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John F. McCann
Director
Nuclear Safety & Licensing

CNRO-2007-00020

April 26, 2007

U. S. Nuclear Regulatory Commission
Attn.: Document Control Desk
Washington, DC 20555-0001

SUBJECT: Request for Alternative W3-ISI-003
Proposed Alternative to Extend the Second 10-Year Inservice Inspection Interval for Reactor Vessel Internal Weld Examinations

Waterford Steam Electric Station, Unit 3
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Letter from NRC to Entergy Operations, Inc, *Arkansas Nuclear One, Unit 1 – Request for Relief from the Requirements of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (TAC No. MD1399)*, December 8, 2006

2. Letter from NRC to Entergy Nuclear Operations, Inc., *Indian Point Nuclear Generating Unit No. 2 – Relief Request (RR) No. 73 (TAC No. MC7306)*, February 22, 2006

Dear Sir or Madam:

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes an alternative to the requirements of ASME Section XI, paragraph IWB-2412, *Inspection Program B*, for Waterford Steam Electric Station, Unit 3 (Waterford 3). Waterford 3 is currently in its second inservice inspection (ISI) interval, which began July 1, 1997 and ends June 30, 2007. ASME Section XI IWA-2430(d) allows a one-year extension of an interval, which would extend the interval to June 30, 2008. (Use of this one-year extension does not require approval from the NRC.) In order to comply with Code requirements, second interval examination of the reactor vessel welds (Examination Category B-A), the nozzle-to-vessel welds and inner radius sections (Examination Category B-D), and reactor vessel nozzle-to-piping welds (Examination Category B-J), must be performed during Waterford 3's spring 2008 refueling outage (RF15). Entergy proposes to perform these examinations during the fall 2009 refueling outage (RF16). Because RF16 is beyond June 30, 2008, Entergy is submitting Request for Alternative W3-ISI-003 (see Enclosure 1), which proposes an additional extension to the second ISI interval. Entergy believes extending the inspection interval to the end of RF16 for these examinations continues to provide an acceptable level of quality and safety. Entergy is submitting this request as a result of an ongoing initiative with the Westinghouse Owners Group for extending the Inservice Inspection requirements.

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The NRC staff has approved similar requests for Arkansas Nuclear One, Unit 1, and Indian Point Energy Center, Unit 2 (see References 1 and 2, respectively).

Entergy requests NRC approval by March 13, 2008 in order to support planning activities for RF15. Should you have any questions regarding this submittal, please contact Guy Davant at (601) 368-5756.

This letter contains one commitment as identified in Enclosure 2.

Very truly yours,

JG Burford for

JFM/GHD/ghd

Enclosures: 1. Request for Alternative W3-ISI-003
2. Licensee-Identified Commitments

cc: Mr. J. S. Forbes (ECH)
Mr. O. Limpias (WPO)
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U. S. Nuclear Regulatory Commission
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ENCLOSURE 1

CNRO-2007-00020

REQUEST FOR ALTERNATIVE
W3-ISI-003

ENTERGY OPERATIONS, INC.
WATERFORD STEAM ELECTRIC STATION, UNIT 3
REQUEST FOR ALTERNATIVE
W3-ISI-003

I. COMPONENTS

The affected component is the Waterford Steam Electric Station, Unit 3 (Waterford 3) reactor vessel; specifically, the following ASME Section XI Examination Categories and Item Numbers covering examinations of the reactor vessel. These examination categories and item numbers are from Table IWB-2500-1 of the 1992 Edition of ASME Section XI.

Examination Category	Item Number	Description
B-A	B1.11	Circumferential Shell Welds
B-A	B1.12	Longitudinal Shell Welds
B-A	B1.21	Circumferential Head Welds
B-A	B1.22	Meridional Head Welds
B-A	B1.30	Shell-to-Flange Weld
B-A	B1.40	Head-to-Flange Weld
B-A	B1.50	Repair Welds
B-A	B1.51	Beltline Region Repair Welds
B-D	B3.90	Nozzle-to-Vessel Welds
B-D	B3.100	Nozzle Inner Radius Areas
B-J	B9.11	Circumferential Welds in Piping (only for the reactor vessel inlet and outlet nozzle to piping welds)

Code Class: 1

References:

1. Westinghouse Owners Group Topical Report WCAP-16168-NP, *Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval*

2. Letter from the NRC to the Westinghouse Owners Group, *Acceptance for Review of Westinghouse Owners Group (WOG) Topical Report WCAP-16168-NP, Rev. 1,* Risk-Informed Extension of Reactor Vessel In-Service Inspection Interval (TAC No. MC9768), September 19, 2006

3. ASME Code Case N-691, “Application of Risk-Informed Insights to Increase the Inspection Interval for Pressurized Water Reactor Vessels,” Section XI, Division 1, November 2003

4. NRC Regulatory Guide 1.174, Revision 1, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis,” November 2002

5. R. Gramm of the NRC to G. Bischoff of the WOG, “Summary of teleconference with the Westinghouse Owners Group regarding potential one cycle relief of reactor pressure vessel shell weld inspections at pressurized water reactors related to WCAP-16168-NP, *Risk-Informed Extension of Reactor Vessel In-Service Inspection Intervals*,” January 27, 2005

6. Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds during Pre-Service and Inservice Examinations*

7. NRC Memorandum, Thadani to Collins, “Technical Basis for Revision of the Pressurized Thermal Shock (PTS) Screening Criteria in the PTS Rule (10 CFR 50.61),” December 31, 2002

8. Westinghouse Owners Group letter WOG-05-100, *Cover Letter and Template for WOG Members' Use to Request a One Operating Fuel Cycle RV ISI Relief Request (MUHP-5097/5098/5099, Tasks 2008/2059)*, March 3, 2005

9. Westinghouse Owners Group Topical Report WCAP-16088-NP, *Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation,* Rev. 1

10. Letter W3F1-2003-0075 from Entergy Operations, Inc. to the NRC, *License Amendment Request NPF-38-250, Revision to Pressure/Temperature and Low Temperature Overpressure Protection Limits for 32 Effective Full Power Years,* October 22, 2003

11. Letter from the NRC to Entergy Operations, Inc., *Waterford Steam Electric Station, Unit 3 – Issuance of Amendment Re: Pressure Temperature Limit Curves to 32 Effective Full Power Years with Power Uprate (TAC No. MC1156),* June 16, 2004

12. NRC Regulatory Guide 1.154, *Format and Content of Plant-Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors*

13. NRC Reactor Vessel Integrity Database, Version 2.0.1

II. CODE REQUIREMENTS

ASME Section XI IWB-2412, *Inspection Program B*, requires volumetric examination of essentially 100% of reactor vessel and piping pressure-retaining welds identified in Table IWB-2500-1 once each 10-year interval. IWA-2430(d) allows inspection intervals to be extended by as much as one year if this adjustment does not cause successive intervals to be altered by more than one year.

III. PROPOSED ALTERNATIVE

Pursuant to 10 CFR 50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) proposes an alternative from the requirement of IWB-2412 that pertains to volumetric examination of reactor vessel and piping pressure-retaining welds, Examination Categories B-A, B-D, and B-J welds identified in Section 1, above. IWB-2412 requires these examinations be performed once each 10-year interval. Entergy proposes to extend the inservice inspection (ISI) interval for the identified Examination Categories B-A, B-D, and B-J welds to the end of RF16 (approximately 17 months beyond the currently scheduled interval and the Code-allowed one-year extension). This request applies only to similar metal welds and not to dissimilar metal welds.

This extension will allow time for NRC to review industry efforts documented in References 1 and 2 that justify using risk-informed insights to show that extending the inspection interval from 10 to 20 years results in a change in reactor vessel failure frequency that satisfies the requirements of NRC Regulatory Guide 1.174 (Reference 3). These efforts use ASME Section XI Code Case N-691 (Reference 4) as a basis for using risk-informed insights to show that extending the inspection interval from 10 to 20 years results in a change in reactor vessel failure frequency that satisfies the requirements of NRC Regulatory Guide 1.174 (Reference 3).

IV. BASIS FOR PROPOSED ALTERNATIVE

A. Background

Waterford 3 is currently in its second inservice inspection (ISI) interval, which began July 1, 1997 and ends June 30, 2007. ASME Section XI IWA-2430(d) allows a one-year extension of an interval, which would extend the interval to June 30, 2008. (Use of this one-year extension does not require approval from the NRC.) In order to comply with Code requirements, second interval examination of the reactor vessel welds (Examination Category B-A), the nozzle-to-vessel welds and inner radius sections (Examination Category B-D), and reactor vessel nozzle-to-piping welds (Examination Category B-J), must be performed during Waterford 3's spring 2008 refueling outage (RF15). Entergy proposes to perform these examinations during the fall 2009 refueling outage (RF16).

B. Basis for Proposed Alternative

The requirements for a technical basis to extend the 10-year reactor vessel ISI interval to the end of RF16 are contained in a letter from R. Gramm of the NRC to G. Bischoff of the Westinghouse Owners Group (WOG), dated January 27, 2005 (Reference 5). The technical justification consists of five areas:

1. Plant-specific reactor vessel ISI history
2. PWR reactor vessel ISI history
3. Degradation mechanisms in the reactor vessel
4. Material condition of the reactor vessel relative to embrittlement
5. Operational experience relative to reactor vessel structural integrity challenging events

Each area is discussed below.

1. Plant-Specific Reactor Vessel ISI History

Waterford 3 is in its second ISI interval for the reactor vessel; therefore, the preservice and one inservice inspection have been performed on the Examination Category B-A, B-D, and B-J welds associated with the reactor vessel. These examinations were performed in accordance with Regulatory Guide 1.150, *Ultrasonic Testing of Reactor Vessel Welds during Pre-Service and Inservice Examinations* (Reference 6), and achieved coverage as shown in Table 1; no reportable indications were found. Based on the examination method and coverage obtained, it is reasonable to conclude that the examinations were of sufficient quality to detect any significant flaws that could challenge reactor vessel integrity.

2. PWR Reactor Vessel ISI History

As part of the technical basis for ASME Code Case N-691 (Reference 3), a survey of reactor vessel ISI history for 14 PWRs was performed. At the time of the survey, these 14 plants represented 301 total years of service and included reactor vessels fabricated by various vendors. The plants surveyed reported that no reportable findings had been discovered during examinations of Examination Category B-A, B-D, and B-J welds associated with the reactor vessels.

It is widely recognized in the fracture mechanics community that fatigue crack growth of embedded flaws is substantially smaller than that of surface breaking flaws. Surface breaking flaws in the reactor vessel cladding are typically a result of lack of fusion defects between bands of cladding. Studies performed by Pacific Northwest National Laboratory for the NRC Pressurized Thermal Shock (PTS) Risk Reevaluation (Reference 7) includes an evaluation that determined that in plants with reactor vessels constructed with multi-pass cladding, two flaws would have to be aligned on top of one another for a flaw to exist through the cladding. The probability of this occurring is very low (< 0.0001). The Waterford 3 reactor vessel is constructed using a single layer of cladding with a thickness of 7/32 inch and, therefore, is not susceptible to the condition evaluated in the PTS risk reevaluation. The pilot plant evaluation, documented in WCAP-16168-NP (Reference 1), used a single layer of cladding and a cladding thickness of 0.25 inch. However, the cladding thickness was rounded to the nearest 1/100th of total vessel thickness resulted in an assumed initial flaw size of 0.263 inch being used in the Combustion Engineering pilot plant study. Therefore, the 7/32-inch cladding of Waterford 3 is bounded by the pilot study.

Most PWRs have performed their first 10-year inspections of the subject welds and many have completed their second and third 10-year inspections. No surface-breaking or unacceptable near-surface flaws have been found in any of these inspections performed per the requirements of Regulatory Guide 1.150 or ASME Section XI Appendix VIII.

3. Degradation Mechanisms in the Reactor Vessel

The welds for which the subject examinations are conducted are similar metal low-alloy steel welds. The only currently known degradation mechanisms for this type of weld is fatigue due to thermal and mechanical cycling from operational transients. Studies have shown that while flaw growth of simulated flaws in a reactor vessel would be small, the operational transient that has the greatest contribution to flaw growth is the cooldown transient. The cooldown transient is a low-frequency transient and is not expected to occur more than a few instances during the requested inspection extension period. Therefore, any flaw growth during the requested deferral period is expected to be inherently small.

According to the WOG (Reference 8), the fatigue usage factors for the welds in the subject examinations are much less than the ASME Code design limit of 1.0 after 40 years of operation. These usage factors are calculated using a very conservative design duty cycle. It is very unlikely that more than a few of these events (e.g., heatup or cooldown) would actually occur during the extension period of this proposed alternative.

4. Material Condition of the Reactor Vessel Relative to Embrittlement

The reactor vessel beltline is the limiting area in terms of embrittlement for the subject examinations. The composition of each material in the reactor vessel beltline, along with fluence and embrittlement data, is contained in Westinghouse Owners Group Topical Report WCAP-16088-NP, *Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation,* Rev. 1 (Reference 9). This WCAP was developed to support power uprate at Waterford 3 and reflects fluence associated with the uprate. Entergy submitted the WCAP to the NRC with the power uprate license amendment request (Reference 10), which was approved by the NRC staff (Reference 11). In reviewing this information, the NRC staff concluded that the effect of the uprate was negligible. This information is provided in Table 2.

It is widely recognized that the greatest possible challenge to reactor vessel integrity for a PWR is pressurized thermal shock (PTS). A PTS event can be generally described as rapid cooling of the reactor vessel concurrent with or followed by a significant increase in pressure. 10 CFR 50.61 currently provides PTS screening criteria of RT_{PTS} equal to 270°F for plates and axial welds and RT_{PTS} equal to 300°F for circumferential welds. Based on the beltline material information contained in WCAP-16088-NP, the lower shell circumferential weld, M-1004-2, is the limiting material with regards to the PTS screening criteria. The RT_{PTS} value at 32 effective full power years (EFPY) for this weld is 53°F, which is well below the current PTS screening criterion of 300°F.

The NRC and industry recognize that a large amount of conservatism exists in the current PTS screening criteria. In the NRC PTS risk re-evaluation (Reference 7), results show that it may be possible to remove an amount of conservatism equivalent to reducing a plant's RT_{PTS} value by at least 70°F. While the exact amount of conservatism that will be removed has not been determined, it is clear that Waterford 3 will remain well below the current PTS screening criteria during the extension period.

5. <u>Operational Experience Relative to Reactor Vessel Structural Integrity Challenging Events</u>

As stated above, the greatest possible challenge to reactor vessel integrity for a PWR is PTS. Plants have taken steps, through emergency operating procedures (EOPs) and operator training, to reduce the likelihood of a PTS event. Due to implementing such measures, the number of occurrences of PTS events fleet-wide is very small. When considered over the combined fleet-wide PWR operating history, the frequency of PTS events is very small. When considering the frequency of PTS events and the length of the requested extension, the probability of a PTS event occurring during the requested extension is also very low. Combining the low probability of a PTS event with the low probability of a flaw existing in the reactor vessel (given the previously-discussed inspection history), the probability of reactor vessel failure due to PTS is also very small.

The NRC PTS risk re-evaluation (Reference 7), identifies three types of accident sequences that cause the more-severe PTS events and, thereby, dominate the risk. Waterford 3 has implemented EOPs and operator training to provide assurance that the likelihood of a severe PTS event due to these sequences that would challenge the integrity of the reactor vessel, provided a flaw is present, is very low. These sequences and details of the associated operating procedures are discussed below.

- *Sequence 1 - Any transient with reactor trip followed by one stuck-open pressurizer safety relief valve that re-closes after approximately 1 hour*

  Severe PTS events require the failure to properly control high pressure injection.

  This event is characterized as a pressurizer steam space loss-of-coolant accident (LOCA) that results in an uncontrolled Reactor Coolant System (RCS) depressurization. Upon receipt of a reactor trip, operators would enter the reactor trip recovery procedure (OP-902-000). It is expected that as RCS pressure continues to decrease, the Engineered Safety Features Actuation System (ESFAS) will actuate on low RCS pressure, providing High Pressure Safety Injection (HPSI) flow to the system. Operators would transition to the LOCA recovery procedure (OP-902-002). Reactor coolant pumps (RCPs) would be secured due to the loss of sub-cooled margin (SCM).

  RCS pressure/temperature (PT) would continue to drop until system inflow matches system outflow followed by system pressure stabilization.

  The LOCA recovery procedure provides guidance to control RCS low pressure within limits of the RCS PT curve provided SCM is adequate. Assuming the pressurizer safety relief valve closes, operators would recognize rising RCS pressure, validate that HPSI throttling criteria had been met, and throttle or secure HPSI to prevent system re-pressurization.

  If the pressurizer safety relief valve were to reclose, Waterford 3 procedures are structured such that upon stabilizing system pressure, operators would restart the RCPs once the restart criteria were met. Operators will control RCS pressure and temperature using the auxiliary spray system and the steam generators.

  Sequence 2 - Large loss of secondary steam from steam line break or stuck-open atmospheric dump valves

  Severe PTS events require the failure to properly control auxiliary feedwater flow rate and destination (e.g., away from affected steam generators), and failure to properly control high-pressure injection.

  This event is characterized as an overcooling event. Upon receiving a reactor trip, operators would enter the reactor trip recovery procedure. The affected steam generator pressure and level would decrease resulting in automatic actuation of Emergency Feedwater (EFW) and the Main Steam Isolation Signal (MSIS). These systems would function to close the affected main steam isolation valve (MSIV), close the affected main feedwater isolation valve, and provide EFW to the unaffected steam generator. Guidance is provided to operators for verifying proper MSIS and EFW response (OP-902-004). In the event of system or component malfunction, contingency actions are provided to operators to manually control the systems.

Operators would verify automatic initiation of HPSI to make up for RCS inventory shrink. (If not initiated, operators are directed to initiate HPSI.) After the steam generator fully depressurizes, which terminates overcooling, timely operator actions are necessary to "steam" the unaffected steam generator so that RCS temperature can be stabilized; otherwise, the transient could then proceed to an overheating condition, which could result in high RCS temperature and pressure. Procedure steps provide this direction to stabilize RCS temperature and to allow securing HPSI flow provided specific criteria are satisfied. These criteria are the same as in the LOCA recovery procedure for securing HPSI flow.

- <u>Sequence 3</u> – *A small-break LOCA that exceeds normal makeup capacity*

  The severity of a PTS event depends on break location (the worst location appears to be in the pressurizer line) and primary injection system flowrate and water temperature.

  A LOCA is an accident that is caused by a break in the RCS pressure boundary. The break can be as large as a double-ended guillotine break in the hot leg, or as small as a break which results in a loss of RCS fluid at a rate that is just in excess of the available makeup capacity of the plant.

  Small- and large-break LOCAs differ in their effect on the post-LOCA RCS heat removal process. For a large-break LOCA, the only path necessary for RCS heat removal in both the short and long term is the break flow with core boil-off. For small breaks, heat removal out the break is not sufficient to provide cooling and, therefore, heat removal using the steam generators is required. The Waterford 3 EOPs and operator training take this into account and provide the necessary guidance to supplement the cooling through the break by use of primary-to-secondary heat transfer with the steam generators. Guidance is provided to bound all break spectrums.

  Waterford 3 EOPs are written to industry standards. These standards are monitored through the WOG Operations Support Committee with the expectation of maintaining a high degree of technical accuracy and operating experience in their basis.

Waterford 3 operator training stresses fundamental EOP coping strategies in both the classroom and simulator forum. Included in the curriculum are procedure entry conditions, floating steps, fundamental rules, mitigation strategies, time critical actions, and background information from the basis documents. Simulator evaluation scenarios utilize critical tasks as the basis for pass/fail for crew performance. These critical tasks are those chosen to be of the utmost importance to ensure the health and safety of the public are preserved and typically include preserving and protecting fission product barriers.

V. CONCLUSION

10 CFR 50.55a(a)(3) states:

"Proposed alternatives to the requirements of (c), (d), (e), (f), (g), and (h) of this section or portions thereof may be used when authorized by the Director of the Office of Nuclear Reactor Regulation. The applicant shall demonstrate that:

(i) The proposed alternatives would provide an acceptable level of quality and safety, or

(ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety."

The current requirements for inspecting reactor vessel pressure-retaining welds have been in effect since the 1989 Edition of ASME Section XI. The industry has expended significant cost and radiological exposure to perform these inspections. The results have shown no service-induced flaws in the reactor vessel for ASME Section XI Examination Category B-A, B-D, or B-J welds associated with the reactor vessel. ASME Section XI Code Case N-691 and industry efforts have shown that risk insights can be used to extend the reactor vessel ISI interval from 10 to 20 years. This extension satisfies the change in risk requirements of Regulatory Guide 1.174 and, in accordance with 10 CFR 50.55a(a)(3)(i), maintains an acceptable level of quality and safety.

Based on these efforts having shown that the risk of reactor vessel failure with a 10-year inspection interval extension is low and achieves an acceptable level of quality and safety, it is reasonable to conclude that an extension of Waterford 3's second ISI interval to the end of RF16 will also achieve an acceptable level of quality and safety. Furthermore, items A through E of Section IV, above, provide a qualitative basis that the risk associated with extending the inspection interval to the end of RF16, is small. Based on this, Entergy considers the proposed alternative for the subject examinations at Waterford 3 to provide an acceptable level of quality and safety. Therefore, Entergy requests that the NRC staff approve the proposed alternative pursuant to 10 CFR 50.55a(a)(3)(i).

TABLE 1
WATERFORD 3 ISI RESULTS

Weld ID	ASME Weld Category	Date Last Inspected	% Coverage Obtained	# of Reportable Indications	# of Indications Currently being Monitored	Growth of indications currently being monitored (in)
01-001	B-A	1995	100	0	0	N/A
01-002	B-A	1995	46	0	0	N/A
01-003	B-A	1995	46	0	0	N/A
01-004	B-A	1995	46	0	0	N/A
01-005	B-A	1995	46	0	0	N/A
01-006	B-A	1995	46	0	0	N/A
01-007	B-A	1995	46	0	0	N/A
01-008	B-A	1995	62	0	0	N/A
01-009	B-A	1995	67	0	0	N/A
01-010	B-A	1995	100	0	0	N/A
01-011	B-A	1995	100	0	0	N/A
01-012	B-A	1995	85	0	0	N/A
01-013	B-A	1995	74	0	0	N/A
01-014	B-A	1995	100	0	0	N/A
01-015	B-A	1995	100	0	0	N/A
01-016	B-A	1995	100	0	0	N/A
01-017	B-A	1995	98	0	0	N/A
01-018	B-A	1995	98	0	0	N/A
01-019	B-A	1995	98	0	0	N/A
01-020	B-A	1995	79	0	0	N/A
01-021	B-D	1995	49	0	0	N/A
01-022	B-D	1995	97	0	0	N/A
01-023	B-D	1995	97	0	0	N/A
01-024	B-D	1995	49	0	0	N/A
01-025	B-D	1995	97	0	0	N/A
01-026	B-D	1995	97	0	0	N/A
01-027	B-D	1995	100	0	0	N/A
01-028	B-D	1995	100	0	0	N/A
01-029	B-D	1995	100	0	0	N/A

Weld ID	ASME Weld Category	Date Last Inspected	% Coverage Obtained	# of Reportable Indications	# of Indications Currently being Monitored	Growth of indications currently being monitored (in)
01-030	B-D	1995	100	0	0	N/A
01-031	B-D	1995	100	0	0	N/A
01-032	B-D	1995	100	0	0	N/A
12-001	B-J	1988	70	0	0	N/A
15-001	B-J	1989	100	0	0	N/A
14-011	B-J	1989	100	0	0	N/A
08-001	B-J	1989	100	0	0	N/A
06-001	B-J	1989	100	0	0	N/A
10-013	B-J	1995	100	0	0	N/A

TABLE 2
WATERFORD 3 REACTOR VESSEL MATERIAL INFORMATION

Major Material Region Description				Cu [wt%]	Ni [wt%]	Unirradiated RT_{NDT}		RT_{PTS} @ 32 EFPY
#	Type	Heat[1]	Location			[°F]	Method	
1	Plate	56484-1	Intermediate Shell M-1003-1	0.020	0.710	-30.0	MTEB 5-2	20
2	Plate	56488-1	Intermediate Shell M-1003-2	0.020	0.670	-50.0	MTEB 5-2	0
3	Plate	56512-1	Intermediate Shell M-1003-3	0.020	0.700	-42.0	MTEB 5-2	8
4	Plate	57286-1	Lower Shell M-1004-1	0.030	0.620	-15.0	MTEB 5-2	35
5	Plate	57326-1	Lower Shell M-1004-2	0.030	0.580	22.0	MTEB 5-2	53
6	Plate	57359-1	Lower Shell M-1004-3	0.030	.0620	-10.0	MTEB 5-2	40
7	Weld	BOLA/HODA	Intermediate Shell Axial Welds 101-124A/C	0.020	0.960	-60.0	Plant Specific	7
8	Weld	83653	Lower Shell Axial Welds 101-142A/C	0.030	0.200	-80.0	Plant Specific	7
9	Weld	88114	Intermediate-to-Lower Shell Girth (Circumferential) Weld 101-171	0.050	0.160	-70.0	Plant Specific	-30

[1] Heat data taken from NRC Reactor Vessel Integrity Database, Version 2.0.1 (Reference 13).

ENCLOSURE 2

CNRO-2007-00020

LICENSEE-IDENTIFIED COMMITMENTS

LICENSEE-IDENTIFIED COMMITMENTS

COMMITMENT	TYPE (Check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will perform the inservice inspection of the Examination Category B-A, B-D, and B-J welds associated with the reactor vessel during the subsequent Waterford 3 fall 2009 refueling outage (RF16).	✓		Fall 2009 refueling outage (RF16)