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

January 4, 2012

CORRECTIONS

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.

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

preexisting hatch hatch

SEE NOTE 1

To facilitate the replacement of the Waterford Unit 3 SG~~s~~, 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

hatch was

or as reconciled to a later edition

See Note 2 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 See Note 3 ~~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~~ affected areas of the 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.

See Note 4

See Note 5

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.

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Date: January 4, 2012