UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 4, 2012

Vice President, Operations
Entergy Operations, Inc.
Waterford Steam Electric Station, Unit 3
17265 River Road
Killona, LA 70057-3093

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3 – REQUEST FOR ALTERNATIVE TO ASME IWE-5221 REGARDING POST-REPAIR TESTING OF STEEL CONTAINMENT VESSEL OPENING (TAC NO. ME6795)

Dear Sir or Madam:

By letter dated July 27, 2011, Entergy Operations, Inc. (Entergy, the licensee), submitted Request for Alternative W3-CISI-002, pursuant to paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR). In its submittal, the licensee requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for post-repair leakage inspection of the Waterford Steam Electric Station, Unit 3 (Waterford 3), steel containment vessel. Entergy will be replacing the Waterford 3 steam generators during the 18th refueling outage, commencing in the fall of 2012. The licensee's proposed alternative test method for containment leak testing is in lieu of a Type A integrated leak rate test as required by ASME Code, Section XI, IWE-5221, "Leakage Test." The proposed alternative is applicable to Waterford 3's third 10-year inservice inspection interval which began on May 31, 2008.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed its review of the licensee's request and concludes that the proposed alternative provides an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the proposed one-time alternative for the third 10-year inservice inspection interval during the Waterford 3 Cycle 18 refueling outage, when the steam generators are planned to be replaced.

All other ASME Code, Section XI requirements for which an alternative was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The NRC staff's safety evaluation is enclosed. If you have any questions, please contact Kaly Kalyanam at 301-415-1480 or kaly.kalyanam@nrc.gov.

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Safety Evaluation

cc w/encl: Distribution via ListServ

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF FOR ALTERNATIVE W3-CISI-002, CONTAINMENT LEAK TESTING

ENTERGY OPERATIONS, INC.

WATERFORD STEAM ELECTRIC STATION, UNIT 3

DOCKET NO. 50-382

1.0 INTRODUCTION

By letter dated July 27, 2011 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112150195), Entergy Operations, Inc. (Entergy, the licensee), submitted Request for Alternative W3-CISI-002, pursuant to paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations* (10 CFR). In its submittal, the licensee requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, for post-repair leakage inspection of the Waterford Steam Electric Station, Unit 3 (Waterford 3), steel containment vessel. Entergy will be replacing the Waterford 3 steam generators (SGs) during the 18th refueling outage, commencing in the fall of 2012. The licensee's proposed alternative test method for containment leak testing is in lieu of a Type A integrated leak rate test (ILRT) as required by ASME Code, Section XI, IWE-5221, "Leakage Test." The proposed alternative is applicable to Waterford 3's third 10-year inservice inspection interval which began on May 31, 2008.

This safety evaluation addresses the ability of the proposed alternative to ensure the continued ability of the steel containment vessel to provide an acceptable level of quality and safety after the SG replacement activity.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a, "Codes and Standards," incorporates by reference the 2001 Edition through 2003 Addenda of Section XI of the ASME Code. Paragraph IWE-5221, of Subsection IWE of the ASME Code, Section XI, requires a leakage rate test following any repair and replacement activity. Paragraph IWE-5221 specifies that the leakage rate test be conducted in accordance with the provisions of 10 CFR Part 50, Appendix J, paragraph IV.A, "Containment modification," which states, in part,

> Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test, as applicable for the area affected by the modification.

Enclosure

The licensee's Code of record is the ASME Code, Section XI, 2001 Edition through 2003 Addenda.

3.0 TECHNICAL EVALUATION

3.1 ASME Code Component Affected (as stated by the licensee)

Component Numbers:	Waterford 3 Seismic Category 1, Class MC, Steel Containment Vessel (SCV)
Code References:	American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, 2001 Edition through 2003 Addenda
Examination Category:	E-A, Containment Surfaces
Item Number:	E1.11
Description:	Proposed Alternative to ASME IWE-5221, "Leakage Test"
Unit/Inspection:	Waterford 3 / Third (3rd) 10-year inspection interval
Interval Applicability:	May 31, 2008 thru July 30, 2017

The Waterford Steam Electric Station, Unit 3 (Waterford 3) SCV was originally constructed as an ASME [Code] Class MC vessel in accordance with ASME [Code] Section III, Subsection NE, 1971 Edition through Summer 1971 Addenda.

3.2 Applicable Code Requirement (as stated by the licensee)

ASME [Code] Section XI, Paragraph IWE-5221 states:

> *Except as noted in IWE-5222, repair/replacement activities performed on pressure retaining boundary of Class MC or Class CC components shall be subjected to a pneumatic leakage test in accordance with the provisions of Title 10 Part 50 of the Code of Federal Regulations, Appendix J, Paragraph IV.A.*

10CFR50, Appendix J, Paragraph IV.A states in part,

> *Any major modification, replacement of a component which is part of the primary reactor containment boundary, or resealing a seal-welded door, performed after the preoperational leakage rate test shall be followed by either a Type A, Type B, or Type C test as applicable for the area affected by the modification.*

3.3 Duration of Proposed Alternative (as stated by the licensee)

The performance of a localized leak test is a one-time alternative for the ASME Code repair/replacement activity associated with the Waterford 3 SCV affected by the replacements of the Waterford 3 steam generators and RVCH [reactor vessel closure head].

3.4 Reason for Request (as stated by the licensee)

The Waterford 3 SCV is a free standing steel pressure vessel, consisting of an all-welded vertical cylinder with a hemispherical upper dome and an ASME ellipsoidal bottom head. The SCV was designed, fabricated, erected, and tested in accordance with the requirements of Section III, Subsection NE of the ASME Code for Class MC Components, 1971 Edition, Summer 1971 Addenda. During original construction of the SCV, a construction hatch was installed for transporting major components into and out of containment. This hatch consists of a 32 ft diameter steel barrel which is capped on the inside of containment with a torispherical hatch cover butt-welded to the barrel. The construction hatch is located in the northeast quadrant of containment at a centerline elevation of 63.5 ft. The construction hatch is depicted in Waterford 3 FSAR [Final Safety Analysis Report] Figures 1.2-17 and 1.2-20.

Entergy will be replacing the Waterford 3 steam generators (SGs) and reactor vessel closure head (RVCH) during the fall 2012 refueling outage. These replacement activities require the opening of the SCV construction hatch to provide access for the removal of the original steam generators (OSGs) and RVCH as well as the installation of the replacement SGs (RSGs) and replacement RVCH. Following replacement of these major components, the SCV construction hatch will be restored to its original design requirements.

Once the SCV has been restored, a leakage test in accordance with IWE-5221 would be required. ASME [Code] IWE-5221 specifies that Class MC components undergo pneumatic leakage testing by either a Type A, Type B, or Type C test in accordance with Paragraph IV.A of 10CFR50, Appendix J. Entergy believes that for the nature of the repair which restores the butt weld to ASME [Code] requirements can be more effectively performed by an alternative leakage test.

3.5 Proposed Alternative and Basis for Use (as stated by the licensee)

Proposed Alternative

Entergy proposes to perform a localized leakage test on the SCV repair area in lieu of the Type A, integrated leak rate test (ILRT) specified by ASME [Code] Section XI, Paragraph IWE-5221 after restoration of the SCV pressure boundary. Specifically, the SCV hatch cover repair weld will be tested under a localized leakage "bubble test" by pressurizing the containment vessel to greater than or equal to the design pressure (P_a) which is 44.0 pounds per square inch gauge (psig). The bubble test of the repair weld will be performed after a hold time of at least 10 minutes. The test acceptance criteria will be zero detectable leakage which will be determined by the absence of bubble formation using a leak detection medium in accordance with test procedures. A VT-2 inspection will be performed with the test pressure held at or above 44.0 psig which will structurally test the SCV repair weld. Any leakage identified will be corrected and the test will be re-performed. The NDE personnel performing the VT-2 visual inspection will be certified in accordance with the requirements of [American National Standards Institute/American Society for Nondestructive Testing] ANSI/ASNT CP-189, "Qualification and Certification of Nondestructive Testing." This leakage test shall be performed prior to entry into Mode 4 after restoration of the SCV boundary.

The localized leakage bubble test on the pressure boundary weld area of the SCV will provide a more effective examination than the Type A test as required by ASME [Code] IWE-5221. Therefore, an alternative to the requirement of Paragraph IWE-5221 is requested pursuant to 10CFR50.55a(a)(3)(i) in that the proposed alternative provides an acceptable level of quality and safety.

Basis for Use

The repair and replacement activities associated with temporary removal and reinstallation of the Waterford 3 SCV construction hatch will be performed in accordance with the requirements of the 2001 Edition through 2003 Addenda of ASME [Code] Section XI. ASME [Code] Section XI, Paragraph IWA-4411 states that welding and installation activities shall be performed in accordance with the Owner's requirements and the original Construction Code. Fabrication and installation activities (i.e., cutting and welding) will be performed in accordance with the original Construction Code of Subsection NE of ASME [Code] Section III, or as reconciled to a later edition. The restoration of the construction hatch and associated weld will return the structural integrity of the SCV to its original design requirements.

Prior to performing the repair weld, the surfaces to be welded will be prepared and cleaned of any scale, rust, moisture, or other surface contaminants. A complete penetration weld will be applied over the 360° circumference of the hatch cover to barrel interface. The weld filler metal shall have a specified minimum tensile strength of 70 ksi [kilopounds per square inch] consistent with the original SCV hatch cover weld requirement. This weld will be performed by qualified personnel in accordance with ASME [Code] Section III requirements. Post weld examinations will be performed on the SCV repair which will include a full radiography of the weld, as well as a general visual examination on the SCV hatch repair area. Therefore, the SCV construction hatch will be restored to its SCV design requirements and examined to assure weld integrity.

The proposed localized leakage bubble test will provide further confirmation of SCV leak tight integrity for the weld repair. This bubble test will assure zero leakage at the repair area, while a Type A test measures total containment leakage. The acceptance criterion for leakage of the repair weld will assure that there is zero leakage around the weld. This acceptance criterion is a more stringent criterion than that of a Type A test. Pressurization to greater than or equal to design pressure will assure the structural integrity of the SCV. Therefore, if there is any leakage of the SCV at the repair weld, it would be identified by the bubble test, and corrected.

The ILRT requires additional scheduled time, manpower, dose, and test instrumentation to be installed throughout containment. The ILRT takes longer to perform and virtually stops other work from taking place inside of containment for an extended period. In addition, the ILRT provides less assurance of the quality of the repair weld of the containment vessel since it could allow some leakage through the repair weld. Therefore, a localized leak test provides a more accurate and direct method of assuring the leak tight integrity of the repair weld. The localized leak bubble test is considered a superior test for determining leakage at the repaired area as compared to a Type A test.

The proposed localized leakage test for the SCV hatch repair is also consistent with Section 9.2.4, "Containment Repairs and Modifications," of [Nuclear Energy Institute] NEI 94-01, Revision 2 ... which states:

> *Repairs and modifications that affect the containment leakage integrity require local leakage rate testing or short duration structural tests as appropriate to provide assurance of containment integrity following the modification or repair. This testing shall be performed prior to returning the containment to operation.*

The combination of a full radiography (meeting the construction code radiography acceptance criteria) and the localized leak test of the repair weld (while at design pressure) will confirm the integrity of the steel containment vessel. In accordance with the requirements of 10CFR50.55a (a)(3)(i), Entergy believes that the localized leak test provides an acceptable level of quality and safety in lieu of the ASME Code required test.

3.6 NRC Staff Evaluation

To facilitate the replacement of the Waterford Unit 3 SGs, the free-standing SCV of Waterford Unit 3 will be breached. An opening will be cut in the SCV in order to remove and replace the SGs. After the SG replacement, the SCV sections removed will be reattached through welding. Paragraph IWE-5221 of Section XI of the ASME Code requires that leakage rate testing be conducted to ensure the integrity of the repairs before returning the SCV to operable status. In lieu of the Type A, Type B, or Type C leakage rate test, the licensee proposed to perform a series of examinations and a leak test subjecting the SCV to accident pressure, to verify the leak tightness and integrity of the liner welds and the SCV.

The licensee has proposed to perform the activities described below as a part of the SCV restoration effort. The sections of the SCV that were removed will be rewelded in place in accordance with the requirements of Section III, Subsection NE of the ASME Code for

Class MC Components, 1971 Edition, Summer 1971 Addenda (Entergy's Code of Record requirements). Before performing the repair weld, the surfaces to be welded will be cleaned and examined by magnetic particle or liquid penetrate testing of the weld preparation area, and 100-percent radiography of the final repair weld will be performed. In addition, a VT-2 examination of the SCV pressure boundary welds will be conducted. To perform a weld leak test, the containment will be pressurized to a test pressure P_a of at least 44 psig for a minimum of 10 minutes. A bubble test of the repair weld and a VT-2 visual inspection will then be performed with the pressure held at or above 44 psig. A zero leakage criterion will be used for weld acceptance, which is determined by the absence of any bubbles. All NDE personnel who perform the VT-2 visual inspection will be certified in accordance with the requirements of ANSI/ASNT CP-189, "Qualification and Certification of Nondestructive Testing." The NRC staff concludes that the ASME Code, Section XI, Article IWA-4000 requirements of Repair/Replacement activities and the requirements of detecting evidence of leakage from pressure retaining components are met, and therefore, acceptable.

The personnel performing the VT-2 visual be certified in accordance with the requirements of ANSI/ASNT CP-189, "Qualification and Certification of Nondestructive Testing," and, therefore, the NRC staff concludes that the ASME Code, Section XI, IWA-2300 requirements of personnel performing qualification and certification of nondestructive examination are adequately met and therefore, are acceptable.

In summary, (1) the modified containment meets the pre-service non-destructive examination test requirements (i.e., as required by the construction code), (2) the locally welded areas are examined for essentially zero leakage using a soap bubble, or an equivalent, test, (3) the entire containment is subjected to the peak calculated containment design-basis accident pressure for a minimum of 10 minutes (steel containment) and 1 hour (concrete containment), and (4) the outside surfaces of concrete containments are visually examined as required by the ASME Code, Section XI, Subsection IWL, during the peak pressure, and that the outside and inside surfaces of the steel surfaces are examined as required by the ASME Code, Section XI, Subsection IWE, immediately after the test. Therefore, the NRC staff concludes that the proposed alternative will provide adequate assurance of structural integrity.

4.0 CONCLUSION

Based on the above, the NRC staff has determined that the proposed alternative tests provide an acceptable level of quality and safety. Therefore, pursuant to 10 CFR 50.55a(a)(3)(i), the NRC staff authorizes the use of the proposed one-time alternative for the third 10-year inservice inspection interval during the Waterford 3 Cycle 18 refueling outage, when the SGs are planned to be replaced.

All other ASME Code, Section XI requirements for which an alternative was not specifically requested and approved in this relief request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: D. Hoang

Date: January 4, 2012

The NRC staff's safety evaluation is enclosed. If you have any questions, please contact Kaly Kalyanam at 301-415-1480 or kaly.kalyanam@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch IV
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
Safety Evaluation

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