, or associated weld, provided the following requirements are met:

1 GENERAL REQUIREMENTS

A. A full-structural weld overlay shall be applied by deposition of weld reinforcement (weld overlay) on the outside surface of circumferential welds F002A and J-3 in Alloy 600 safe ends (P-No. 43) to piping components (P-No. 8), inclusive of the Alloy 82 and E308 welds that join the items.

B. This attachment applies to dissimilar metal welds between P-No. 8 and P-No. 43 materials joined with austenitic F-No. 43 filler metal, and to welds between P-No. 8 and P-No. 8 materials as described in 1A above.

1) Weld overlay filler metal shall be austenitic nickel alloy (28% Cr min., ERNiCrFe-7A) applied 360° around the circumference of the item, and deposited using a Welding Procedure Specification (WPS) for groove welding, qualified in accordance with the Construction Code and Owner's Requirements and identified in the Repair/Replacement Plan.

C. Prior to deposition of the weld overlay, the surface to be weld overlaid shall be examined using the liquid penetrant method. Indications with major dimension greater than 1/16 in. (1.5 mm) shall be removed, reduced in size, or weld repaired in accordance with the following requirements:

(1) One or more layers of weld metal shall be applied to seal unacceptable indications in the area to be repaired with or without excavation. The thickness of these layers shall not be used in meeting weld reinforcement design thickness requirements. Peening the unacceptable indication prior to welding is permitted.
(2) If weld repair of indications identified in 1.C is required, the area where the weld overlay is to be deposited, including any local weld repairs or initial weld overlay layer, shall be examined by the liquid penetrant method. The area shall contain no indications with major dimension greater than 1/16 in. (1.5 mm) prior to the application of the structural layers of the weld overlay.

D. Weld overlay deposits shall meet the following requirements: The austenitic nickel alloy weld overlay shall consist of at least two weld layers deposited using a filler material with a Cr content of at least 28%. The first layer of weld metal deposited may not be credited toward the required thickness. Alternatively, for BWR applications, a diluted layer may be credited toward the required thickness, provided the portion of the layer over the austenitic base material, austenitic filler material weld contain at least 20% Cr and the Cr

content of the deposited weld metal is determined by chemical analysis taken from a mockup prepared in accordance with the WPS for the production weld.

E. A new weld overlay shall not be installed over the top of an existing weld overlay that has been in service.

2 CRACK GROWTH AND DESIGN

A. Crack Growth Calculation of Flaws in the Original Weld or Base Metal.

The size of all flaws postulated in the original weld or base metal shall be used to define the life of the overlay (defined as the end of the 40 year plant design life plus the 20 year license extension period). In no case shall the inspection interval be longer than the life of the overlay. The inspection interval shall be as specified in 3.C. Crack growth in the original weld or base metal, due to both stress corrosion and fatigue, shall be evaluated. Flaw characterization and evaluation shall be based on the postulated flaw, if ultrasonic examination of the weld and base material is not performed.

(1) For repair overlays, the initial flaw size for crack growth in the original weld or base metal shall be based on the postulated flaw, if no pre-overlay ultrasonic examination is performed.
(2) For postulated flaws in the original weld or base metal, the axial flaw length shall be set at 1.5 in. (38 mm) or the combined width of the weld plus buttering, whichever is greater. The circumferential flaw length shall be assumed to be 360°.
(3) Flaw growth evaluations shall include the residual stress results to demonstrate that favorable stress distribution in the original weld has been performed.

B. Structural Design and Sizing of the Overlay.

The design of the weld overlay shall satisfy the following, using the assumptions and flaw characterization restrictions in 2.A. The following design analysis shall be completed in accordance with IWA-4311.

(1) The axial length and end slope of the weld overlay shall cover the weld and heat-affected zones on each side of the weld and shall provide for load redistribution from the item into the weld overlay and back into the item without violating applicable stress limits of NB-3200. Any laminar flaws in the weld overlay shall be evaluated in the analysis to ensure that load redistribution complies with the above. These requirements will usually be satisfied if the weld overlay full-thickness length extends axially beyond the projected flaw by at least $0.75\sqrt{(Rt)}$, where R is the outer radius of the item and t is the nominal wall thickness of the item.
(2) Unless specifically analyzed in accordance with 2B(1), the end transition taper of the overlay shall not exceed 30°. A slope of not more than 1:3 is recommended.
(3) For determining the combined length of circumferentially-oriented flaws, in the underlying base material or weld, multiple flaws shall be treated as one flaw of length equal to the sum of the lengths of the individual flaws characterized in accordance with IWA-3300.

(4) For circumferentially oriented flaws, in the underlying base material or weld, the flaws shall be assumed to be 100% through the original wall thickness for the entire circumference of the item.

(5) For axial flaws in the underlying base material or weld, the flaws shall be assumed to be 100% through the original wall thickness of the item for the entire axial length of the flaw or combined flaws, as applicable.

(6) The overlay design thickness shall be verified using only the weld overlay thickness conforming to the deposit analysis requirements of 1D. The combined wall thickness at the weld overlay and the effects of any discontinuities (e.g., another weld overlay or reinforcement for a branch connection) within a distance of $0.75\sqrt{(Rt)}$ from the toes of the weld overlay, including the flaw size assumptions defined in 2B(4) or (5) above, shall be evaluated and meet the requirements of IWB-3640.

(7) The effects of any changes in applied loads, as a result of weld shrinkage from the entire overlay, on other items in the piping system (e.g., support loads and clearances, nozzle loads, and changes in system flexibility and weight due to the weld overlay) shall be evaluated. Existing flaws previously accepted by analytical evaluation shall be evaluated in accordance with IWB-3640, IWC-3640, or IWD-3640, as applicable. These evaluations shall meet the requirements of ASME Section III NB-3200 and NB-3600.

3 EXAMINATION

In lieu of all other examination requirements, the examination requirements herein shall be met for the life of the weld overlay. Nondestructive examination methods shall be in accordance with IWA-2200, except as specified herein. Nondestructive examination personnel shall be qualified in accordance with IWA-2300. Ultrasonic examination procedures and personnel shall be qualified in accordance with Appendix VIII, Supplement 11 and Relief Request NDE-R002.

A. Acceptance Examination

(1) The weld overlay shall have a surface finish of 250 micro-in. (6.3 micrometers) RMS or better and a contour that provides for ultrasonic examination in accordance with procedures qualified in accordance with Appendix VIII. The weld overlay shall be inspected to verify acceptable configuration.

(2) The weld overlay and the adjacent base material for at least ½ in. (13 mm) from each side of the weld shall be examined using the liquid penetrant method. Surface examination shall be performed on weld attached thermocouple removal areas in accordance with NB-4435(b)(3). The weld overlay shall satisfy the surface examination acceptance criteria for welds of the Construction Code or NB-5300. The adjacent base metal shall satisfy the surface examination acceptance criteria for base material of the Construction Code or NB-2500.

(3) The acceptance examination volume A-B-C-D in Fig. 1(a) plus the heat-affected zone beneath the fusion zone C-D shall be ultrasonically examined to assure adequate fusion (i.e., adequate bond) with the base metal and to detect welding flaws, such as interbead lack of fusion, inclusions, or cracks.

Planar flaws detected in the weld overlay acceptance examination shall meet the pre-service examination standards of Table IWB-3514-2. In applying the acceptance

standards to planar indications within the volume E-F-G-H, in Fig. 1(b), the thickness "t_1" shall be used as the nominal wall thickness in Table IWB-3514-2. For planar indications outside this examination volume, the nominal wall thickness shall be "t_2" as shown in Fig. 1(c), for volumes A-E-H-D and F-B-C-G.

Laminar flaws in the weld overlay shall meet the following:

(a) Laminar flaws shall meet the acceptance standards of Table IWB-3514-3 with the additional limitation that the total laminar flaw shall not exceed 10% of the weld surface area and that no linear dimension of the laminar flaw area exceeds 3.0 in. (76 mm) or 10% of the nominal pipe circumference, whichever is greater.
(b) The reduction in coverage of the examination volume A-B-C-D in Fig. 1(a), due to laminar flaws shall be less than 10%. The uninspectable volume is the volume in the weld overlay underneath the laminar flaws for which coverage cannot be achieved with angle beam examination.
(c) Any uninspectable volume in the weld overlay shall be assumed to contain the largest radial planar flaw that could exist within that volume. This assumed flaw shall meet the preservice examination acceptance standards of Table IWB-3514-2, with nominal wall thickness as defined above for planar flaws. Both axial and circumferential planar flaws shall be assumed.

(4) After completion of all welding activities, affected restraints, supports and snubbers shall be VT-3 visually examined to verify that design tolerances are met.

B. Pre-service Inspection

(1) The examination volume in Fig. 2 shall be ultrasonically examined. The angle beam shall be directed perpendicular and parallel to the piping axis, with scanning performed in four directions, to locate and size any planar flaws that might have propagated into the upper 25% of the base material or into the weld overlay.
(2) The preservice examination acceptance standards of Table IWB-3514-2 shall be met for the weld overlay. In applying the acceptance standards, wall thickness, t_w, shall be the thickness of the weld overlay.
(3) The flaw evaluation requirements rules of IWB-3640 shall not be applied to planar flaws identified during preservice examination that exceed the preservice examination acceptance standards of Table IWB-3514-2.

C. Inservice Inspection

(1) The weld overlay examination volume in Fig. 2 shall be added to the inspection plan. The weld overlay inspection interval shall not be greater than the life of the overlay defined in 2A above. The weld overlay shall be ultrasonically examined during the first or second refueling outage following application.
(2) The weld overlay examination volume in Fig. 2 shall be ultrasonically examined to determine if any new or existing planar flaws have propagated into the outer 25% of the base metal thickness or into the overlay. The angle beam shall be directed perpendicular and parallel to the piping axis, with scanning performed in four directions.

(3) The inservice examination acceptance standards of Table IWB-3514-2 shall be met for the weld overlay. If flaw growth in the weld overlay occurs and inservice examination acceptance standards of Table IWB-3514-2 cannot be met, a determination will be made to prove that the flaw is not SCC. If the cause is determined to be SCC or the cause of the flaw cannot be determined, the flaw shall be repaired and IWB-3600, IWC-3600, or IWD-3600 shall not be used to accept these types of flaws. Flaws due to stress corrosion cracking in the weld overlay that exceed the inservice examination acceptance standards of Table IWB-3514-2 shall not be accepted and will result in removal of the weld overlay and the item shall be repaired or replaced.

(4) Weld overlay examination volumes in Fig. 2 that show no indication of planar flaw growth or new planar flaws shall be placed into a population to be examined on a sample basis, except as required by 3C(1). Twenty-five percent of this population shall be examined at least once during every 10 years.

(5) If inservice examinations reveal planar flaw growth, or new planar flaws that meet the inservice examination acceptance standards of IWB-3514 or acceptance criteria of IWB-3600, the weld overlay examination volume shall be reexamined during the first or second refueling outage following discovery of the growth or new planar flaws.

(6) For weld overlay examination volumes with unacceptable indications in accordance with 3C(3), the weld overlay shall be removed, including the original defective weld, and the item shall be corrected by a repair/replacement activity in accordance with IWA-4000.

D. Additional Examinations.

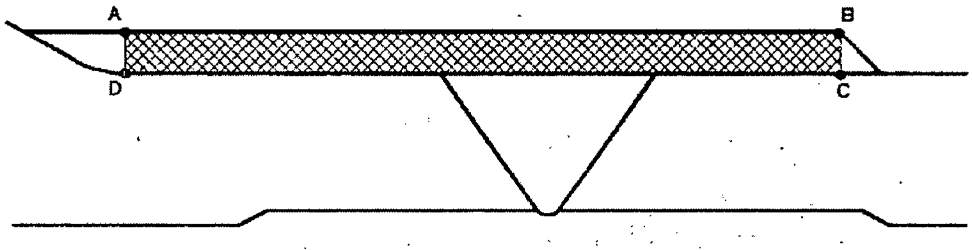
If inservice examinations reveal unacceptable indications according to 3C(3), planar flaw growth into the weld overlay design thickness, or axial flaw growth beyond the specified examination volume, additional weld overlay examination volumes, equal to the number scheduled for the current inspection period, shall be examined prior to return to service. If additional unacceptable indications are found in the second sample, 50% of the total population of weld overlay examination volumes shall be examined prior to return to service. If additional unacceptable indications are found, the entire remaining population of weld overlay examination volumes shall be examined prior to return to service.

4 PRESSURE TESTING

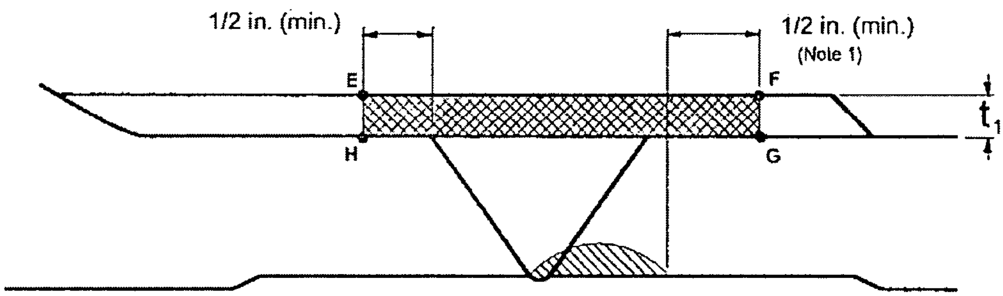
A system leakage test shall be performed in accordance with IWA-5000.

5 DOCUMENTATION

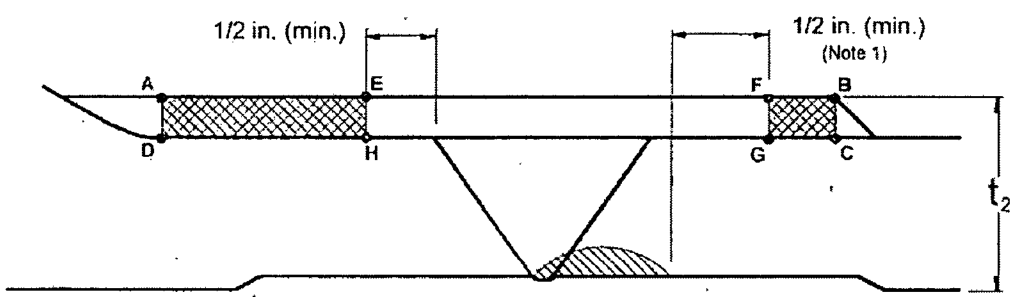
Use of Attachment 1 shall be documented on Form NIS-2a (Ref. Code Case N-532-4).



a. Acceptance Examination Volume A-B-C-D



b. Thickness (t_1) for Table IWB-3514-2



c. Thickness (t_2) for Table IWB-3514-2

Fig. 1 Acceptance Examination Volume and Thickness Definitions

Notes:

(1) For axial or circumferential flaws, the axial extent of the examination volume shall extend at least ½ in. (13 mm) beyond the toes of the original weld.

(2) The weld includes the weld end butter, where applied.

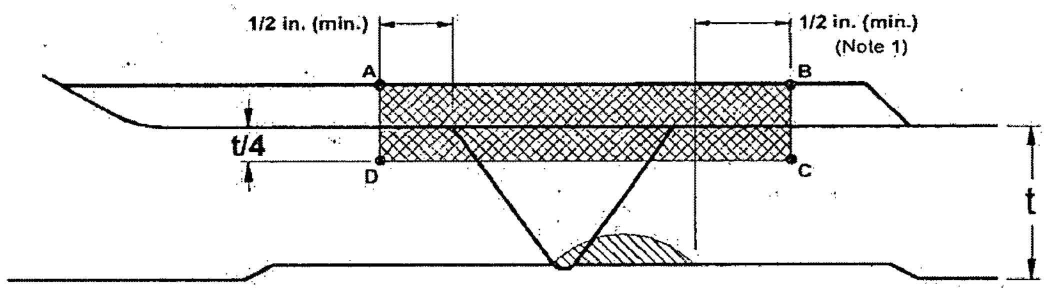


Fig. 2

Preservice and Inservice Examination Volume

Notes:

(1) For axial or circumferential flaws, the axial extent of the examination volume shall extend at least ½ in. (13 mm) beyond the as-found flaw and at least ½ in. (13 mm) beyond the toes of the original weld.

(2) The weld includes the weld end butter, where applied.

ATTACHMENT 2

Barrier (Buffer) Layer to Prevent Hot Cracking in High Sulfur Stainless Steel

Background

During recent dissimilar metal weld (DMW) overlay activities, where use of ERNiCrFe-7A (Alloy 52M) and ERNiCrFe-7 (Alloy 52) has been used for the filler metal, flaws in the first layer have occurred in the portion of the overlay deposited on the austenitic stainless steel portions (safe ends, pipe, etc.) of the assemblies in some cases.

Discussion

The flaw characteristics observed above are indicative of hot cracking. This phenomenon has not been observed on the stainless steel or ENiCrFe-3 (Alloy 182) DMW portions of the assemblies when welding Alloy 52M thereon.

Further studies have determined that this problem may occur when using Alloy 52M filler metal on austenitic stainless steel materials with high sulfur content.

Extensive tests and field experience from WSI indicate that hot cracking can be a concern when the sulfur and silicon content in the diluted weld puddle equals or exceeds 0.014%. The impurity hot cracking threshold level is a function of both the composition of the weld filler materials and the welding parameters that are used because these two factors control the dilution of the solidified weld deposit. This suggests that a combined sulfur plus silicon content of the base material approximately 0.046% will represent a threshold for hot cracking with the weld parameters WSI will use at Duane Arnold.

To reduce the susceptibility of hot cracking occurrence due to welding Alloy 52M on the stainless steel base materials with high sulfur, WSI has selected ER308L filler metal as the preferred filler metals to provide a barrier (buffer) layer between the Alloy 52M and the high sulfur stainless steel base material. These filler metals are compatible with the base material and promote primary weld metal solidification as ferrite rather than austenite. The ferrite is more accommodating of residual elements therein and in the underlying base material thereby significantly reducing the susceptibility to hot cracking. These filler metals are also compatible with the Alloy 52M subsequently welded thereon. However, the barrier layer may consist of ERNiCr-3 (Alloy 82) being used locally at the interface between the Alloy 182 DMW and the stainless steel item. ER308L welding on Alloy 182 may result in cracking of the ER308L weld. Welding on high sulfur stainless steel with Alloy 82 has not been a concern relevant to hot cracking occurrence.

WSI welded two mockups to evaluate the interactive effects, such as hot cracking and lack of fusion, between the Alloy 82/182 DMW, the stainless steel base material, the ER308L and the subsequent Alloy 52M weld overlay. One mockup assembly consisted of a stainless steel pipe (approximately 0.050 wt% sulfur and silicon combined) and an ASTM A106 Grade B pipe with an Alloy 82 groove weld joining them. The second mockup consisted of a stainless steel pipe joined to a cast stainless steel pipe with a stainless butt weld. The other end of the cast stainless steel pipe was joined to a P3 forging using an Alloy 82 groove weld.

For both mockups, the barrier layer and overlay were welded in the same sequence as performed in the field (barrier layer ER308L to within 1/8" of the joining DMW and then four or more layers of Alloy 52M overlay). The barrier layer and overlay welding parameters used in the mockup were similar to those used in the field and controlled the weld dilution by controlling the weld heat input and the Power Ratio.

The following examinations were performed on the final mockup:

- PT was performed on the base materials and joining groove welds
- PT was performed on the ER308L barrier layer
- PT was performed on the first layer of Alloy 52M overlay and on the final layer of the weld overlay
- PDI qualified Phased Array UT was performed on the completed mockup.

One recordable (not rejectable) planar UT indication of less than 0.200" in length was found on one of the mockups unrelated to the barrier layer. Subsequent metallographic examination found this likely to be porosity. Metallographic examination was conducted at EPRI searching for any type of discontinuity, flaw or other anomaly. All samples were removed from selected locations in both mockups and revealed no conditions of concern.

Conclusion

Duane Arnold will use the barrier layer on all the stainless steel items that equal or exceed the criteria prior to overlay. The barrier layer will use ER308L on the stainless steel and may use Alloy 82 on the stainless steel near the DMW to stainless steel fusion zone only.

Structural credit will not be assumed for the barrier layer in determining the required minimum overlay thickness since the alternative does not address the use of stainless steel filler metal.

The barrier layer welding will be performed in accordance with ASME Section IX qualified welding procedure specification(s). PT will be performed on the barrier layer surface and its volume will be included in the final UT of the overlay.

15) Relief Request NDE-R015

1. ASME Code Component(s) Affected

Code Class:	1
References:	ASME Code Section XI, 2001 Edition/ including Addenda through 2003 ASME Code Section XI, 1998 Edition 10 CFR 50.55a(a)(b)(2)(xxvi)
Examination Category:	Not Applicable
Item Number:	Not Applicable
Description:	System pressure test and accompanying VT-2 examination at nominal operating pressure following repair/replacement activities involving mechanical joints
Component:	Main Steam Safety Relief Valve (SRV) PSV 4402

2. Applicable Code Edition and Addenda

ASME Code Section XI, 2001 Edition, including Addenda through 2003, is applicable to the Duane Arnold Energy Center (DAEC) Inservice Inspection Program and Repair/Replacement Program for the Fourth Ten-Year Interval.

ASME Code Section XI, 1998 Edition, is applicable for the pressure testing of mechanical joints included in Class 1, 2, and 3 repair and replacement activities, per 10 CFR 50.55a(a)(b)(2)(xxvi).

3. Applicable Code Requirement

IWA-4540(c) from the 1998 ASME Code Section XI:
Mechanical joints made in the installation of pressure retaining replacements shall be pressure tested in accordance with IWA-5211(a).

IWA-5211(a) from the 1998 ASME Code Section XI:
The pressure retaining components within each system boundary shall be subject to the following applicable system pressure tests under which conditions visual examination VT-2 is performed in accordance with IWA-5240 to detect leakages.

(a) a system leakage test conducted during operation at nominal operating pressure, or when pressurized to nominal operating pressure and temperature.

4. Reason for Request

The start-up sequence at the DAEC has been aborted to replace pilot assembly or complete

pilot assembly and valve body of a Safety/Relief Valve (SRV) (PSV 4402). The SRV is connected to the main steam piping with a bolted, mechanical joint. Replacing the SRV for maintenance is considered a Repair-Replacement activity under the rules of ASME Code Section XI, 2001 Edition, including Addenda through 2003 (the current code of record for DAEC Repair/Replacement Program). Following the replacement, a system leakage test and VT-2 examination are required. The system leakage test is required to be performed at the nominal pressure associated with the reactor at 100% power (approximately 1025 psig).

Several conditions associated with such testing represent an imposition on personnel safety, personnel radiation exposure, and challenges to the normal mode and manner of equipment operation. The SRV is not isolable from the reactor vessel; in order to perform this test, the primary system would need to be pressurized to the inboard system isolation valves. A leakage test and inspection at about 1025 psig cannot be performed during a normal plant startup, due to the excessive temperature and radiological exposure conditions to which licensee personnel would be exposed in the primary containment during the required VT-2 inspection. Extensive valve manipulations, alternative system lineups and procedural controls would be required for heating and pressurizing the primary system to establish the necessary test pressure, while complying with the Technical Specification (TS) requirements for Pressure-Temperature (P/T) Limits, without withdrawal of control rods, i.e., without using nuclear heat.

5. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(3)(ii), NextEra Energy Duane Arnold proposes to perform the system leakage test and VT-2 examination of the mechanical joints on SRV PSV 4402 during the normal operational start-up sequence at a minimum pressure of approximately 940 psig, in lieu of the nominal operating pressure associated with 100% reactor power (approximately 1025 psig). The VT-2 examination will be performed following the hold time required by the ASME Code. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in October 2012), another inspection of the bolted connection will be performed to look for any evidence of leakage at a minimum pressure of approximately 940 psig. This alternative is proposed on a "one-time-only" basis following the repair/replacement of the SRV planned for December 2010.

Basis for Use

NextEra Energy Duane Arnold considered the available methods to reach nominal operating pressure required to perform the system leakage test and VT-2 examination. These methods are discussed below.

Pressurizing System Without Withdrawing Control Rods

NextEra Energy Duane Arnold cannot isolate PSV 4402 from the reactor vessel. Thus, NextEra Energy Duane Arnold would have to manipulate numerous valves, change system lineups, and establish procedural controls for heating and pressurizing the primary system in order to perform the system leakage test and VT-2 examination of the mechanical joints of the SRV without withdrawing control rods, while maintaining compliance with the TS P/T Limits. The

reactor pressure vessel (RPV) would need to be filled with coolant and the steam lines flooded to the inboard main steam isolation valves (MSIVs) to provide a water-solid condition. The pressure increase would be obtained by balancing the flow into the vessel, which is provided by the control rod drive (CRD) system, with the flow out of the vessel provided by the reactor water cleanup (RWCU) system via the dump flow control valve and flow controller. This is the method used during refueling outages to complete the RPV system leakage test.

This test typically takes about two days to accomplish, and the additional valve lineups and system reconfigurations necessary to support this test impose an additional challenge to the affected systems. After completion of the test, system lineups must be restored to support start-up.

Pressurizing System During Normal Start-Up

Using normal startup procedures, the allowed pressure range for conducting the test would typically not be reached until a high power level (greater than 75% of rated). If access to the primary containment were permitted at this power level, personnel would be exposed to excessive radiation levels, including significant exposure to neutron radiation fields, which is contrary to current station ALARA practices. Establishing the 1025 psig test condition at a more moderate power level and in the manner needed to address radiation concerns would require a deviation from the method in which the primary system pressure control system (Electro-Hydraulic Control (EHC) Pressure Set) is normally used, as discussed below.

During a typical plant startup, after achieving criticality, the operating procedure directs the Operator to heat up and pressurize the reactor vessel (while maintaining the heat up rate within TS limits) by withdrawing additional control rods or raising EHC Pressure Set to maintain a turbine bypass valve within a specified "percent open" range. Adjustments to EHC Pressure Set are stopped, by procedure, when reactor pressure reaches 940 psig. The reactor power at that point is typically between 5 and 10% of rated.

While it is technically possible to manipulate these controls to establish the nominal system pressure of 1025 psig at lower power levels, doing so will affect core reactivity and could challenge plant safety systems, such as the reactor protection system (RPS). Changing the EHC settings outside of the normal range of operation for the purpose of performing this test at nominal operating pressure would pose an operational challenge, since this would be outside the normal operating parameters for startup. Procedural revisions would be required, as well as training provided to the Operators, to enable the EHC controls to be manipulated in a manner outside the norm.

Conclusion

Compliance with the Code-required system leakage test and inspection would result in a hardship or unusual difficulty without a compensating increase in the level of quality and safety. Application of this alternative test maintains reasonable levels of personnel safety and reduces the opportunity for the introduction of undesirable operational challenges. While NextEra Energy Duane Arnold does not expect that leakage will occur, any leakage at the bolted connection would be related to the differential pressure across the connection. The reduction in test pressure is less than 10%, and is not, therefore, expected to affect the ability of the VT-2

examination to detect leakage from the bolted connection. In the event that leakage would occur at the mechanical joint at the slightly higher pressure associated with 100% operating power, it would be detected by the drywell monitoring systems, which include drywell pressure monitoring, the containment atmosphere monitoring system, and the drywell floor drain sumps. Leakage monitoring is required by the DAEC Technical Specifications. In addition, if there is an unplanned shutdown with a drywell entry before the next refueling outage (currently scheduled to begin in October 2012), another inspection of the bolted connection will be performed to look for any evidence of leakage at a minimum pressure of approximately 940 psig.

The alternative will provide an acceptable verification of the integrity of the mechanical joint without unnecessary radiation exposure and operational challenges.

6. Duration of Proposed Alternative

NextEra Energy Duane Arnold requests NRC authorization of the aforementioned alternative on a "one-time-only" basis following the replacement of PSV 4402.

7. Precedent

The NRC authorized use of a similar alternative on a "one-time-only" basis at the DAEC in response to the Letter to the USNRC, “Request for Authorization of Alternative Regarding Pressure Test Requirements” dated March 12, 2004 (NG-04-0176) (ML040850127)

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 1	2	3	Comments
B-A	B1.11	Circumferential Shell Welds	Reactor Vessel	Volumetric			0	NDE-R003
	B1.12	Longitudinal Shell Welds	Reactor Vessel	Volumetric			8	
	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	NDE-R008
	B1.40	Head-to-Flange Weld	Reactor Vessel	Volumetric & Surface	1/3	1/3	1/3	NDE-R008 (Partial Exams during 1st & 2nd Periods)
B-D	B3.90	Nozzle-to-Vessel Welds in Reactor Vessel	Reactor Vessel	Volumetric	10	4	4	(1) Exempt by IWB 1220(c) NDE-R013
	B3.100	Nozzle Inside Radius Section in Reactor Vessel	Reactor Vessel	Volumetric	10	4	4	(1) Exempt by IWB 1220(c) NDE-R013
B-F	B5.10	Dissimilar Metal Nozzle-to-safe End Butt Welds NPS 4 or Larger	Various Class 1	Volumetric & Surface	N/A(1)			NDE-R005
	B5.20	Dissimilar Metal Nozzle-to-safe End Butt Welds NPS 4 or Smaller	Various Class 1	Surface				NDE-R005
	B5.30	Reactor Vessel Nozzle-to-Safe End Socket Welds	Various Class 1	Surface				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	
	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	
	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
	B6.200	Nuts, Bushings, & Washers in Pumps	All Class 1	Visual, VT-1				Inspected only when disassembled
B-G-2	B7.10	Bolts, Studs, & Nuts in Reactor Vessel	Various Class 1	Visual, VT-1	1	1	1	
	B7.50	Bolts, Studs, & Nuts in Piping	Various Class 1	Visual, VT-1	1			Inspected only when disassembled
	B7.60	Bolts, Studs, & Nuts in Pumps	Various Class 1	Visual, VT-1				Inspected only when disassembled
	B7.70	Bolts, Studs, & Nuts in Valves	Various Class 1	Visual, VT-1	1			Inspected only when disassembled

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
B-G-2 10 CFR 50.55a Requirement	B7.80	Bolts, Studs, & Nuts in CRD Housings	Reactor Vessel	Visual, VT-1	0			Inspected only when disassembled and bolting reused
B-J	B9.11	Circumferential Welds in Piping NPS 4 or Larger	Various Class 1	Volumetric & Surface	N/A(1)			NDE-R005
	B9.21	Circumferential Welds in Piping Less Than NPS 4	Various Class 1	Surface				NDE-R005
	B9.31	Branch Pipe Connection Welds NPS 4 or Larger	Various Class 1	Volumetric & Surface				NDE-R005
	B9.32	Branch Pipe Connection Welds Less Than NPS 4	Various Class 1	Surface				NDE-R005
	B9.40	Socket Welds	Various Class 1	Surface				NDE-R005
B-K	B10.10	Integrally Welded Attachments to Reactor Vessel	Various Class 1	Surface			1	
	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

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
B-M-2	B12.50	Valve Bodies, Exceeding NPS 4	Various Class 1	Visual, VT-3	3			Selected valves Inspected only when disassembled
B-N-1	B13.10	Vessel Interior	Various Class 1	Visual, VT-3		(2)		Accessible spaces each period
B-N-2	B13.20	Interior Attachments within Beltline Region in Reactor Vessel	Reactor Vessel	Visual, VT-1			11	
	B13.30	Interior Attachments beyond Beltline Region in Reactor Vessel	Reactor Vessel	Visual, VT-3	8	2	17	Some attachments are normally not accessible (bottom head area)
	B13.40	Core Support Structure in Reactor Vessel	Reactor Vessel	Visual, VT-3		(3)		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	
C-A	C1.10	Circumferential Shell Welds	RHR	Volumetric	1		1	
	C1.20	Circumferential Head Welds	RHR	Volumetric		1		

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
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	
	C3.20	Integrally Welded Attachments to Piping	Various Class 2	Surface	3	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	N/A[1]			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				NDE-R005
C-H	C7.10	System Leakage Test of Pressure Retaining Components	Various Class 2	Visual, VT-2	7	9	9	
D-A	D1.20	Integral Attachments - Piping	Various Class 3	Visual, VT-1	2	2	3	
D-B	D2.10	System Leakage Test	Various Class 3	Visual, VT-2	6	12	12	NDE-R007

Examination Category	Item Number	Examination Description	System Identification	Exam Requirements	Period Scheduled 1	2	3	Comments
F-A	F1.10	Class I Component Supports	Various Class 1	Visual, VT-3	15	12	16	
	F1.20	Class II Component Supports	Various Class 2	Visual, VT-3	18	15	22	
	F1.30	Class III Component Supports	Various Class 3	Visual, VT 3	6	8	12	
	F1.40	Supports Other Than Piping Supports (Class 1, 2, and 3)		Visual, VT-3	7	9	7	
R-A(1)	R1.10	No Degradation Mode	Various Class 1 & Class 2	Volumetric	2	8	7	NDE-R005
	R1.11	Thermal Fatigue	Various Class 1 & Class 2	Volumetric	3	2	10	NDE-R005
	R1.14	Corrosion Cracking	Various Class 1 & Class 2	Volumetric	0	0	0	NDE-R005
	R1.16	Intergranular Stress Corrosion Cracking	Various Class 1 & Class 2	Volumetric	14	17	5	NDE-R005
	R1.18	Flow-Accelerated Corrosion	Various Class 1 & Class 2	Volumetric	0	2	2	NDE-R005

Notes:
(1) Examination Categories B-F, B-J, and C-F-2 are evaluated as part of Risk Informed ISI based on EPRI methodology and included in Examination Category R-A.
(2) Examination in the spaces above and below the reactor core for loose or missing parts and debris.
(3) Accessible components based on outage availability.