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

April 3, 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 – CORRECTION TO THE JANUARY 4, 2012, SAFETY EVALUATION FOR 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 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML112150195), Entergy Operations, Inc. (Entergy, the licensee), submitted Request for Alternative W3-CISI-002, for Waterford Steam Electric Station, Unit 3 (Waterford 3), related to the post-repair leakage inspection of the Waterford 3 steel containment vessel. 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.

By letter dated January 4, 2012 (ADAMS Accession No. ML113330137), the U.S. Nuclear Regulatory Commission staff, pursuant to paragraph 50.55a(a)(3)(i) of Title 10 of the *Code of Federal Regulations,* authorized 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 currently planned to be replaced.

By electronic mail dated January 23, 2012 (ADAMS Accession No. ML120370522), Mr. Steve Bennett of your staff informed the NRC staff of certain errors on pages 5 and 6 of the safety evaluation (SE) dated January 4, 2012. The errors relate to the incorrect description of the steel containment vessel hatch opening, testing method, and test pressure. Specific changes are outlined below.

<u>Changes to SE Section 3.6, NRC Staff Evaluation</u>

The current descriptions referred to above in SE Section 3.6 state, in part, that:

> 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 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.

The revised descriptions referred to above, with changes noted in strikeout and boldface font, will state:

> To facilitate the replacement of the Waterford Unit 3 SGs, the free-standing SCV of Waterford Unit 3 will be breached. A~~n opening~~ **pre-existing hatch** will be cut in the SCV in order to remove and replace the SGs. After the SG replacement, the SCV ~~sections~~ **hatch** 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~~ **hatch** of the SCV that ~~were~~ **was** 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) **or as reconciled to a later edition**. 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 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 *affected areas of 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.

Enclosed are corrected SE pages 5 and 6 with revision bars in the right margin indicating the areas of change. These errors did not impact Relief Request W3-CISI-001 and do not change the NRC staff's conclusions regarding this relief request for Waterford 3.

The NRC regrets any inconvenience that this may have caused. If you have any questions regarding this matter, please contact me at (301) 415-1480.

Sincerely,

for

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

Docket No. 50-382

Enclosure:
Corrected SE pages 5 and 6

cc w/encl: Distribution via Listserv

ENCLOSURE

REVISED PAGES 5 AND 6 FROM

U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION DATED JANUARY 4, 2012,

RELATED TO RELIEF REQUEST W3-CISI-002 FOR

WATERFORD STEAM ELECTRIC STATION, UNIT 3 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 <u>NRC Staff Evaluation</u>

To facilitate the replacement of the Waterford Unit 3 SGs, the free-standing SCV of Waterford Unit 3 will be breached. A pre-existing hatch will be cut in the SCV in order to remove and replace the SGs. After the SG replacement, the SCV hatch 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 hatch of the SCV that was 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) or as reconciled to a later edition. Before performing the repair weld, the surfaces to be welded will be cleaned, 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 (4) the affected areas of 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

Corrected by letter dated April 3, 2012

~~testing of the weld preparation area~~, and 100-percent radiography of the final repair weld will be performed....

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 *affected areas of 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.

Enclosed are corrected SE pages 5 and 6 with revision bars in the right margin indicating the areas of change. These errors did not impact Relief Request W3-CISI-001 and do not change the NRC staff's conclusions regarding this relief request for Waterford 3.

The NRC regrets any inconvenience that this may have caused. If you have any questions regarding this matter, please contact me at (301) 415-1480.

Sincerely,

/RA by Joseph M. Sebrosky for/

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

Docket No. 50-382

Enclosure:
Corrected SE pages 5 and 6

cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL4 R/F
RidsAcrsAcnw_MailCTR Resource
RidsNrrDeEmcb Resource
RidsNrrDorlLpl4 Resource
RidsNrrLAJBurkhardt Resource
RidsNrrPMWaterford Resource
RidsRgn4MailCenter Resource
DHoang, DE/EMCB
RidsNrrDorlDpr Resource
RidsOgcRp Resource

ADAMS Accession No. ML120580756

OFFICE	NRR/LPL4/PM	NRR/LPL4/LA	NRR/DE/EMCB/BC	NRR/LPL4/BC
NAME	NKalyanam	JBurkhardt	MMurphy	MMarkley (JSebrosky for)
DATE	3/6/12	3/5/12	3/29/12	4/3/12

OFFICIAL RECORD COPY