January 31, 2007

Mr. Gary Van Middlesworth
Vice President
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9785

SUBJECT: SAFETY EVALUATION OF RELIEF REQUESTS FOR THE FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM - DUANE ARNOLD ENERGY CENTER (TAC NOS. MD2517, MD2518, MD2519, MD2521, MD2522, MD2524)

Dear Mr. Van Middlesworth:

By letter dated June 30, 2006, as supplemented by two letters dated December 21, 2006, FPL Energy Duane Arnold, LLC (FPL Energy or the licensee) submitted requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI requirements for the Duane Arnold Energy Center (DAEC). The subject relief requests are for the fourth 10-year inservice inspection (ISI) interval at DAEC, which began on November 1, 2006.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of relief requests NDE-R001, NDE-R002, NDE-R005, NDE-R006, and NDE-R008 as documented in the enclosed Safety Evaluation (SE). Our SE concludes the following:

1) With respect to relief requests NDE-R001, NDE-R002, NDE-R005, and NDE-R008, the proposed alternatives will provide an acceptable level of quality and safety. Therefore, pursuant to Section 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR), the proposed alternatives are authorized for the remainder of the fourth 10-year ISI interval at DAEC.

2) With respect to relief request NDE-R006, the proposed use of Code Case N-686 will provide reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC, on the basis that complying with the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Use of Code Case N-686 is authorized until such time as the Code Case is published in a future version of Regulatory Guide (RG) 1.147 and incorporated by reference in 10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this Code Case, it must follow all provisions of Code Case N-686 with conditions as specified in RG 1.147 and limitations as specified in 10 CFR 50.55a(b)(5), if any.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in these relief requests remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

It should be noted that in authorizing Relief Request NDE-R001, Technical Requirements Manual Section T 3.7.2 and its Bases become regulatory requirements that may be used in lieu of ASME Code, Section XI requirements for performing ISI and testing of snubbers. Changes to these requirements must be reviewed and approved by the NRC staff for authorization pursuant to 10 CFR 50.55a(a)(3) or as an exemption pursuant to 10 CFR 50.12.

Relief requests NDE-R004 and NDE-R007 are still being evaluated by the NRC staff and the results of the review will be documented by separate letter. In addition, as noted in your letter dated June 30, 2006, relief request NDE-R003 was approved as relief request NDE-R047 by the NRC staff in an SE dated January 6, 2005. Your letter stated that the contents of relief request NDE-R003 was editorially updated, was included for completeness, and that FPL Energy was not requesting NRC approval of this request. Accordingly, the NRC did not review relief request NDE-R003.

If you have any questions regarding this matter, please contact Mr. Karl Feintuch at (301) 415-3079.

Sincerely,

/RA/

Patrick D. Milano, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure:
Safety Evaluation

cc w/encl: See next page

It should be noted that in authorizing Relief Request NDE-R001, Technical Requirements Manual Section T 3.7.2 and its Bases become regulatory requirements that may be used in lieu of ASME Code, Section XI requirements for performing ISI and testing of snubbers. Changes to these requirements must be reviewed and approved by the NRC staff for authorization pursuant to 10 CFR 50.55a(a)(3) or as an exemption pursuant to 10 CFR 50.12.

Relief requests NDE-R004 and NDE-R007 are still being evaluated by the NRC staff and the results of the review will be documented by separate letter. In addition, as noted in your letter dated June 30, 2006, relief request NDE-R003 was approved as relief request NDE-R047 by the NRC staff in an SE dated January 6, 2005. Your letter stated that the contents of relief request NDE-R003 was editorially updated, was included for completeness, and that FPL Energy was not requesting NRC approval of this request. Accordingly, the NRC did not review relief request NDE-R003.

If you have any questions regarding this matter, please contact Mr. Karl Feintuch at (301) 415-3079.

Sincerely,
/RA/
Patrick D. Milano, Acting Chief
Plant Licensing Branch III-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-331

Enclosure:
Safety Evaluation

cc w/encl: See next page

DISTRIBUTION:

PUBLIC	RidsAcrsAcnwMailCenter	RidsNrrDorlLpl3-1
LPL3-1 Reading	RidsRgn3MailCenter	TBloomer, EDO Region 3
RidsNrrPMKFeintuch	RidsNrrPMREnnis	JMcHale
RidsNrrLATHarris	RidsOgcRp	GBedi
TChan	LMrowca	
AKeim	HPhan	

ADAMS Accession No.: ML070090357

OFFICE	LPL3-1/PM	LPL3-1/LA	CPNB/BC	APLB/BC	CPTB/BC	OGC	LPL3-1/BC(A)
NAME	REnnis	THarris	WKoo for TChan	MCaruso for LMrowca	JMcHale	PMoulding	PMilano
DATE	1/31/07	1/11/07	1/17/07	1/16/07	1/12/07	1/30/07	1/31/07

OFFICIAL RECORD COPY

Duane Arnold Energy Center

cc:

Mr. J. A. Stall
Senior Vice President, Nuclear and Chief Nuclear Officer
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

Mr. M. S. Ross
Managing Attorney
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

Mr. R. E. Helfrich
Senior Attorney
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

Mr. W. E. Webster
Vice President, Nuclear Operations
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

John Bjorseth
Site Director
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324

Steven R. Catron
Manager, Regulatory Affairs
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324

U. S. Nuclear Regulatory Commission
Resident Inspector's Office
Rural Route #1
Palo, IA 52324

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
Suite 210
2443 Warrenville Road
Lisle, IL 60532-4351

Mr. M. Warner
Vice President, Nuclear Operations Support
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

Mr. D. A. Curtland
Plant Manager
Duane Arnold Energy Center
3277 DAEC Rd.
Palo, IA 52324-9785

Mr. R. S. Kundalkar
Vice President, Nuclear Engineering
Florida Power & Light Company
P. O. Box 14000
Juno Beach, FL 33408-0420

Daniel K. McGhee
Iowa Department of Public Health
Bureau of Radiological Health
321 East 12th Street
Lucas State Office Building, 5th Floor
Des Moines, IA 50319-0075

Chairman, Linn County
Board of Supervisors
930 1st Street SW
Cedar Rapids, IA 52404

October 12, 2006

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO RELIEF REQUESTS FOR THE

FOURTH 10-YEAR INTERVAL OF THE INSERVICE INSPECTION PROGRAM

FPL ENERGY DUANE ARNOLD, LLC

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

1.0 INTRODUCTION

By letter dated June 30, 2006 (Reference 1), as supplemented by two letters dated December 21, 2006 (References 2 and 16), FPL Energy Duane Arnold, LLC (FPL Energy or the licensee) submitted requests for relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Section XI requirements for the Duane Arnold Energy Center (DAEC). The subject relief requests are for the fourth 10-year inservice inspection (ISI) interval at DAEC, which began on November 1, 2006.

2.0 REGULATORY EVALUATION

The ISI of the ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) Section 50.55a(g), except where specific written relief has been granted by the U.S. Nuclear Regulatory Commission (NRC or Commission) pursuant to 10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the NRC, if the licensee demonstrates that: (i) the proposed alternatives would provide an acceptable level of quality and safety, or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components. The regulation requires that inservice examination of components and system pressure tests conducted during the first 10-year interval, and subsequent intervals, comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The applicable ASME Code of record for the fourth 10-year ISI interval at DAEC is the ASME Code, Section XI, 2001 Edition with Addenda through 2003.

3.0 TECHNICAL EVALUATION

3.1 Relief Request NDE-R001

3.1.1 Components for Which Relief is Requested

The request is applicable to all DAEC safety-related ASME Code Class 1, 2 and 3 snubbers.

3.1.2 Code Requirements

The ASME Code, Section XI, Article IWF-5000, provides ISI requirements for snubbers.

Paragraphs IWF-5200(a) and IWF-5300(a) require that snubber preservice and inservice examinations be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213.

Paragraphs IWF-5200(b) and IWF-5300(b) require that snubber preservice and inservice tests be performed in accordance with OM-4.

Paragraphs IWF-5200(c) and IWF-5300(c) require that integral and nonintegral attachments for snubbers, including lugs, bolting, pins, and clamps, be examined in accordance with the requirements of Subsection IWF.

3.1.3 Licensee's Proposed Alternative

The licensee proposes to continue to use the requirements in DAEC Technical Requirements Manual (TRM) Section T 3.7.2, "Snubbers," for visual examination and 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.

3.1.4 Licensee's Basis for the Alternative

In Reference 1, the licensee provided the basis for the proposed alternative as follows:

> Pursuant to 10 CFR 50.55a(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 T 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 10 CFR 50.55a 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 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 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 [in Reference 1] allows a 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.

3.1.5 NRC Staff Evaluation

The licensee requested authorization of an alternative to the requirements of the ASME Code, Section XI, paragraphs IWF-5200(a), (b) and (c), and IWF-5300(a), (b), and (c). The licensee proposed that the visual examinations and functional testing of ASME Code Class 1, 2 and 3 snubbers be performed in accordance with the requirements of DAEC TRM Section T 3.7.2 and its procedures in lieu of meeting the requirements in the ASME Code, Section XI, paragraphs IWF-5200(a), (b) and (c), and IWF-5300(a), (b) and (c).

The applicable edition of Section XI of the ASME Code for the DAEC fourth 10-year ISI interval is the 2001 Edition through 2003 Addenda. The ASME Code, Section XI, paragraphs IWF-5200(a), (b) and (c), and IWF-5300(a), (b), and (c) reference OM-4, 1987 Edition with OMa-1988 Addenda.

ASME Code, Section XI, paragraphs IWF-5200(a) and IWF-5300(a) require that snubber preservice and inservice examinations be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213. Paragraphs IWF-5200(b) and IWF-5300(b) require that snubber preservice and inservice tests be performed in accordance with OM-4.

Paragraphs IWF-5200(c) and IWF-5300(c) require that integral and nonintegral attachments for snubbers, including lugs, bolting, pins, and clamps, shall be examined in accordance with the requirements of Subsection IWF. The licensee states that the relief 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 ISI Program.

OM-4 specifies the requirements for visual examination (paragraph 2.3), and functional testing (paragraph 3.2) of snubbers. The licensee proposes to use DAEC TRM Section T 3.7.2 surveillance requirements for visual inspection and functional testing of all safety-related snubbers. A visual inspection is the observation of the condition of installed snubbers to identify those that are damaged, degraded, or inoperable as caused by physical means, leakage, corrosion, or environmental exposure. To verify that a snubber can operate within specific performance limits, the licensee performs functional testing that typically involves removing the snubber and testing it on a specially designed stand or bench. The performance of visual examinations is a separate process that complements the functional testing program and provides additional confidence in snubber operability.

DAEC TRM Section T 3.7.2 incorporates Generic Letter (GL) 90-09, “Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions.” GL 90-09 acknowledges that the visual inspection schedule (as contained in OM-4) is excessively restrictive and that licensees with large snubber populations have spent a significant amount of resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements. GL 90-09 states that its alternative schedule for visual inspection provides the same confidence level as that provided by OM-4.

DAEC TRM Section T 3.7.2 defines inservice examination requirements, method of examination, subsequent examination intervals, failure evaluation, inservice operability test requirements, initial snubber sample size, additional sampling, failure evaluation, test failure mode groups, and corrective actions for the 10 percent sample plan that are similar to those provided by OM-4. OM-4 requirements and TRM Section T 3.7.2 requirements are compared and summarized in the following table and followed by a detailed review:

	Criteria	ASME/ANSI OM Part 4 -1988	DAEC, TRM Section T 3.7.2 Requirements and Bases
Inservice Examination			
1.	Visual Examination	Paragraph 2.3.1.1, Visual Examination, states that snubber visual examinations shall identify impaired functional ability due to physical damage, leakage, corrosion, or degradation.	TRM Technical Surveillance Requirement (TSR) 3.7.2.2 requires that visual inspections shall verify that there are: (a) no indications of damage or impaired operability; (b) attachments to the foundation or supporting structure are secure; and (c) fasteners for the attachment of the snubber to component and snubber anchorage are secure.
2.	Visual Examination Interval Frequency	Paragraph 2.3.2.2 provides examination interval frequency and additional examination requirements.	TRM Table T3.7.2-1 provides snubber visual inspection interval frequency.
3.	Method of Visual Examination	IWF-5200(a) and IWF-5300(a) require use of the VT-3 visual examination method described in IWA-2213.	DAEC states that IWA-2213 will be used to identify the examination technique utilized and IWA-2300 will be used to qualify/certify the VT-3 examiners.
4.	Subsequent Examination Intervals	Paragraph 2.3.2 provides guidance for inservice examination intervals based on the number of unacceptable snubbers discovered.	TRM Table T3.7.2-1 provides snubber visual inspection intervals based on the number of unacceptable snubbers discovered. These requirements are similar to the provisions of NRC GL 90-09.
5.	Inservice Examination Failure Evaluation	Paragraph 2.3.4 states that snubbers not meeting examination and acceptance criteria shall be evaluated to determine the cause of unacceptability.	TRM T 3.7.2, Required Action A.1.1 and A.1.2 accomplish the same requirements as OM-4, Paragraph 2.3.4. TRM Bases TB 3.7.2 states that a snubber is considered unacceptable if it fails the acceptance criteria of the visual inspection. The cause for rejection of a snubber during visual inspection is clearly established and remedied for that snubber.

	Criteria	ASME/ANSI OM Part 4 -1988	DAEC, TRM Section T 3.7.2 Requirements and Bases
Inservice Operability Test			
1.	Inservice Operability Test Requirements	Paragraph 3.2.1.1, states that snubber operational readiness tests shall verify activation, release rate, and breakaway force or drag force by either an in-place or bench test.	TSR 3.7.2.3, states that snubbers shall be functionally tested either in-place or in a bench test. TSR 3.7.2.3(b) states that the hydraulic snubber functional test is to verify: (1) activation is achieved within specified range of velocity or acceleration in tension and compression, and (2) bleed or release rate is within the specified range in compression or tension. TSR 3.7.2.3(c) states that the mechanical snubber functional test is to verify: (1) drag force of any snubber in tension or compression is less than specified maximum drag force; (2) activation is achieved within the specified range of velocity or acceleration in both tension and compression; and (3) snubber release rate is within the specified range in compression and tension.
2.	Snubber Sample Size	Paragraph 3.2.3 states that each defined test plan group shall use either a 10% sampling plan; a “37 testing sample plan;” or a “55 testing sample plan” during each refueling outage.	TSR 3.7.2.3 states that functional tests will be performed in-place or in a bench on a representative sample of 10% of each type (mechanical or hydraulic) of snubber in use.
3.	Additional Sampling	The snubbers which have been found unacceptable per the testing criteria shall be subject to paragraph 3.2.3.1 (b), which states that the additional sample size must be at least one-half the size of the initial sample size of the “defined test plan group” of snubbers.	TRM Section T 3.7.2, Required Action B.3, requires an additional 5% sample of snubbers, based upon the type of snubber that failed.

	Criteria	ASME/ANSI OM Part 4 -1988	DAEC, TRM Section T 3.7.2 Requirements and Bases
4.	Failure Evaluation	Paragraph 3.2.4.1 states that snubbers not meeting the operability testing acceptance criteria in paragraph 3.2.1 shall be evaluated to determine the cause of the failure.	TRM Section T 3.7.2, Required Actions B.4.1 and B.4.2 require that the licensee determine that the cause of failure is not due to manufacture or design deficiency; or perform TSR 3.7.2.3 on all snubbers subject to the same defect. TRM Bases TB 3.7.2, states that if a snubber being tested is found inoperable, the subject snubbers are to be replaced or restored to operable status and an engineering evaluation performed. This evaluation is to determine the snubber mode of failure.
5.	Test Failure Mode Groups	Paragraph 3.2.4.2 states that unacceptable snubber(s) shall be categorized into failure mode group(s). A test failure mode group(s) shall include all unacceptable snubbers that have a given failure mode, and all other snubbers subject to the same failure mode.	TRM Section T 3.7.2 does not specifically address “Failure Mode Groups,” for unacceptable snubbers. However, TRM accomplishes a similar intent as “Failure Mode Grouping.”
6.	Corrective Actions for 10% Sample Plan	Paragraph 3.2.5.1 states that unacceptable snubbers shall be repaired, modified, or replaced.	TRM Bases TB 3.7.2 states that when a snubber is found inoperable, the snubber is to be replaced or restored to operable status and an engineering evaluation performed.

Inservice Examination Requirements

(1) Visual Examination

TRM TSR 3.7.2.2 states that visual inspections shall verify that: (a) there are no indications of damage or impaired operability; (b) attachments to the foundation or supporting structure are secure; and (c) fasteners for the attachment of the snubber anchorage are secure. The visual examination per TRM TSR 3.7.2.2 verifies visible indication of damage or impaired operability of snubbers as well as its attachments and supports. OM-4, paragraph 2.3.1.1, requires snubber visual examinations to identify impaired functional ability due to physical damage, leakage, corrosion, or degradation. Therefore, because the TRM Section T 3.7.2 snubber visual examination requirements will, like OM-4, identify indications of damage or impaired operability, the NRC staff finds them to be equivalent to the snubber visual examination requirements of OM-4 paragraph 2.3.1.1, and will provide an acceptable level of quality and safety.

(2) Visual Examination Interval Frequency

TRM Table T3.7.2-1 provides snubber visual inspection interval frequency requirements that are different than the OM-4 visual inspection interval requirements. Table T3.7.2-1 incorporates the visual inspection interval frequency as specified in GL 90-09, "Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions." GL 90-09 acknowledges that the visual inspection interval frequency (as contained in OM-4) is excessively restrictive and that licensees with large snubber populations have spent a significant amount of resources and have subjected plant personnel to unnecessary radiological exposure to comply with the visual examination requirements. GL 90-09 states that its alternative schedule (interval frequency) for visual inspection provides the same confidence level as that provided by OM-4. Therefore, because TRM Section Table T3.7.2-1 incorporates the visual inspection interval frequency guidance specified in GL 90-09, the NRC staff finds that TRM Section T 3.7.2 will provide an acceptable level of quality and safety.

(3) Method of Visual Examination

IWF-5200(a) and IWF-5300(a) requires that preservice and inservice examination be performed in accordance with OM-4, using the VT-3 visual examination method described in IWA-2213. IWA-2213 states that VT-3 examinations are conducted to determine the general mechanical and structural condition of components and their supports by verifying parameters such as clearance, settings, and physical displacements; and to detect discontinuities and imperfections, such as loss of integrity at bolts and 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. The licensee states that IWA-2213 will be used to identify the examination technique utilized and IWA-2300 will be used to qualify/certify the VT-3 examiners. Therefore, because the licensee will be using the same visual examination method required by OM-4, the NRC staff finds that the licensee's method of snubber visual inspection provides an acceptable level of quality and safety.

(4) Subsequent Examination Intervals

TRM Table T3.7.2-1 establishes subsequent snubber visual inspection intervals based on the number of unacceptable snubbers discovered, in lieu of OM-4, paragraph 2.3.2 requirements. These requirements are equivalent to the guidance provided in GL 90-09, which has been approved for use by the NRC. Therefore, the NRC staff finds that the subsequent examination intervals contained in TRM Table T3.7.2-1 provide an acceptable level of quality and safety.

(5) Inservice Examination Failure Evaluation

OM-4, paragraph 2.3.4.1 requires that snubbers not meeting examination criteria be evaluated to determine the cause of unacceptability. OM-4, paragraph 2.3.4.2, states that snubbers found unacceptable may be tested in accordance with the requirements of paragraph 3.2. In TRM Section T 3.7.2, Required Actions A.1.1 and A.1.2 accomplish the same objectives as OM-4, paragraph 2.3.4. TRM Bases TB 3.7.2 states that a snubber is considered unacceptable if it fails the visual inspection acceptable criteria of TSR 3.7.2.2. The cause for rejection of a snubber during visual inspection is clearly established and remedied for that snubber. Therefore, the NRC staff finds that the TRM's inservice examination failure evaluation requirements provide an acceptable level of quality and safety.

<u>Inservice Operability Test Requirements</u>

(1) Inservice Operability Test

TSR 3.7.2.3, states that snubbers shall be functionally tested either in-place or in a bench test. TSR 3.7.2.3(b) states that the hydraulic snubber functional test is to verify (1) activation is achieved within specified range of velocity or acceleration in tension and compression, and (2) bleed or release rate is within the specified range in compression or tension. TSR 3.7.2.3(c) states that the mechanical snubber functional test is to verify (1) drag force of any snubber in tension or compression is less than the specified maximum drag force; (2) activation is achieved within the specified range of velocity or acceleration in both tension and compression; and (3) snubber release rate is within the specified range in compression and tension. OM-4, paragraph 3.2.1.1, states that snubber operational readiness tests verify activation, release rate, and breakaway force or drag force by either an in-place or bench test. Because the TRM test, like the OM-4 test, verifies activation, release rate, and drag force with an in-place or bench test, the NRC staff finds that the TRM requirements are equivalent to the snubber operability test requirements of OM-4, paragraph 3.2.1. Therefore, the NRC staff finds that the TRM operability test requirements provide an acceptable level of quality and safety.

(2) Snubber Sample Size

TSR 3.7.2.3 states that functional tests will be performed in-place or in a bench on a representative sample of 10 percent of each type (mechanical or hydraulic) of snubber in use. This functional testing will be performed every refueling outage. OM-4, Section 3.2.3 requires either a 10 percent testing sampling plan, a "37 testing sample plan," or a "55 testing sample plan." The DAEC TRM is using a 10 percent sample criteria which is equivalent to the 10 percent sample testing requirements of OM-4. As a result, the NRC staff considers the number of snubbers tested during outages to be equivalent to the OM-4 requirements. Therefore, the NRC staff finds that the TRM snubber sample size provides an acceptable level of quality and safety.

(3) Additional Sampling

TRM Section T 3.7.2, Required Action B.3, requires that if snubbers are found inoperable during performance testing of TSR 3.7.2.3, an additional 5 percent sample of snubbers, based upon the type of snubber that failed, will be tested. OM-4, paragraph 3.2.3.1(b) states that the additional sample size must be at least one-half the size of the initial sample size of the "defined test plan group" of snubbers. That is, for a 10 percent sample program, an additional 5 percent of the same type of snubber in the overall population would need to be tested. Since the DAEC TRM uses a 10 percent sample size for snubber testing (as discussed in the previous paragraph), the NRC staff finds that the TRM T 3.7.2 requirement to sample an additional 5 percent is equivalent to the additional sampling requirements in OM-4, paragraph 3.2.3.1(b) and, as such, will provide an acceptable level of quality and safety.

(4) Inservice Operability Failure Evaluation

OM-4 paragraph 3.2.4.1 requires that snubbers not meeting operability testing acceptance criteria in paragraph 3.2.1 be evaluated to determine the cause of the failure. The cause of failure evaluation requires review of other unacceptable snubbers and determination of whether other snubbers of similar design would require further examination. TRM Section T 3.7.2, Required Actions B.4.1 and B.4.2 require that the licensee determine that the cause of failure is not due to manufacture or design deficiency, or that TSR 3.7.2.3 be performed on all snubbers subject to same defect. TRM Bases TB 3.7.2 states that if a snubber being tested is found inoperable, the subject snubbers are to be replaced or restored to operable status and an engineering evaluation performed. This evaluation is to determine the snubber mode of failure. The NRC staff finds that the requirements in TRM Section T 3.7.2 and its Bases provide corrective actions for evaluation of inoperable snubbers that are equivalent to the OM-4 requirements and, as such, will provide an acceptable level of quality and safety.

(5) Test Failure Mode Groups

OM-4 paragraph 3.2.4.2 requires that unacceptable snubber(s) be categorized into failure mode group(s). A test failure mode group shall include all unacceptable snubbers that have a given failure mode, and all other snubbers subject to the same failure mode. TRM Bases TB 3.7.2 states that if a snubber being tested is found inoperable, the subject snubbers are to be replaced or restored to operable status and an engineering evaluation will be performed. This evaluation is to determine the snubber mode of failure. The licensee states that failures are not defined in terms of failure mode groups. The TRM, however, does require sample expansions based on the type of failure:

- TRM 3.7.2, Required Action A.1.1, requires the licensee to “Clearly establish and remedy the cause of the rejection for that snubber and for other snubbers that may be generically susceptible.” The NRC staff considers that this requirement would establish the failure mode group.

- TRM 3,7.2, Required Action B.3, requires sample expansions based on snubber type failure.

- TRM, Required Actions B.4.1 and B.4.2 address the testing of all snubbers found subject to the same manufacturing or design deficiency.

- TRM TSR 3.7.2.1 addresses actions required following a system transient.

- TRM Bases, 3.7.2 further describes “Generically Susceptible Snubbers” as snubbers subject to the same environmental conditions.

TRM Section T 3.7.2 does not specifically address “Failure Mode Groups,” for unacceptable snubbers. However, the NRC staff considers that the surveillance requirements of TRM Section T 3.7.2, including its Bases, accomplish the same intent as “Failure Mode Grouping.” Therefore, the NRC staff finds that the TRM requirements are equivalent to the OM-4 requirements, and will provide an acceptable level of quality and safety.

(6) Inservice Operability Testing Corrective Actions (for 10 percent testing sample plan)

OM-4, paragraph 3.2.5.1 requires that unacceptable snubbers be adjusted, repaired, modified, or replaced. TRM Bases TB 3.7.2 states that when a snubber is found inoperable, the subject snubbers are to be replaced or restored to operable status and engineering evaluation performed. The NRC staff finds that the TRM corrective actions associated with unacceptable snubbers at DAEC are equivalent to the corrective actions in the OM-4 requirements and, as such, will provide an acceptable level of quality and safety.

3.1.6 Conclusion for Relief Request NDE-R001

Based on the above evaluation, the NRC staff finds that snubber visual examinations and functional testing, conducted in accordance with DAEC TRM Section T 3.7.2, provides reasonable assurance of snubber operability and provides a level of quality and safety equivalent to that of the ASME Code, Section XI, subarticles IWF-5200(a) and (b), and IWF-5300(a) and (b). Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee's proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC.

It should be noted that in authorizing Relief Request NDE-R001, TRM Section T 3.7.2 and its Bases become regulatory requirements that may be used in lieu of ASME Code, Section XI requirements for performing ISI and testing of snubbers. Changes to these requirements must be reviewed and approved by the NRC staff for authorization pursuant to 10 CFR 50.55a(a)(3) or as an exemption pursuant to 10 CFR 50.12.

3.2 Relief Request NDE-R002

3.2.1 Components for Which Relief is Requested

The request is applicable to Class 1, pressure retaining welds in piping subject to ASME Code, Section XI, 2001 Edition, Appendix VIII, Supplement 11, "Qualification Requirements for Full Structural Overlaid Wrought Austenitic Piping Welds" examination.

Examination Categories: B-F, B-J

Item Numbers: B5.10, B5.20, B5.30, B9.11, B9.21, or B9.31 that are overlaid

3.2.2 Code Requirements

The fourth 10-year ISI examinations are to 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, Appendix VIII and supplements to Appendix VIII of the 2001 Edition of Section XI must be used.

The licensee proposed to use the industry's Performance Demonstration Initiative (PDI) program as an alternative to the following paragraphs of Appendix VIII, Supplement 11:

Paragraph 1.1(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.

Paragraph 1.1(d)(1): 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 the ultrasonic detection or characterization of the cracking. Specimens containing [intergranular stress corrosion cracking] IGSCC shall be used when available.

Paragraph 1.1(e)(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.

Paragraph 1.1(e)(2): Specimens shall be divided into base and overlay grading units. Each specimen shall contain one or both types of grading units.

Paragraph 1.1(e)(2)(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 interference.

Paragraph 1.1(e)(2)(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.

Paragraph 1.1(e)(2)(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.

Paragraph 1.1(e)(2)(b)(1): An overlay grading unit shall include the overlay material and the base metal-to-overlay interference of at least 6 sq. in. The overlay grading unit shall be rectangular, with minimum dimensions of 2 in.

Paragraph 1.1(e)(2)(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.

Paragraph 1.1(e)(2)(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, 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.

Paragraph 1.1(f)(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.

Paragraph 1.1(f)(3): Base metal cracking used for length sizing demonstrations shall be oriented circumferentially.

Paragraph 1.1(f)(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.

Paragraph 2.0: 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.

Paragraph 2.1: 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.

Paragraph 2.2(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.

Paragraph 2.3: 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.

Paragraph 3.1: 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.

Paragraph 3.2(a): The [root mean square] 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 75% through-base-metal position.

Paragraph 3.2(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.

Paragraph 3.2(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.

3.2.3 Licensee's Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposed using the PDI program in lieu of the requirements of ASME Section XI, 2001 Edition, Appendix VIII, Supplement 11. The Electric Power Research Institute (EPRI) PDI program is described in the submittal dated June 30, 2006.

3.2.4 Licensee's Basis for the Alternative

The licensee's submittal dated June 30, 2006, provided the following basis for the proposed alternative:

> Paragraph 1.1(d)(1), requires that all base metal flaws be cracks. As illustrated below [in the licensee's submittal], 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)/Electric Power Research Institute (EPRI)) agreement 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 a least one inch, rather than the six (6) 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 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 [in the licensee's submittal] 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 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 [in the licensee's submittal].

3.2.5 NRC Staff Evaluation

The U.S. nuclear utilities created the PDI to implement performance demonstration requirements contained in Appendix VIII of Section XI of the Code. To this end, PDI has developed a program for qualifying equipment, procedures, and personnel for examinations of weld overlays in accordance with the ultrasonic testing (UT) criteria of Appendix VIII, Supplement 11. Prior to the Supplement 11 program, EPRI was maintaining a performance demonstration program for weld overlay qualification under the Tri-party Agreement between the NRC, EPRI, and the BWROG (Reference 11). Instead of having two programs with similar objectives, the NRC staff recognized the PDI program for weld overlay qualifications as an acceptable alternative to the Tri-party Agreement (Reference 12).

The PDI program is routinely assessed by the NRC staff for consistency with current Code and proposed Code changes. At present, the PDI program does not fully comport with the existing requirements of Supplement 11. PDI presented the differences at public meetings in which the NRC participated (References 13, 14). The differences relate to flaw location within test specimens and fabricated flaw tolerances. The changes in flaw location, permitted using test specimens from the Tri-party Agreement, and the changes in fabrication flaw tolerances provide UT acoustic responses similar to the responses associated with intergranular stress corrosion cracking. Based on the discussions at these public meetings and the review presented in a PNNL technical letter report submitted in support of the NRC staff's analysis of Carolina Power and Light Company's request for relief (RR-31), the NRC staff determined that the PDI program provides an acceptable level of quality and safety.

Evaluations of the differences identified in the PDI program with Supplement 11, Paragraphs 1.1(b), 1.1(d)(1), 1.1(e)(1), 1.1(e)(2), 1.1(e)(2)(a)(1), 1.1(e)(2)(a)(2), 1.1(e)(2)(a)(3), 1.1(3)(2)(b)(1), 1.1(e)(2)(b)(2), 1.1(e)(2)(b)(3), 1.1(f)(1), 1.1(f)(3), 1.1(f)(4), 2.0, 2.1, 2.2(d), 2.3, 3.1, 3.2(b), and 3.2(c) are as discussed below.

Paragraph 1.1(b) of Supplement 11 states limitations to the maximum thickness for which a procedure may be qualified. The Code states that "The specimen set must include at least one specimen with overlay thickness within minus 0.10-inch to plus 0.25-inch of the maximum nominal overlay thickness for which the procedure is applicable." The Code requirement addresses the specimen thickness tolerance for a single specimen set, but is confusing when multiple specimen sets are used. The PDI proposed alternative states that "the specimen set shall include specimens with overlay not thicker than 0.10-inch more than the minimum thickness, nor thinner than 0.25-inch of the maximum nominal overlay thickness for which the examination procedure is applicable." The proposed alternative provides clarification on the application of the tolerance. The tolerance is unchanged for a single specimen set; however, it clarifies the tolerance for multiple specimen sets by providing tolerances for both the minimum and maximum thicknesses. The proposed wording eliminates confusion while maintaining the intent of the overlay thickness tolerance. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(d)(1) requires that all base metal flaws be cracks. PDI determined that certain Supplement 11 requirements pertaining to location and size of cracks would be extremely difficult to achieve. For example, flaw implantation requires excavating a volume of base material to allow a pre-cracked coupon to be welded into this area. This process would add weld material to an area of the specimens that typically consists of only base material, and could potentially make ultrasonic examination more difficult and not representative of actual field conditions. In an effort to satisfy the requirements, PDI developed a process for fabricating flaws that exhibit crack-like reflective characteristics. Instead of all flaws being cracks as required by Paragraph 1.1(d)(1), the PDI weld overlay performance demonstrations contain at least 70 percent cracks, with the remainder being fabricated flaws exhibiting crack-like reflective characteristics. The fabricated flaws are semi-elliptical with tip widths of less than 0.002-inches. The licensee provided further information describing a revision to the PDI program alternative to clarify when real cracks, as opposed to fabricated flaws, will be used; "Flaws shall be limited to the cases where implantation of cracks produces spurious reflectors that are uncharacteristic of actual flaws." The NRC has reviewed the flaw fabrication process, compared the reflective characteristics between actual cracks and PDI-fabricated flaws, and found that the fabricated flaws for this application are acceptable. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(e)(1) requires that at least 20 percent but not less than 40 percent of the flaws be oriented within ±20 degrees of the axial direction (of the piping test specimen). Flaws contained in the original base metal heat-affected zone satisfy this requirement; however, PDI excludes axial fabrication flaws in the weld overlay material. PDI has concluded that axial flaws in the overlay material are improbable because the overlay filler material is applied in the circumferential direction (parallel to the girth weld); therefore, fabrication anomalies would also be expected to have major dimensions in the circumferential direction. The NRC staff finds this approach to implantation of fabrication flaws to be reasonable; therefore, the NRC staff considers PDI's application of flaws oriented in the axial direction to be acceptable. Therefore, the NRC staff finds this PDI program revision acceptable.

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

Paragraph 1.1(e)(2) requires that specimens be divided into base metal and overlay grading units. The PDI program adds clarification with the addition of the word "fabrication" and ensures flaw identification by ensuring that all flaws will not be masked by other flaws with the addition of "Flaws shall not interfere with ultrasonic detection or characterization of other flaws." The NRC staff finds that PDI's change provides an acceptable clarification regarding flaw identification. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(e)(2)(a)(1) requires that a base grading unit include at least 3 inches of the length of the overlaid weld, and that the base grading unit include the outer 25 percent of the overlaid weld and base metal on both sides. The PDI program reduced the criteria to 1-inch of the length of the overlaid weld and eliminated from the grading unit the need to include both sides of the weld. The proposed change permits the PDI program to continue using test specimens from the existing weld overlay program that have flaws on both sides of the welds. These test specimens have been used successfully for testing the proficiency of personnel for over 16 years. The weld overlay qualification is designed to be a near-side (relative to the weld) examination, and it is improbable that a candidate would detect a flaw on the opposite side of the weld due to the sound attenuation and re-direction caused by the weld microstructure. However, the presence of flaws on both sides of the original weld (outside the PDI grading unit) may actually provide a more challenging examination, as candidates must determine the relevancy of these flaws, if detected. Therefore, the NRC staff concludes that PDI's use of the 1-inch length of the overlaid weld base grading unit and elimination from the grading unit of the need to include both sides of the weld, as described in the revised PDI program alternative, is acceptable. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(e)(2)(a)(2) requires that when base metal cracking penetrates into the overlay material, a portion of the base grading unit shall not be used as part of the overlay grading unit. The PDI program adjusts for the changes in Paragraph 1.1(e)(2)(a)(2) and conservatively states that when base metal flaws penetrate into the overlay material, no portion of it shall be used as part of the overlay fabrication grading unit. The PDI program also provided clarification by the addition of the term "flaws" for "cracks" and the addition of "fabrication" to overlay grading unit. The NRC staff agrees that the PDI program alternative provides clarification and conservatism. Therefore, the NRC staff finds this PDI program revision acceptable.

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

Paragraph 1.1(e)(2)(b)(1) requires that an overlay grading unit include the overlay material and the base metal-to-overlay interface of at least 6 square inches. The overlay grading unit shall be rectangular, with minimum dimensions of 2 inches. The PDI program reduces the base metal-to-overlay interface to at least 1 inch (in lieu of a minimum of 2 inches) and eliminates the minimum rectangular dimension. This criterion is necessary to allow use of existing examination specimens that were fabricated in order to meet NRC GL 88-01, "NRC Position on IGSCC in BWR Austenitic Stainless Steel Piping." This criterion may be more challenging than the Code because of the variability associated with the shape of the grading unit. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(e)(2)(b)(2) requires that unflawed overlay grading units be surrounded by unflawed material for 1 inch around its entire perimeter. The PDI program redefines the area by noting that unflawed overlay fabrication grading units shall be separated by at least 1 inch of unflawed material at both ends and sufficient area on both sides to preclude interfering reflections from adjacent flaws. The relaxation in required area on the sides of the specimens, while still ensuring no interfering reflections, may be more challenging than the Code because of the possibility for having a parallel flaw on the opposite side of the weld. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(e)(2)(b)(3) requirements are contained in the PDI program. In addition, the PDI program requires that initial procedure qualification contain three times the number of flaws required for a personal qualification. To qualify new values of essential variables, the equivalent of at least one personal qualification set is required. PDI's additions are more rigorous than the Code requirements. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 1.1(f)(1) requirements are contained in the PDI program, with the clarifying change of substituting the term "flaws" for "cracks." In addition, the PDI program includes the requirements that sizing sets contain a distribution of flaw dimensions to verify sizing capabilities. The PDI program also requires that initial procedure qualification contain three times the number of flaws required for a personal qualification. To qualify new values of essential variables, the equivalent of at least one personal qualification set is required. PDI's additions are more rigorous than the Code requirements. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraphs 1.1(f)(3) and 1.1(f)(4) were clarified by the PDI program by replacing the term "cracking" with "flaws" because of the use of alternative flaw mechanisms. The NRC staff finds this PDI program clarification to be acceptable.

Paragraph 2.0 requirements are contained in the PDI program alternative. In addition, the PDI program states for clarification that the overlay fabrication flaw test and the base metal flaw test may be performed separately. The NRC staff finds this PDI program clarification to be acceptable.

Paragraphs 2.1 and 2.2(d) were clarified by the PDI program by the addition of the terms "metal" and "fabrication". The NRC staff determined that the clarifications provide acceptable classification of the terms they are enhancing. The NRC staff finds this PDI program clarification to be acceptable.

Paragraph 2.3 states that, for depth sizing tests, 80 percent of the flaws shall be sized at a specific location on the surface of the specimen presented to the candidate. This provision requires detection and sizing tests to be separate. PDI revised the weld overlay program to allow sizing to be conducted either in conjunction with, or separately from, the flaw detection test. If performed in conjunction with detection, and if the detected flaws do not meet the Supplement 11 range criteria, additional specimens will be presented to the candidate with the regions containing flaws identified. Each candidate will be required to determine the maximum depth of flaw in each region. For separate sizing tests, the regions of interest will also be identified and the maximum depth and length of each flaw in the region will similarly be determined. In addition, PDI stated that grading units are not applicable to sizing tests, and that each sizing region will be large enough to contain the target flaw, but small enough such that candidates will not attempt to size a different flaw. The NRC staff agrees that the above clarification provides a basis for implementing sizing tests in a systematic, consistent manner that meets the intent of Supplement 11. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraphs 3.1 and 3.2 of Supplement 11 state that procedures, equipment and personnel (as a complete ultrasonic system) are qualified for detection or sizing of flaws, as applicable, when certain criteria are met. The PDI program allows procedure qualification to be performed separately from personnel and equipment qualification. Historical data indicate that, if ultrasonic detection or sizing procedures are thoroughly tested, personnel and equipment using those procedures have a higher probability of successfully passing a qualification test. In an effort to increase this passing rate, PDI has elected to perform procedure qualifications separately in order to assess and modify essential variables that may affect overall system capabilities. For a procedure to be qualified, the PDI program requires three times as many flaws to be detected (or sized) as shown in Supplement 11 for the entire ultrasonic system. The personnel and equipment are still required to meet Supplement 11; therefore, the PDI program exceeds ASME requirements for personnel, procedures and equipment qualification. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 3.2(a) is clarified by the PDI program by replacing the term “cracking” with “flaws” because of the use of alternative flaw mechanisms. The NRC staff finds this PDI program clarification to be acceptable.

Paragraph 3.2(b) requires that all extensions of base metal cracking into the overlay material by at least 0.10 inch be reported as being intrusions into the overlay material. The PDI program omits this criterion because of the difficulty in actually fabricating a flaw with a 0.10-inch minimum extension into the overlay, while still knowing the true state of the flaw dimensions. However, the PDI program requires that cracks be depth-sized to the tolerance specified in the Code, which is 0.125 inches. Since the Code tolerance is close to the 0.10-inch value of Paragraph 3.2(b), any crack extending beyond 0.10-inch into the overlay material would be identified as such from the characterized dimensions. The reporting of an extension in the overlay material is redundant for performance demonstration testing because of the flaw sizing tolerance. Therefore, the NRC staff finds this PDI program revision acceptable.

Paragraph 3.2(c) is renumbered to Paragraph 3.2(b) but retained in its entirety. The NRC staff finds this PDI program change to be acceptable.

3.2.6 Conclusion for Relief Request NDE-R002

Based on the above evaluation, the NRC staff has determined that the licensee’s proposed alternative to use the PDI program for weld overlay qualifications as described in the submittal, in lieu of Supplement 11 to Appendix VIII of Section XI of the Code, will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the licensee’s proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC.

3.3 Relief Request NDE-R005

3.3.1 Components for Which Relief is Requested

The request is applicable to ASME Code Class 1 and 2 welds.

Examination Categories: Class 1 B-F and B-J, Class 2 C-F-2

Item Numbers: B5.10, B5.20, B5.30, B9.11, B9.21. B9.32, B9.40, C5.51, C5.81

3.3.2 Code Requirements

ASME Code Section XI 2001 Edition with 2003 Addenda, IWB-2500-1 requires, in part, that for each successive 10-year ISI interval, 100 percent of Category B-F welds for the ASME Class 1 piping 4" nominal pipe size (NPS) and greater be selected for volumetric and surface examination.

IWB-2500-1 requires, in part, that for each successive 10-year ISI interval, 100 percent 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 ISI interval, 100 percent 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 percent 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 ISI interval, 25 percent 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 ISI interval, 25 percent 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 ISI interval, 7.5 percent of Category C-F-2 piping welds 3/8" or greater nominal wall thickness for ASME Class 2 piping greater than 4" NPS be selected for surface and volumetric examination.

IWC-2500-1 requires, in part, that for each successive 10-year ISI interval, at least 7.5 percent of Category C-F-2 piping welds greater than 1/5" nominal wall thickness for ASME Class 2 piping greater than or equal to 2" NPS and less than or equal to 4" NPS be selected for surface and volumetric examination.

IWC-2500-1 requires, in part, that for each successive 10-year ISI interval, 7.5 percent of Category C-F-2 piping socket welds be selected for surface examination.

IWC-2500-1 requires, in part, that for each successive 10-year ISI interval, 7.5 percent of Category C-F-2 welds for ASME Class 2 pipe branch connections of branch piping 2" NPS or greater be selected for surface examination.

3.3.3 Licensee's Proposed Alternative

As discussed in the FPL Energy submittal dated December 21, 2006 (Reference 2), pursuant to 10 CFR 50.55a(a)(3)(i), the licensee proposes to use the DAEC Risk-Informed Inservice Inspection (RI-ISI) program as an alternative to the ASME Section XI, 2001 Edition through 2003 Addenda, inspection requirements for Class 1 and 2 piping welds (Examination Categories B-F, B-J, and C-F-2).

The DAEC RI-ISI program was initially submitted to the NRC in a letter dated March 29, 2002 (Reference 3), and supplemented in a letter dated September 6, 2002 (Reference 4). The DAEC RI-ISI program was reviewed and approved by the NRC for use in the third 10-year ISI interval in a letter dated January 17, 2003 (Reference 5). The licensee proposes to extend the same RI-ISI program from the third 10-year ISI interval to the fourth 10-year ISI interval.

3.3.4 Licensee's Basis for the Alternative

In Reference 2, the licensee provided the basis for the proposed alternative as follows:

> 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 6]. 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 [of Reference 2]. 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. Since the final consequence ranking has not changed and since the number of exams is not being decreased, the cumulative risk impact analysis performed in support of the original RI-ISI submittal is unaffected. This evaluation, which used bounding values of conditional core damage probability (CCDP) and conditional large/early release probability (LERP), concludes that unacceptable risk impacts will not occur from implementation of the RI-ISI program and that acceptance criteria of Regulatory Guide 1.174 are satisfied. 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 9] and continues to meet EPRI TR-112657 and [Regulatory] 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 5]. 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 2001 Edition 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 [probabilistic risk assessment] 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. The base large early release frequency is 1.23E-06 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.

3.3.5 NRC Staff Evaluation

The licensee is requesting relief for continued use of the approved RI-ISI program plan in the fourth 10-year ISI interval instead of the ASME Section XI program. An acceptable RI-ISI program plan is expected to meet the five key principles of risk-informed decisionmaking, discussed in Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping" (Reference 7), Standard Review Plan 3.9.8 (Reference 8), NUREG-0800 Chapter 19 (Reference 15), and EPRI Topical Report TR-112657, Rev. B-A (Reference 6), as stated below.

1. The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change.

2. The proposed change is consistent with the defense-in-depth philosophy.

3. The proposed change maintains sufficient safety margins.

4. When proposed changes result in an increase in core damage frequency (CDF) and/or Large Early Release Frequency (LERF), the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement.

5. The impact of the proposed change should be monitored by using performance measurement strategies.

The first principle is met in this relief request because an alternative ISI program may be authorized pursuant to 10 CFR 50.55a(3)(i) and, therefore, an exemption request is not required.

The second and third principles require assurance that the alternative program is consistent with the defense-in-depth philosophy and that sufficient safety margins are maintained, respectively. The methodology used to develop the fourth 10-year RI-ISI program interval is unchanged from the methodology approved for use in the third 10-year RI-ISI interval program at DAEC. Assurance that the second and third principles are met is based on the application of the approved methodology and not on the particular inspection locations selected. Therefore, the NRC staff finds that the second and third principles are met.

The fourth principle requires an estimate of the change in risk, and the change in risk is dependent on the number and location of inspections in the proposed ISI program compared to the number and location of inspections that would be inspected using the requirements of ASME Section XI. Pursuant to EPRI Topical Report TR-112657 Revision B-A, a change in risk measurement should consider the discontinuance of ASME Code required inspections, as well as any new inspections resulting from the application of its methodology. The PRA inputs reported in the DAEC fourth interval RI-ISI relief to calculate the change in risk are derived from the DAEC Revision 5B PRA model. The baseline CDF from this model excluding internal flooding initiated sequences was calculated to be 1.10E-5 per year. The baseline LERF was calculated to be 1.23E-6 per year.

The licensee states in Reference 2 that the DAEC PRA has been through a peer review as part of the BWROG 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. As further clarified in Reference 2, all major issues and observations from the BWROG certification (i.e., Level A and B facts and observations) associated with the LERF calculation have been addressed and incorporated into the Revision 5 PRA model. The licensee indicates in Reference 2 that the DAEC RI-ISI program has been developed in accordance with the EPRI methodology contained in EPRI Topical Report TR-112657 Revision B-A and continues to meet Regulatory Guide 1.174 risk acceptance criteria.

As described in Reference 2, the licensee has re-evaluated the risk assessment by updating and revising the consequence ranking. Table 1 of Reference 2 shows that the consequence ranking of one reactor core isolation cooling pipe segment and one high pressure coolant injection pipe segment was increased from low to medium as an effect from the updated probabilistic safety assessment model. Although the risk category associated with these pipe segments increased from Category 7 to Category 6, the change did not affect the exam selections because neither Category 7 nor Category 6 pipe segments need to be inspected. The risk impact analysis performed in support of the original DAEC RI-ISI submittal was conducted using the upper bound values of conditional core damage probability and conditional large early release probability, and the upper bound conditional probabilities are the same for Category 6 and Category 7. Since the risk ranking of line segments has not changed and the number of exams has not decreased as a result of this relief request, the NRC staff agrees that the results and conclusions of the original cumulative risk impact analysis are unaffected.

Given the above considerations concerning the increase in risk and DAEC PRA quality, the staff finds that the licensee's analysis provides assurance that the fourth key principle is met. Therefore, the continued use of the RI-ISI program will not cause the NRC safety goals to be exceeded.

With regard to the fifth principle of risk-informed decisionmaking, the DAEC Relief Request NDE-R005 states that the DAEC RI-ISI program was developed in accordance with the EPRI methodology contained in Topical Report TR-112657 Revision B-A. This program has been updated after a rigorous review of inputs and technical elements of the original submittal consistent with the intent of NEI 04-05, "Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems" (Reference 9) and thus continues to be a living program and meets the risk acceptance criteria of Regulatory Guide 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (Reference 10). Therefore, the NRC staff finds that the fifth key principle is also met.

Based on the above discussion, the NRC staff finds that the five key principles of risk-informed decisionmaking are met by the licensee's proposed fourth 10-year RI-ISI interval program plan. Therefore, the NRC staff concludes that the proposed RI-ISI program is an acceptable alternative to the ASME Section XI inspection requirements for Class 1 and Class 2 Code Examination Category B-F, B-J, and C-F-2 piping welds.

3.3.6 Conclusion for Relief Request NDE-R005

Based on the above evaluation, the NRC staff concludes that the proposed alternative will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC.

3.4 Relief Request NDE-R006

3.4.1 Components for Which Relief is Requested

The request is applicable to various components in ASME 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, C1.20, D2.10, F1.10, F1.20, F1.30, F1.40

3.4.2 Code Requirements

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

IWA-2210, "Visual examinations," requires that visual examinations be conducted in accordance with Section V, Article 9, Table IWA-2210-1 (shown below), as well as with 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 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 Examinations,” requires that VT-1 examinations be conducted to detect discontinuities and imperfections on the surface of components, including such conditions as cracks, wear, corrosion, or erosion.

IWA-2212, “VT-2 Examinations,” requires that: (a) VT-2 examinations be 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; and (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 that VT-3 examinations be 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, footcandles (Note 1)	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)

Notes:

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

3.4.3 Licensee's Proposed Alternative

Pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee 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 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.

3.4.4 Licensee's Basis for the Alternative

As the proposed alternative, 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).

Code Case N-686, Table 1

Visual Examination	Minimum Illumination, footcandles (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 of 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.

The licensee's basis for use of Code Case N-686 is as described below.

The 1989 Edition of ASME Section XI, which was the applicable ASME Code for the DAEC third 10-year interval, did not specify distance and illumination requirements for visual examinations; however, per an Erratum, in December 2003, the code of record for DAEC has requirements for distance and illumination. Subsequently, Code Case N-686 has been incorporated into the 2004 Edition and 2005 Addenda of ASME Code.

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

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 6 feet of the surfaces being examined or use remote examination equipment that provides demonstrated equivalent resolution. For an examiner to be with 6 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 required to erect and take 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.

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 6 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 determined the general mechanical and structural condition of a component or a component support. Table IWA-22210-1 requires the examiner to be within 4 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 or a component support. In addition, as discussed above, the use of remote examination equipment involves more plant personnel.

The industry has over 30 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 qualification 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 level of quality and safety due to As Low As Reasonably Achievable (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 6 feet and VT-3 examinations at distances well in excess of 4 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.

Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the licensee 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.

3.4.5 NRC Staff Evaluation

The licensee stated that the 1989 Edition of ASME Section XI was the applicable ASME Code of record for DAEC for the third 10-year ISI interval. The 1989 Edition of ASME Section XI did not specify distance and illumination requirements for visual examinations. Distance and illumination requirements were later added. Code Case N-686 was developed and incorporated into ASME Section XI, 2001 Edition, 2003 Addenda. However, in December 2003, an Erratum was issued that restored the visual examination requirements to the 2002 Addenda version, which specified distance and illumination requirements. Code Case N-686 was again incorporated into the 2004 Edition and the 2005 Addenda of ASME Section XI. The applicable Code edition and addenda for DAEC’s fourth 10-year ISI interval is ASME Code, Section XI, 2001 Edition through the 2003 Addenda, which was affected by the Erratum of December 2003.

The licensee stated that in order to meet the distance requirements and to gain access to areas of complete VT-2 and VT-3 visual examinations required by the ASME Code of record, paragraphs IWA-2210 through IWA-2213 and Table IWA-2210-1, remote visual equipment would have to be used or scaffolding would need to be erected and removed to perform the VT examinations. Erecting and removing of scaffolding would cause additional radiation exposure to plant personnel. The NRC staff agrees that this would constitute a hardship for the licensee.

The NRC staff reviewed the licensee's request to use ASME Code Case N-686 in lieu of the requirements of ASME Code, Section XI paragraphs IWA-2210 through IWA-2213 and Table IWA-2210-1. Since the 1989 Edition of ASME Section XI (i.e., Code of record for the DAEC third 10-year ISI interval) did not specify distance and illumination requirements for VT examinations, VT-2 and VT-3 visual examination procedures at DAEC were not required to include distance and illumination requirements for visual examinations. The NRC staff finds that industry experience in performing visual examinations to the less prescriptive requirements for proximity and illumination, along with the expertise of qualified visual examiners, provides reasonable assurance of the structural integrity of the components being examined. As such, the NRC staff finds that requiring the licensee to meet the distance and illumination requirements specified in ASME Code, Section XI, 2001 Edition through the 2003 Addenda, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The NRC staff notes that ASME Code Case N-686 has been approved by the ASME Code Committee and that the participating NRC staff members have not raised objections to the subject Code Case.

3.4.6 Conclusion for Relief Request NDE-R006

Based on the above evaluation, the NRC staff concludes that the proposed use of Code Case N-686 will provide reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC, on the basis that complying with the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Use of Code Case N-686 is authorized until such time as the Code Case is published in a future version of Regulatory Guide (RG) 1.147 and incorporated by reference in 10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this Code Case, it must follow all provisions of Code Case N-686 with conditions as specified in RG 1.147 and limitations as specified in 10 CFR 50.55a(b)(5), if any.

3.5 Relief Request NDE-R008

3.5.1 Components for Which Relief is Requested

The request is applicable to Reactor Pressure Vessel (RPV) shell-to-flange weld and head-to-flange weld.

Examination Category: B-A

Item Numbers: B1.30, B1.40

3.5.2 Code Requirements

The ASME Code, 2001 Edition through 2003 Addenda, Section XI, Appendix I, Subparagraph I-2110 requires that UT of reactor vessel shell-to-flange and head-to-flange welds be conducted in accordance with Article 4 of ASME Code Section V, supplemented by the requirements of Table I-2000-1.

3.5.3 Licensee's Proposed Alternative

The licensee states that the fourth 10-year ISI 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 and 2003 Addenda is prohibited. The licensee will use personnel and procedures qualified to the 2001 Edition with no addenda; Section XI, Appendix VIII, Supplement 4, "Qualification Requirements for the Clad/Base Metal Interface of Reactor Vessel;" and Section XI, Appendix VIII, Supplement 6, "Qualification Requirements for Reactor Vessel Welds other than Clad/Base Metal Interface," as administered by the EPRI PDI program as amended by 10 CFR 50.55a.

3.5.4 Licensee's Basis for the Alternative

As described by the licensee, Appendix VIII requirements were developed with a rigorous, item-specific performance demonstration to ensure the effectiveness of UT examinations. The performance demonstration administered by 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 the 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 Section V, Article 4.

The licensee stated that although Appendix VIII is not required for the RPV shell to flange and RPV head-to-flange welds, the use of Appendix VIII, Supplements 4 and 6 criteria for detection of flaws in these welds will be equal to or will exceed the requirements established by Section V, Article 4.

3.5.5 NRC Staff Evaluation

The 2001 Edition, 2003 Addenda of ASME Code, Section XI, IWA-2232 states, "Ultrasonic examination shall be conducted in accordance with Appendix I." Subarticle I-2120 states that vessels greater than 2 inches in thickness shall be examined in accordance with Section V, Article 4, as supplemented by Table I-2000-1. Section V, Article 4 provides a prescriptive-based process for qualifying UT of procedures and the scanning requirements for examinations. The prescriptive-based UT uses detailed criteria for setting up and calibrating equipment, calculating coverage, and detecting indications. The capability of a prescriptive-based UT examination is demonstrated with calibration blocks made from representative material containing holes and notches. Performance-based UT requires that detailed criteria be used for performance demonstration tests. The results for the tests are compared against statistically developed screening criteria. The tests are performed on representative mockups containing flaws similar to those found in operating plants. The NRC staff agrees that the performance-based tests demonstrate the effectiveness of UT personnel and procedures.

In lieu of Subarticle I-2120, the licensee proposed performing UT examinations of the subject welds in accordance with the 2001 Edition, 2003 Addenda of ASME Code, Section XI, Appendix VIII, Supplements 4 and 6, as modified by 10 CFR 50.55a. Section 50.55a limits the use of Appendix VIII to the 2001 edition with no addenda. Appendix VIII is a performance-based UT method. Examinations are performed with the scanning requirements for Supplements 4 and 6 that are provided in 10 CFR 50.55a(b)(2)(xv)(G), and the scanning volume identified in the ASME Code, Section XI, Figure IWB-2500-4 for the shell-to-flange weld and Figure IWB-2500-5 for the head-to-flange weld. The scanning requirements are: (1) for the examination of the inner 15 percent through-wall volume, scanning will be performed in four orthogonal directions to the maximum extent possible with procedures and personnel qualified to Appendix VIII, Supplement 4; (2) the remainder of the inner 15 percent through-wall volume is considered fully examined if coverage is obtained in at least one parallel and one perpendicular direction using personnel and procedures qualified for single side examination in accordance with Supplement 6; and (3) the remaining 85 percent through-wall volume is considered fully examined if coverage is obtained in one parallel and one perpendicular direction using procedures and personnel qualified for single side examination. Single side qualification criteria are provided in 10 CFR 50.55a(b)(2)(xv)(G)(2) and 10 CFR 50.55a(b)(2)(xvi). The licensee will follow 10 CFR 50.55a(b)(2)(xv)(G) for the vessel-to-flange and head-to-flange welds.

The NRC staff concludes that the procedures, equipment, and personnel qualified to Appendix VIII through the PDI program have shown a high probability of flaw detection, and have increased the reliability of examinations of weld configurations within the scope of the PDI program. Therefore, the NRC staff finds that the licensee's proposed alternative will provide an acceptable level of quality and safety.

3.5.6 Conclusion for Relief Request NDE-R008

Based on the above evaluation, the NRC staff concludes that the licensee's proposed alternative to use the 2001 Edition of the ASME Code, Section XI, Appendix VIII, Supplements 4 and 6 as modified by 10 CFR 50.55a(b)(2)(xv) and 10 CFR 50.55a(b)(2)(xvi) for the RPV shell-to-flange and head-to-flange welds will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC.

4.0 CONCLUSION

The following summarizes the conclusions as discussed in SE Sections 3.1 through 3.5:

1) With respect to relief requests NDE-R001, NDE-R002, NDE-R005, and NDE-R008, the proposed alternatives will provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the proposed alternatives are authorized for the remainder of the fourth 10-year ISI interval at DAEC.

2) With respect to relief request NDE-R006, the proposed use of Code Case N-686 will provide reasonable assurance of structural integrity. Therefore, pursuant to 10 CFR 50.55a(a)(3)(ii), the proposed alternative is authorized for the remainder of the fourth 10-year ISI interval at DAEC, on the basis that complying with the Code requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Use of Code Case N-686 is authorized until such time as the Code Case is published in a future version of RG 1.147 and incorporated by reference in 10 CFR 50.55a(b). At that time, if the licensee intends to continue implementing this Code Case, it must follow all provisions of Code Case N-686 with conditions as specified in RG 1.147 and limitations as specified in 10 CFR 50.55a(b)(5), if any.

All other ASME Code, Section XI requirements for which relief was not specifically requested and approved in these relief requests remain applicable, including third-party review by the authorized Nuclear Inservice Inspector.

5.0 REFERENCES

1. Letter from G. V. Middlesworth (FPL Energy) to NRC, “Duane Arnold Energy Center, Fourth Ten-Year Inservice Inspection Plan,” dated June 30, 2006.

2. Letter from G. V. Middlesworth, (FPL Energy) to NRC, “Duane Arnold Energy Center, Responses to Requests for Additional Information and Revised Relief Requests NDE-R005 and NDE-R007 - Fourth 10-Year Inservice Inspection Program,” dated December 21, 2006.

3. Letter from G. V. Middlesworth (Nuclear Management Company) to NRC, “Duane Arnold Energy Center, Alternative to the ASME Boiler and Pressure Vessel Code Section XI Requirements for Class 1 and 2 Piping Welds, Risk Informed Inservice Inspection Program,” dated March 29, 2002.

4. Letter from K. S. Putnam (Nuclear Management Company), to NRC, “Request for Additional Information Regarding the Duane Arnold Energy Center Risk-Informed Inservice Inspection Program,” dated September 6, 2002.

5. Letter from D. S. Hood (NRC) to M. A. Peifer (Nuclear Management Company), “Duane Arnold Energy Center - Risk-Informed Inservice Inspection Program (TAC NO. MB4751),” dated January 17, 2003.

6. EPRI Topical Report TR-112657, Revision B-A, “Revised Risk-Informed Inservice Inspection Evaluation Procedure, Final Report,” dated December 1999.

7. Regulatory Guide 1.178, “An Approach for Plant-Specific Risk-Informed Decision Making: Inservice Inspection of Piping,” dated September 2003.

8. NUREG-0800, Chapter 3.9.8, “Standard Review Plan for Trial Use for the Review of Risk-Informed Inservice Inspection of Piping,” dated September 2003.

9. NEI 04-05, “Living Program Guidance to Maintain Risk-Informed Inservice Inspection Programs for Nuclear Plant Piping Systems,” dated April 2004.

10. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” July 1998.

11. Tri-party Agreement between NRC, EPRI, and the BWROG, “Coordination Plan for NRC/EPRI/BWROG Training and Qualification Activities of NDE (Nondestructive Examination) Personnel,” dated July 3, 1984.

12. Letter from William H. Bateman (NRC) to Michael Bratton (PDI Chairman), “Weld Overlay Performance Demonstration Administered by PDI as an Alternative to Generic Letter 88-01 Recommendations,” dated January 15, 2002.

13. Memorandum from Donald G. Naujock (NRC) to Terence L. Chan (NRC), “Summary of Public Meeting Held June 12 through June 14, 2001, with PDI Representatives,” dated November 29, 2001.

14. Memorandum from Donald G. Naujock (NRC) to Terence L. Chan (NRC), “Summary of Public Meeting Held January 31- February 2, 2001, with PDI Representatives,” dated March 22, 2001.

15. NUREG-0800, Chapter 19, “Use of Probabilistic Risk Assessment in Plant-Specific, Risk-Informed Decisionmaking: General Guidance,” dated November 2002.

16. Letter from G. V. Middlesworth, (FPL Energy) to NRC, “Duane Arnold Energy Center, Response to Request for Additional Information Related to Relief Request NDE-R001” dated December 21, 2006.

Principal Contributors: A. Keim
H. Phan
G. Bedi
R. Ennis

Date: January 31, 2007