Entergy

Entergy Operations, Inc.
17265 River Road
Killona, LA 70057-3093
Tel 504-739-6660
Fax 504-739-6698
mchisum@entergy.com

Michael R. Chisum
Site Vice President
Waterford 3

W3F1-2017-0004

February 6, 2017

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

SUBJECT: Responses to Request for Additional Information from Sets 4, 5 and 6 Regarding the License Renewal Application for Waterford Steam Electric Station, Unit 3 (Waterford 3)
Docket No. 50-382
License No. NPF-38

REFERENCES:

1. Entergy letter W3F1-2016-0012 "License Renewal Application, Waterford Steam Electric Station, Unit 3" dated March 23, 2016.
2. NRC letter to Entergy "Requests for Additional Information for the Review of the Waterford Steam Electric Station, Unit 3, License Renewal Application – Set 3" dated October 12, 2016.
3. NRC letter to Entergy "Requests for Additional Information for the Review of the Waterford Steam Electric Station, Unit 3, License Renewal Application – Set 5" dated November 7, 2016.
3. NRC letter to Entergy "Requests for Additional Information for the Review of the Waterford Steam Electric Station, Unit 3, License Renewal Application – Set 6" dated November 7, 2016.

Dear Sir or Madam:

By letter dated March 23, 2016, Entergy Operations, Inc. (Entergy) submitted a license renewal application (Reference 1).

In References 2, 3 and 4, the NRC staff made Requests for Additional Information (RAI) Sets 3, 5 and 6 needed to complete its review. Three RAIs contained in these requests required longer response times than the others. Enclosure 1 provides the responses to these RAIs.

There are no new regulatory commitments contained in this submittal. If you require additional information, please contact the Regulatory Assurance Manager, John Jarrell, at 504-739-6685.

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 6, 2017.

Sincerely,

MRC/AJH

Enclosure: 1. RAI Responses – Waterford 3 License Renewal Application

cc: Kriss Kennedy
Regional Administrator
U. S. Nuclear Regulatory Commission
Region IV
1600 E. Lamar Blvd.
Arlington, TX 76011-4511
RidsRgn4MailCenter@nrc.gov

NRC Senior Resident Inspector
Waterford Steam Electric Station Unit 3
P.O. Box 822
Killona, LA 70066-0751
Frances.Ramirez@nrc.gov
Chris.Speer@nrc.gov

U. S. Nuclear Regulatory Commission
Attn: Phyllis Clark
Division of License Renewal
Washington, DC 20555-0001
Phyllis.Clark@nrc.gov

U. S. Nuclear Regulatory Commission
Attn: Dr. April Pulvirenti
Washington, DC 20555-0001
April.Pulvirenti@nrc.gov

Louisiana Department of Environmental Quality
Office of Environmental Compliance
Surveillance Division
P.O. Box 4312
Baton Rouge, LA 70821-4312
Ji.Wiley@LA.gov

Enclosure 1 to

W3F1-2017-0004

RAI Responses
Waterford 3 License Renewal Application

RAI 4.2.1-1 (Set 3)

Background:

LRA Section 4.2.1 describes Waterford Unit 3 reactor vessel neutron fluence calculations and that the methods used satisfy the criteria set forth in Regulatory Guide (RG) 1.190. The LRA also states that these methods have been approved by the NRC and are described in detail in WCAP-14040-A, Revision 4, and WCAP-16083-NP-A, Revision 0.

The staff noted that WCAP-18002-NP, Revision 0 describes neutron embrittlement TLAAs related to Waterford Unit 3 reactor vessel integrity. Specifically, Section 2 of WCAP-18002-NP, Revision 0 indicates the following:

- WCAP-14040-A, Revision 4, and WCAP-16083-NP-A, Revision 0 describe NRC-approved fluence methods, which include the one-dimensional/two-dimensional (1D/2D) flux synthesis technique to obtain a three-dimensional (3D) neutron flux. These WCAP reports also mention the 3D neutron transport calculation code, TORT.
- The neutron fluence values of Waterford Unit 3 reactor vessel were calculated using a Westinghouse-developed code, RAPTOR-M3G similar to TORT.

Issue:

It is not clear whether the applicant's fluence method, which uses the RAPTOR-M3G code, has been incorporated into the current licensing basis including staff's review and approval.

Request:

1. Clarify whether the applicant's fluence method, which uses the RAPTOR-M3G code, has been incorporated into the current licensing basis.

2. If RAPTOR-M3G is not part of the current licensing basis:

   a. Provide justification for the use of the code.

   b. Clarify how the plant-specific dosimetry data of Waterford Unit 3 were used in measurement benchmarks to confirm the adequacy of use of the RAPTOR-M3G code for Waterford Unit 3 reactor vessel fluence calculations.

Waterford 3 Response

1. The use of RAPTOR-M3G for reactor vessel neutron fluence calculations has been incorporated into the WF3 current license basis by EC 68581. The guidance in NEI 96-07 Section 4.3.8.2 was used to determine that the transition to RAPTOR-M3G is not a departure from a method of evaluation, and was implemented via a 50.59 evaluation.

2. The response to parts 2a and 2b are not applicable based on response to part 1.

RAI 4.2.3-1 (Set 5)

Background:

LRA [License Renewal Application] Section 4.2.3 describes the applicant's time-limited aging analysis on pressurized thermal shock (PTS). During the audit, the staff noted that the following report describes more detailed information on the PTS analysis: WCAP-18002-NP, Revision 0, "Waterford Unit 3 Time-Limited Aging Analysis on Reactor Vessel Integrity," dated July 2015.

WCAP-18002-NP, Revision 0, indicates that the initial unirradiated reference temperature (called $RT_{NDT(U)}$ or initial RT_{NDT}) of lower shell plate M-1004-2 is updated from 22 °F to 0 °F. The WCAP report also indicates that this update is based on drop-weight and transverse-orientation Charpy V-notch test data per ASME Code Section III, NB-2300 in comparison with the previously determined value (22 °F) based on NRC Branch Technical Position (BTP) MTEB 5-2, which is comparable to the current BTP 5-3 in NUREG-0800, 2007.

The staff also noted that the previously determined initial RT_{NDT} value (22 °F) is described in Section 5 of WCAP-16088-NP, Revision 1, "Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation," dated September 2003 (ADAMS ML041620063).

Issue:

The LRA, including LRA Table 4.2-3, does not describe a specific provision of ASME Code Section III, NB-2331 that the applicant used in updating the initial RT_{NDT} of lower shell plate M-1004-2. In a similar manner, the staff noted that additional information is necessary to clarify whether the applicant's test data were adequately used in updating the initial RT_{NDT} values for the following beltline materials: (a) intermediate shell plates M-1003-1, M-1003-2 and M-1003-3; and (b) lower shell plates M-1004-1 and M-1004-3.

Request:

In order to demonstrate that the applicant's test data were adequately used in updating the initial RT_{NDT} of the beltline materials discussed above, describe the specific provision of ASME Code Section III, NB-2331 that the applicant used. As part of the response, provide the temperature (T_{Cv}) representing a minimum of 50 ft-lb absorbed energy and 35 mil lateral expansion as obtained in transverse-orientation Charpy V-notch tests for each material if such temperature was determined in the evaluation of material properties.

Waterford 3 Response

The initial RT_{NDT} values for the six Waterford Unit 3 beltline plates have been historically documented as shown in Table 2-1 of WCAP-16088-NP, Revision 2. Per Waterford Unit 3 FSAR, Revision 308, Table 5.3-13, these initial RT_{NDT} values were determined using Branch Technical Position (BTP) MTEB 5-2. The title and revision of the BTP series have changed over time. The older BTP MTEB 5-2 is comparable to the newer NUREG-0800, BTP 5-3. Therefore, the initial

RT_{NDT} and initial upper-shelf energy (USE) values for all of the beltline and extended beltline plate or forging materials were reevaluated. For those materials that already have documented values, the basis for those values was reconsidered, and new values were assigned as appropriate. Per Table 2-10 of WCAP-18002-NP, Revision 0, only the three cylindrical shell courses of plate material (3 upper shell plates, 3 intermediate shell plates, and 3 lower shell plates) reach fluence greater than or equal to 1×10^{17} n/cm^2 (E > 1.0 MeV) at 55 EFPY. Thus, reactor vessel nozzles are not considered herein.

Initial RT_{NDT} and initial USE values are defined for each of the nine reactor vessel plates. Although the historical values made use of BTP MTEB 5-2 and longitudinal test data, transverse data is available for the nine reactor vessel plates. The transverse data will be used per the provisions of ASME Code Section III, Subarticle NB-2331. For completeness, this response identifies the methodologies used for determination of the initial RT_{NDT} values for the extended beltline plate materials and the initial USE values, per ASTM E185-82 for all nine reactor vessel plates discussed herein.

Subarticle NB-2331 of Section III of the ASME Code Summer of 1972 Addenda to the 1971 Edition or later requires both drop-weight test data as well as Charpy V-notch test data from transverse specimens for determination of initial RT_{NDT} values.

ASME Code Section III, NB-2331, "Material for Vessels"

Pressure-retaining materials for vessels, other than bolting, shall be tested as follows:

(a) Establish a reference temperature RT_{NDT}; this shall be done as follows:

(1) Determine a temperature T_{NDT} that is at or above the nil-ductility transition temperature by drop weight tests.

(2) At a temperature not greater than T_{NDT} + 60°F (T_{NDT} + 33°C), each specimen of the C_v test (NB-2321.2) shall exhibit at least 35 mils (0.89 mm) lateral expansion and not less than 50 ft-lb (68 J) absorbed energy. Retesting in accordance with NB-2350 is permitted. When these requirements are met, T_{NDT} is the reference temperature RT_{NDT}.

(3) In the event that the requirements of (2) above are not met, conduct additional C_v tests in groups of three specimens (NB-2321.2) to determine the temperature T_{Cv} at which they are met. In this case the reference temperature RT_{NDT} = T_{Cv} – 60°F (T_{Cv} – 33°C). Thus, the reference temperature RT_{NDT} is the higher of T_{NDT} and [T_{Cv} – 60°F (T_{Cv} – 33°C)].

(4) When a C_v test has not been performed at T_{NDT} + 60°F (T_{NDT} + 33°C), or when the C_v test at T_{NDT} + 60°F (T_{NDT} + 33°C) does not exhibit a minimum of 50 ft-lb (68 J) and 35 mils (0.89 mm) lateral expansion, a temperature representing a minimum of 50 ft-lb (68 J) and 35 mils (0.89 mm) lateral expansion may be obtained from a full C_v impact curve developed from the minimum data points of all the C_v tests performed.

(b) Apply the procedures of NB-2331(a) to NB-2331(b)(1), (2), and (3):

(1) the base material;

(2) the base material, the heat affected zone, and weld metal from the weld procedure qualification tests in accordance with NB-4330;

(3) the weld metal of NB-2431.

The test data necessary to use these Code provisions for determination of the initial RT_{NDT} values are documented in the Waterford Unit 3 Certified Material Test Reports (CMTRs) or C-PENG-ER-004 for each of the nine Waterford Unit 3 reactor vessel cylindrical shell course plate materials (3 upper shell plates, 3 intermediate shell plates, and 3 lower shell plates). Using this test data, the initial RT_{NDT} values for these materials were determined directly from the data or by using a hyperbolic tangent curve fit through the minimum data points, in accordance with ASME Code Section III, Subarticle NB-2331, Paragraphs (a)(2) and (a)(4), respectively. The differences between the initial RT_{NDT} values summarized in the various Waterford Unit 3 analyses of record (e.g. WCAP-16088-NP, Revision 2) and the new values are due to use of the hyperbolic tangent curve-fitting method with the minimum data points as well as a closer look at all available data and testing information (e.g., transverse Charpy V-notch results).

In some cases, hyperbolic tangent curve fits are not needed since the available data is sufficient to determine the initial RT_{NDT} value. The orientation of the test specimen (transverse vs. longitudinal) is considered since it is vital in terms of the methodology used for initial RT_{NDT} determination. NUREG-0800, BTP 5-3, provides guidance for instances where the available data is not sufficient to meet ASME Code Section III, Subarticle NB-2331 provisions for determination of initial RT_{NDT}. However, for this reevaluation, NUREG-0800, BTP 5-3 is no longer required or used for the nine Waterford Unit 3 reactor vessel cylindrical shell course plates.

The initial RT_{NDT} values for Waterford Unit 3 that have been historically used for the beltline plate materials are documented in Table 2-1 of WCAP-16088-NP, Revision 2. The values from WCAP-16088-NP, Revision 2 are used in the prior reactor vessel integrity analyses of record and were considered part of the licensing basis for Waterford Unit 3. Table 1 below compares the prior licensing basis values with the updated values. As shown in Table 1, the initial RT_{NDT} values included in the original licensing basis are more conservative than the updated values for the three lower shell plates and are less conservative than the updated values for the three intermediate shell plates.

The new initial RT_{NDT} values for the six traditional beltline intermediate and lower shell plate materials supersede those that have been historically used. The initial RT_{NDT} values for the three extended beltline upper shell plate materials were used in reactor vessel integrity evaluations for the first time in WCAP-18002-NP, Revision 0. Thus, historical values are not listed in Table 1. In summary, the values utilized in the Waterford Unit 3 evaluation of the reactor vessel integrity TLAAs (WCAP-18002-NP, Revision 0) should now be considered the current licensing basis. Values for the nine reactor vessel plates, as documented in Table 1, are relevant to the 20-year license renewal activities for the plant. Table 2 shows the specific provision of ASME Code Section III, NB-2331 used and the temperature (T_{Cv}) representing a minimum of 50 ft-lb absorbed energy and 35 mil lateral expansion as obtained in transverse-orientation Charpy V-notch tests for each of the Waterford Unit 3 plate materials.

Table 1

Comparison of Waterford Unit 3 Reactor Vessel Plate Initial RT_{NDT} and Initial USE

Material Description	$RT_{NDT(U)}$ [(a)] (°F)	Updated $RT_{NDT(U)}$ (°F)	Initial USE [(a)] (ft-lb)	Updated[(b)] Initial USE (ft-lb)
Intermediate Shell Plate M-1003-1	-30	-25.1	94	108
Intermediate Shell Plate M-1003-2	-50	-20	97	132
Intermediate Shell Plate M-1003-3	-42	-20	90	111
Lower Shell Plate M-1004-1	-15	-37.6	106	135
Lower Shell Plate M-1004-2	22	0	141	141
Lower Shell Plate M-1004-3	-10	-20	94	118
Upper Shell Plate M-1002-1	- - -	-15.4	- - -	104
Upper Shell Plate M-1002-2	- - -	-1.4	- - -	95
Upper Shell Plate M-1002-3	- - -	-20	- - -	120

Notes for Table 1:

(a) Per WCAP-16088-NP, Rev. 2.

(b) Initial USE values for the six reactor vessel traditional beltline plate materials were updated in WCAP-17969-NP as part of the Capsule 83° analysis.

Table 2

Summary of Waterford Unit 3 Reactor Vessel Plate Initial RT_{NDT} Determination

Material Description	$RT_{NDT(U)}$ (°F)	T_{CV} (°F)	Charpy Limiting Parameter	T_{NDT} (°F)	Overall Limiting Parameter	ASME Section III, Subarticle NB-2331, Paragraph (a)(2), (a)(3) or (a)(4) Applicability
Intermediate Shell Plate M-1003-1	-25.1	34.9	Impact energy	-30	T_{CV} [(a)]	(a)(4)
Intermediate Shell Plate M-1003-2	-20	40	N/A [(c)]	-40	T_{CV} [(a)]	(a)(3)
Intermediate Shell Plate M-1003-3	-20	40	N/A [(c)]	-30	T_{CV} [(a)]	(a)(3)
Lower Shell Plate M-1004-1	-37.6	22.4	Impact energy	-40	T_{CV} [(a)]	(a)(4)
Lower Shell Plate M-1004-2	0	47.0	Impact energy	0	T_{NDT} [(b)]	(a)(3)
Lower Shell Plate M-1004-3	-20	10	N/A [(c)]	-20	T_{NDT} [(b)]	(a)(2)
Upper Shell Plate M-1002-1	-15.4	44.6	Impact energy	-20	T_{CV} [(a)]	(a)(4)
Upper Shell Plate M-1002-2	-1.4	58.6	Impact energy	-20	T_{CV} [(a)]	(a)(4)
Upper Shell Plate M-1002-3	-20	40	N/A [(c)]	-20	T_{NDT} [(b)]	(a)(2)

Notes for Table 2

(a) $RT_{NDT(U)} = T_{CV} – 60°F$ (Charpy Limited)

(b) $RT_{NDT(U)} = T_{NDT}$ (Drop Weight Limited)

(c) Both the impact energy (ft-lb) and the lateral expansion (mils) exceeded the minimum values of 50 ft-lb and 35 mils, respectively, at the same tested temperature

RAI 4.7.4-1 (Set 6)

Background:

LRA Section 4.7.4 provides the applicant TLAA for the aging evaluation of reactor vessel internals (RVI), other than those associated with applicant's metal fatigue TLAA for these components. The applicant identifies that the aging evaluations of irradiation-assisted stress corrosion cracking and loss of fracture toughness due to thermal aging and neutron irradiation embrittlement in its 2003 extended power uprate (EPU) license amendment request are analyses that conform to the definition of a TLAA in 10 CFR 54.3(a). The applicant stated that the implementation of LRA AMP B.1.33, Reactor Vessel Internals Program, will ensure that these TLAAs are acceptable in accordance with 10 CFR 54.21(c)(1))(iii).

The license amendment request for the EPU was submitted on November 3, 2003, and approved in an NRC-issued safety evaluation (SE) dated April 15, 2005 (ML051030068). Section 2.1.4 of the SE identifies that the projected neutron fluences for RVI components in the vicinity of the reactor core will range from 3.0 – 5.0 X 1022 n/cm2 (E > 0.1 MeV) through 40 years of licensed operations.

Issue:

EPRI Report MRP-191 estimates that RVI components in the core shroud would generally have neutron fluences ranging from 1.0 – 5.0 X 1022 n/cm2 through 60 years of licensed operations. The staff needs additional demonstration that the neutron fluence values for these types of RVI components through 60 years of licensed operation will not exceed the fluence estimates for the components in Table 4-7 of the MRP-191 report. Otherwise, the staff will need further assessment of the inspection bases for core shroud assembly components if the 60-year projected fluences for these components will exceed those specified for the components in MRP-191.

Request:

Justify (with a technical explanation) why the projected neutron fluences for RVI core shroud components through 60 years of operations are considered bounded by the fluence estimates for these components in Table 4-7 of the MRP-191 report. Otherwise, clarify what the impact will be on the FMECA assessment for these components and the inspection plan for RVI components if the 60-year neutron fluence value for any RVI core shroud component will exceed the neutron fluence estimate for the component in Table 4-7 of the MRP-191 report.

Waterford 3 Response

MRP-227-A (Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines) provides inspection and evaluation guidelines for managing aging effects in pressurized water reactor vessel internal components. Specifically, the guidelines are applicable to reactor vessel internal structural components, including core shroud components. MRP-191 describes the process and results of categorizing Westinghouse and Combustion Engineering (CE) designed pressurized water reactor (PWR) internals components according to age-related degradation and significance. Its results are a key element in developing the inspection and evaluation guidelines of MRP-227-A.

By letter dated December 16, 2013, as supplemented by letters dated January 19, June 18, July 9, 2015, Entergy submitted an aging management program for the reactor vessel internals at Waterford 3. MRP-227-A and its supporting reports were used as the technical bases for developing WF3's aging management program.

The fluence values for RVI components in EPRI Report MRP-191 are estimates. They are not bounding values above which the MRP-191 evaluations would be invalid.

EPRI issued letter MRP-2013-025 (MRP-227-A Applicability Template Guideline, October 14, 2013 (NRC ADAMS Accession No. ML 13322A454) to establish a range of conditions for which the MRP-227-A inspection and evaluation guidelines are applicable. For a CE-designed reactor, MRP-2013-025 identified that neutron fluence and heat generation rates are acceptable for applicability of the inspection and evaluation guidance of MRP-227-A if reactor parameters meet the following threshold values.

- Active fuel to fuel alignment plate (FAP) distance > 12.4 inches
- Average core power density < 110 Watts/cm^3
- Heat generation figure of merit, F ≤ 68 Watts/cm^3

These threshold values address the plant-specific applicability of MRP-227-A in the axial and the radial direction, which includes the core shroud component locations.

In 2015, Westinghouse evaluated the WF3 reactor against the above threshold values. As indicated in the NRC safety evaluation for the WF3 aging management program for reactor vessel internals, WF3 provided the plant-specific values of the heat generation rate, the maximum average core power density, and the FAP distance. The NRC staff reviewed these values and determined that they comply with the values in MRP-2013-025. Therefore, the staff concluded that the licensee satisfied the guidelines related to fuel management issues addressed in MRP-2013-025. As discussed in MRP-2013-025, those guidelines were established to ensure that plant-specific fluence levels remained within acceptable values to ensure continuing applicability of MRP-227-A. Therefore, there is no impact on the inspection plan for RVI components if the 60-year neutron fluence value for any RVI core shroud component exceeds the neutron fluence estimate for the component in Table 4-7 of the MRP-191 report.