Phyllis

From:	Clark, Phyllis
Sent:	Wednesday, October 12, 2016 6:09 AM
To:	'mchisum@entergy.com'
Cc:	'jjarrel@entergy.com'; 'dfrey@entergy.com'; 'lmurr91@entergy.com'; 'GYOUNG4 @entergy.com'; 'Dlach@entergy.com'; 'ataylo1@entergy.com'; 'aharris@entergy.com'; 'Milster, Leia Elizabeth'; Ramirez, Frances C; Speer, Chris; Diaz-Sanabria, Yoira K; Wittick, Brian; Morey, Dennis; RidsNrrDlr Resource; RidsNrrDlrRpb1 Resource; RidsNrrDlrRerb Resource; RidsNrrDlrRarb Resource; RidsNrrDlrRasb Resource; RidsNrrDlrRsrg Resource; RidsNrrPMWaterford Resource; 'RidsRgn4MailCenter@nrc.gov'; Folk, Kevin; Keegan, Elaine; Mitchell, Jeffrey; Buford, Angela; Prinaris, Andrew; Wong, Albert; Huynh, Alan; Akhavannik, Huda; Sweat, Tarico; Medoff, James; Gavula, James; Lopez, Juan; Cuadrado de Jesus, Samuel; Min, Seung K; Obadina, Sarah; Sadollah, Mohammad; Rogers, Billy; Brittner, Donald; Fu, Bart; Allik, Brian; Lehman, Bryce; Gardner, William; Thomas, George; Mink, Aaron; Doutt, Clifford; Holston, William; Yoo, Mark; Pulvirenti, April; McIntyre, David; Burnell, Scott; Moreno, Angel; Kennedy, Kriss; Scott, Catherine
Subject:	REF: WATERFORD STEAM ELECTRIC STATION, UNIT 3, LICENSE RENEWAL APPLICATION – RAIs SET 3 (TAC NO. MF7492)
Attachments:	Waterford 3 LRA Set 3 Enclosure (Final 60 Day Response Time) (10 4 2016)....docx

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

Mr. Michael R. Chisum
Site Vice President

SUBJECT: REQUESTS FOR ADDITIONAL INFORMATION FOR THE REVIEW OF THE WATERFORD STEAM ELECTRIC STATION, UNIT 3, LICENSE RENEWAL APPLICATION – SET 3 (TAC NO. MF7492)

Dear Mr. Chisum:

By letter dated March 23, 2016, Entergy Operations, Inc. submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, to renew the operating license NPF-38 for Waterford Steam Electric Station, Unit 3. The staff of the U.S. Nuclear Regulatory Commission (NRC or the staff) is reviewing the information contained in the license renewal application and has identified areas where additional information is needed to complete the review.

The enclosed requests for additional information were discussed with Mr. Alan Harris and a mutually agreeable date for the response is within 60 days from the date of this correspondence. Some RAIs from the draft version were moved to Sets 2 and 4 due to different response times requested. If you have any questions, please contact me at 301-415-6447 or by e-mail at Phyllis.Clark@nrc.gov.

Sincerely,

Phyllis Clark

Phyllis Clark, Project Manager
Projects Branch 1
Division of License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure:
As stated

cc: Listserv

ADAMS Accession No.: **ML16285A339** ***via email**

OFFICE	PM:RPB1:DLR	PM:RPB1:DLR	BC:RASB:DLR	BC:RARB:DLR	BC:RPB1:DLR	PM:RPB1:DLR
NAME	PClark	J Mitchell*	B Wittick*	D Morey*	YDiaz-Sanabria*	PClark
DATE	10/4/2016	10/6/2016	10/11/2016	10/6/2016	10/11/2016	10/11/2016

WATERFORD STEAM ELECTRIC STATION, UNIT 3
LICENSE RENEWAL APPLICATION
REQUESTS FOR ADDITIONAL INFORMATION – SET 3
(TAC NO. MF7492)

RAI B.1.16-2

Background:

Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-LR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL Report and when evaluation of the matter in the GALL Report applies to the plant.

LRA Section B.1.16 states that the Inservice Inspection - IWF Program, with enhancements, is consistent with GALL Report AMP XI.S3, “ASME Section XI, Subsection IWF.” The GALL Report AMP XI.S3, ASME Section XI, Subsection IWF states that the ASME Code, Section XI, Subsection IWF constitutes an existing mandated program applicable to managing aging of ASME Class 1, 2, 3 and MC component supports. The AMP “monitoring and trending” program element, states that examinations that reveal indications which exceed the acceptance standards and require corrective measures are extended to include additional examinations in accordance with IWF-2430. IWF-2420 states that to the extent practical, the same supports selected for examination during the first inspection interval shall be examined during each successive inspection interval. During its onsite audit, the staff reviewed operating experience and found:

1. For component supports and/or related hardware examined during IWF sampling inspections, degraded conditions were identified and, although conditions were “acceptable-as-is”, the components were re-worked/repaired to as-new condition or replaced. Since it was determined that the as-found condition did not affect the support’s capability to perform its design function or exceed the threshold of ASME Section IWF-3400, “Acceptance Criteria,” the applicant determined the actions associated with ASME Sections IWF-2420, “Successive Inspections,” and IWF-2430, “Additional Examinations,” criteria were not required and thus did not apply those code provisions.

2. Component supports and/or related hardware where degraded conditions were identified and re-worked/repaired or replaced as the result of walkdowns or other activities and not directly tied to an ISI-IWF inspection. However, the staff did not find evidence of an evaluation to determine whether the supports repaired were supports that were part of the IWF sample that is periodically inspected by the ISI-IWF program. Conferencing

with the applicant during the onsite audit indicated that such a process to identify whether repaired component supports are in the IWF inspection sample may not exist.

Issue:

Given that the program will manage aging of the entire ASME Code component support population through inspections of a representative population, any ISI-IWF sampled support that is re-worked to as-new condition or replaced would no longer be representative of other supports in the IWF component support population. Subsequent ISI-IWF inspections of the same sample may not represent the age-related degradation of the remaining population of supports that are not inspected. The applicant's LRA and associated basis documents do not provide a discussion of how this issue is addressed in the AMP, or if the current processes consider expansion or change of the ASME-based IWF sample if a component support and/or related hardware within the IWF sample were electively reworked or replaced. In addition, it is not clear whether a re-worked component support that is part of the ISI-IWF sample but not identified through the ASME ISI-IWF inspection, (but rather via a walkdown, other program, or some other means), would continue to be included in the ISI-IWF AMP program sample. As a result, it is not clear how the program will ensure that the ISI-IWF AMP component support inspection sample reflects the age-related degradation of the remaining population of IWF supports that are not inspected.

Request:

For situations in which a component that is inspected under the IWF sample is re-worked such that it no longer represents age-related degradation of the entire population, describe how the Inservice Inspection - IWF Program will continue to be effective in managing the aging effects of similar/adjacent components that are not included in the IWF inspection sample.

Other RAIs applicable to TRP 043 but addressed under another TRP:

RAI B.1.6-1: TRP 041 (Containment Inservice Inspection – IWE)
RAI B.1.6-2: TRP 041 (Containment Inservice Inspection – IWE)
RAI B.1.1-2: TRP 019 (Bolting Integrity)

RAI B.1.29-1

Background:

LRA Section B.1.29 states that the One-Time Inspection – Small-Bore Piping Program will be consistent with GALL Report AMP XI.M35. It also states that "…this program provides a one-time volumetric or opportunistic destructive inspection of a 3-percent sample or maximum of 10 ASME Class 1 piping butt weld locations and a 3-percent sample or a maximum of 10 ASME Class 1 socket weld locations that are susceptible to cracking."

Issue:

LRA Section B.1.29 does not provide the total population of welds for each weld type or the total number of these welds that will be included in the volumetric examinations.

Request:

Characterize the inspection sample size by completing the table below.

	Total Number of Welds at WF3	Total Number of Welds to Be Inspected
Class 1 Small-Bore Piping Butt Welds		
Class 1 Small-Bore Piping Socket Welds		

Revise the summary description LRA Section A.1.29 to specify (a) the weld population, and (b) the inspection sample size.

RAI B.1.30-3

Background:

LRA Table 3.3.2-7, “Emergency Diesel Generator System,” states that stainless steel expansion joints exposed to exhaust gas will be managed for cracking using the Periodic Surveillance and Preventive Maintenance program. The program description table in LRA Section B.1.30 states that the program inspection activity for the emergency generator system will be to “monitor the surface condition of the expansion joint to verify the absence of cracking.”

Issue:

It is unclear to the staff what inspection activities are included in monitoring the surface condition of the stainless steel expansion joints in order to verify the absence of cracking.

Request:

Provide details about the monitoring activities in the Periodic Surveillance and Preventive Maintenance program for the surface condition of the emergency diesel generator stainless steel expansion joints to verify the absence of cracking. Provide a discussion on any relevant parameters, such as, inspection technique/methodology, minimum detectable crack size, and allowable crack size based on the configuration. Also include any changes to the program description table and appropriate program elements to reflect any of these additional details.

RAI B.1.30-4

Background:

LRA Table 3.3.2-7, “Emergency Diesel Generator System,” states that stainless steel heat exchanger tubes externally exposed to lubricating oil, fuel oil, and treated water will be managed for loss of material due to wear using the Periodic Surveillance and Preventive Maintenance program. The program description table in LRA Section B.1.30 states that a visual inspection of the surface condition of a representative sample of stainless steel heat exchanger tubes will be performed to manage loss of material due to wear.

Issue:

Because access to the external surfaces of heat exchanger tubes is typically very limited, it is unclear to the staff whether a visual inspection of the tubes’ external surfaces can be reasonably expected to detect loss of material due to wear.

Request:

Justify the adequacy of the visual inspection in the Periodic Surveillance and Preventive Maintenance program to detect loss of material due to wear for emergency diesel generator cooler heat exchanger tubes. Provide information on any relevant parameters, such as, tube diameter, tube spacing, wear locations, access points for visual inspection.

RAI B.1.30-5

Background:

LRA Table 3.3.2-3, "Component Cooling and Auxiliary Component Cooling Water System," states that aluminum heat exchanger fins and carbon steel heat exchanger tubes exposed to condensation will be managed for reduction of heat transfer using the Periodic Surveillance and Preventive Maintenance program and loss of material using the External Surfaces Monitoring program.

The program description table in LRA Section B.1.30 states that visual or other NDE techniques will be used to inspect a representative sample of dry cooling tower radiator tubes and fins to manage loss of material and fouling that could result in a reduction of heat transfer capability.

Issue:

It is unclear to the staff if loss of material will be managed using the External Surfaces Monitoring or Periodic Surveillance and Preventive Maintenance program. If loss of material for the fins and tubes will be managed by the Periodic Inspection and Preventive Maintenance program, then it is unclear to the staff how a visual inspection can detect loss of material of carbon steel heat exchanger tubes due to limited visibility from fin attachment and tube spacing.

Request:

State if loss of material for the aluminum heat exchanger fins and carbon steel heat exchanger tubes will be managed using the External Surfaces Monitoring or the Periodic Surveillance and Preventive Maintenance program. If loss of material for the fins and tubes will be managed by the Periodic Surveillance and Preventive Maintenance program, justify the adequacy of a visual inspection to detect loss of material for the carbon steel heat exchanger tubes exposed to condensation. Provide information on any relevant parameters, such as, tube outside diameter, tube spacing, number of tube rows, fin outside diameter, fin spacing and attachment to the tube.

RAI B.1.34-2

Background:

The “scope of program” program element of GALL AMP XI.M31 states that the program includes all reactor vessel beltline materials as defined by 10 CFR 50, Appendix G, Section II.F. LRA Section 4.2.1 and Table 4.2-1 identify the Waterford Unit 3 reactor vessel beltline materials that are exposed to 60-year (55-EFPY) fluence greater than $1x10^{17}$ n/cm^2 (E > 1 MeV). Specifically, LRA Table 4.2-1 indicates that 55-EFPY fluence for the following materials at the clad/metal interface (0T) is $5.82x10^{17}$ n/cm^2 (E > 1 MeV): (a) upper shell plates M-1002-1, 2, and 3; (b) upper shell longitudinal welds 101-122A, B and C; and (c) upper to intermediate shell circumferential welds 106-121.

The 40-year (32-EFPY) fluence for these upper shell plates and associated welds is approximately estimated as $3.37x10^{17}$ n/cm^2 (E > 1 MeV) by linear interpolation. The fluence estimate suggests that these upper shell materials are also identified as beltline materials for 32 EFPY.

Issue:

The upper shell materials discussed in the background section are not identified as beltline materials in the evaluation for the 32-EFPY pressure-temperature (P-T) limits described in WCAP-16088-NP, Revision 1, “Waterford Unit 3 Reactor Vessel Heatup and Cooldown Limit Curves for Normal Operation,” September 2003 (ADAMS ML041620063). Specifically, Table 2-2, “Summary of the Waterford Unit 3 Reactor Vessel Beltline Material Chemistry Factors,” in WCAP-16088-NP, Revision 1 does not identify these upper shell materials as reactor vessel beltline materials.

Identification of the upper shell materials as beltline materials between the 32-EFPY evaluation and 55-EFPY evaluation is unclear. In addition, consistent consideration of these upper shell materials as beltline materials in the 32-EFPY to 55-EFPY P-T limits should be assured.

Request:

Reconcile inconsistency in identifying the upper shell plates and welds as reactor vessel beltline materials between the 32-EFPY evaluation and 55-EFPY evaluation discussed in the issue section of this RAI.

RAI B.1.36-4

Background:

LRA Section B.1.36, "Service Water Integrity," states that this program manages components as described in the Waterford 3 response to NRC Generic Letter 89-13 ["Service Water System Problems Affecting Safety-Related Equipment"]. Waterford's response to Generic Letter 89-13, dated January 29, 1990, for Action III (associated with establishing routine inspection of the open-cycle service water system to ensure that corrosion, erosion, silting, and biofouling cannot degrade the performance of systems supplied by service water), states:

> The LP&L [the prior licensee] erosion/corrosion program for Waterford was thoroughly presented in its supplemental response to Generic Letter 89-08 (re W3P89-1592, dated November 17, 1989). Components from auxiliary component cooling water (ACCW) – the Waterford safety-related "open" service water system – will be added to that program before the start of the next refueling outage.

During its recent audit, the staff noted that condition report CR-WF3-2009-00614 and -00852 (which refer to an earlier occurrence in condition report CR-WF3-1997-1316), discuss cavitation erosion damage in piping spool pieces and in valves ACC-126A and -126B. The staff notes that the ACCW system is excluded in current system susceptibility evaluation for the Flow-Accelerated Corrosion program. In addition, although aging management program evaluation report WF3-EP-14-00007, "Aging Management Program Evaluation Report Non-Class 1 Mechanical", Section 4.12, "Service Water Integrity," cites Entergy Nuclear Management Manual procedure EN-DC-340, "Microbiologically Influenced Corrosion (MIC) Monitoring Program," for implementation of some aspects in several program elements, it does not cite Entergy Nuclear Management Manual procedure EN-DC-315, "Flow-Accelerated Corrosion," for managing erosion issues associated with Action III of Generic Letter 89-13.

Issue:

The staff notes that Generic Letter 89-08, "Erosion/Corrosion-Induced Pipe Wall Thinning," (cited in the letter dated January 29, 1990), established the Flow-Accelerated Corrosion program, as discussed in the GALL Report AMP XI.M17. Because components from the auxiliary component cooling water system are currently not included in the implementation of the program associated with Generic Letter 89-08 (i.e., Flow-Accelerated Corrosion program), it is unclear to the staff what current Service Water Integrity program activities are associated with Generic Letter 89-13, Action III.

In addition, although there have been historical issues with cavitation erosion in portions of the auxiliary component cooling water system, it is unclear to the staff what existing program currently manages the associated loss of material due to erosion. The Service Water Integrity program shows the interrelationship with Entergy Nuclear Management Manual procedure EN-DC-340; however, that program does not appear to include erosion mechanisms. Also the current system susceptibility evaluation for the Flow-Accelerated Corrosion program does not include the auxiliary component cooling water system. The current associated Entergy Nuclear Management Manual Service Water Integrity program procedure EN-DC-184, "NRC Generic Letter 89-13 Service Water Program," Attachment 9.2 [3] includes the "Service Water System Piping / Component Inspection and Maintenance Program Element" that corresponds to Generic Letter 89-13, Action III. However, Attachment 9.1 "Procedural Interrelationships" currently does not show any connection with the Flow-Accelerated Corrosion program, as

implied by Waterford's original Generic Letter 89-13 response. In addition, there is no proposed enhancement to revise the Service Water Integrity program procedures to credit the activities in the Flow-Accelerated Corrosion program as accomplishing portions of Generic Letter 89-13, Action III.

<u>Request</u>:

As it relates to statement in LRA Section B.1.36 regarding managing components "as described in the [Waterford 3] response to NRC Generic Letter 89-13,"

a) Describe what current Service Water Integrity program activities are associated with Waterford's previous commitment for Generic Letter 89-13, Action III.
    i) Include a discussion of how the existing program manages the loss of material due to erosion in portions of the ACCW system near valves ACC-126A and ACC-126B.
    ii) Include a discussion of how the "service water piping inspection program," as required under Entergy Nuclear Management Manual procedure EN-DC-184, Attachment 9.2[3](4) is currently implemented.
b) Clarify whether the planned enhancement to the Flow-Accelerated Corrosion will be credited as part of the Service Water Integrity program by accomplishing portions of Generic Letter 89-13, Action III. Either show that the Service Water Integrity program currently credits the Flow-Accelerated Corrosion program, or provide the bases for why an enhancement to the Service Water Integrity program is not needed.

RAI B.1.36-5

<u>Background</u>:

Applicant operating experience shows blockage due to sediment for a drain line in the ACCW system. Based on Entergy Nuclear Management Manual procedure EN-DC-184, Waterford's Generic Letter 89-13 service water program requires flushing and flow testing to ensure flow blockages do not form within infrequently used flow paths. However, LRA Section B.1.36 includes an enhancement to revise the Service Water Integrity program procedures to flush redundant, infrequently flowed, and stagnant lines to ensure there is no blockage and to inspect selected system low points such as drains.

<u>Issue</u>:

Although the ACCW system includes corrosion inhibitors and uses demineralized water for make-up, sufficient fouling to block drain lines existed within the system. It is unclear to the staff whether Waterford enhanced the Service Water Integrity program after finding the drain line blockage, in light of the enhancement described in the LRA.

<u>Request:</u>

Explain what changes were made to the Service Water Integrity program as a result of condition report CR-WF3-2009-00843 regarding the adequacy of flushing. If the current program includes the associated flushing program element from Entergy Nuclear Management Manual procedure EN-DC-184, clarify what portions of the Service Water Integrity Program procedures need to be revised for the enhancement.

RAI B.1.38-2

Background:

Section 54.21(a)(3) of 10 CFR requires the applicant to demonstrate that the effects of aging for structures and components will be adequately managed so that the intended function(s) will be maintained consistent with the current licensing basis for the period of extended operation. As described in SRP-LR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL Report and when evaluation of the matter in the GALL Report applies to the plant. The SRP-LR also states that if an applicant takes credit for a program in the GALL Report, it is incumbent on the applicant to ensure that the conditions and operating experience at the plant is bounded by the conditions and operating experience for which the GALL Report program was evaluated.

LRA Section B.1.38, "Structures Monitoring," states that the Structures Monitoring Program is an existing program, with enhancements, that will be consistent with GALL Report AMP XI.S6. The "operating experience" program element in LRA Section B.1.38 and the Structures Monitoring Section in WF3-EP-14-00003, "Operating Experience Review Results – Aging Management Program Effectiveness," discuss plant-specific operating experience associated with concrete cracking, surface rust, deficiencies in sealant, and exposed steel reinforcement at several WF3 structures, and concludes that the program provides reasonable assurance that the effect of aging will be managed such that components will continue to perform their intended functions consistent with the current licensing basis through the period of extended operation.

During the AMP audit the staff reviewed condition reports CR-WF3-2006-00755, CR-WF3-2009-06945, and CR-WF3-2010-05582, that documented corrective actions to address several plant-specific operating experiences associated with corrosion on structural steel, supports and components. The staff also reviewed condition report CR-WF3-2015-00947 that assessed the history (since January 2010) of several conditions reports associated with the keyword "corrosion." The corrective actions associated with this condition report resulted in further inspections of areas susceptible to external corrosion and the development of an external corrosion and coating procedure (UNT-006-032) for safety related systems and components. During the walkdowns, the staff also observed corroded bolts/nuts from a structural steel column, exposed concrete rebars, and structural steel with different levels of corrosion in several structures located outdoors (e.g., CR-WF3-2016-4481, CR-WF3-2016-4482).

Issue:

Based on the staff review of this plant-specific operating experience and staff observed conditions during the audit walkdowns for outdoor structures, it is not clear to the staff (1) how the structures monitoring program captures the operating experience (i.e. the existing corrosion concerns from recent inspections) and whether the conditions and operating experience at the plant is bounded by the conditions and operating experience for which the GALL Report program was evaluated in Section XI.S6, and (2) whether and how the Structures Monitoring Program specified inspection frequency of 5 years remains adequate, considering the recent operating experience, to ensure no loss of intended functions during the period of extended operation for those structures with ongoing exterior corrosion concerns.

Request:

1. Clarify the basis for the Structures Monitoring Program specified inspection frequency of 5 years, considering the most recently presented operating experience, to ensure no loss of intended function during the period of extended operation. Response should include, as a minimum:
    a. a summary of the plant-specific operating experience associated with existing corrosion concerns from recent inspections of structures and structural supports in scope of license renewal, including the actions taken to address and disposition this operating experience, and
    b. how the structures monitoring program captures this operating experience (i.e. the existing corrosion concerns) to ensure that the conditions and operating experience at the plant is bounded by the conditions and operating experience for which the GALL Report program was evaluated.
2. Update the LRA and FSAR supplement, as appropriate, to be consistent with the response to the above requests.

RAI 3.1.1.74-2:

Background:

LRA item 3.1.1-74 addresses aging of steam generator upper assembly and separators. During its review of LRA item 3.1.1-74, the staff noted an applicant inspection report for steam generator components which indicated that feedwater pipe vibrations were observed during Cycle 19: "180 Day Steam Generator Tube Inspection Report for the 19th Refueling Outage Waterford Steam Electric Station, Unit 3," November 06, 2014, page 10 (ADAMS No. ML14314A032).

The applicant inspection report also indicated that, due to observed feedwater pipe vibrations, the applicant performed a visual inspection of feedring structural supports during which no anomalies were noted. The observed vibrations can cause loss of material due to wear in the feedwater ring and supports within the steam generator.

Issue:

The LRA does not clearly address which aging management review (AMR) items are used to manage loss of material due to wear for steam generator feedwater ring and supports. In addition, the LRA does not address applicant's evaluations or actions to minimize the feedwater piping vibrations for aging management.

Request:

1. Clarify whether loss of material due to wear is an aging effect requiring management for steam generator feedwater ring and supports. If so, describe AMR items which are used to manage loss of material for these components, including aging management programs.

2. Discuss applicant's evaluations or actions to minimize feedwater piping vibrations that can promote aging degradation of feedwater piping components within the steam generator.

RAI 4.2.1-1

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.

RAI 4.3.1-2

Background:

LRA Sections 4.7.2 and 4.7.3 describe the TLAAs associated with the Leak-Before-Break Analysis and Postulation of High Energy Line Break (HELB) Locations, respectively. The LRA states that transient cycles were analyzed for both of the TLAAs and that the transients will be monitored using the Fatigue Monitoring Program. The applicant dispositioned these TLAAs in accordance with 10 CFR 54.21(c)(1)(iii).

Issue:

LRA Sections 4.7.2 and 4.7.3 did not clarify which transient will be monitored for these TLAAs. The staff is unclear if these transients are within the scope of the Fatigue Monitoring Program.

Request:

For both the Leak-Before-Break Analysis TLAA and Postulation of High Energy Line Break (HELB) Locations TLAA:

a) Identify which transients were used in the analyses.

b) Confirm that these transients will be monitored under the Fatigue Monitoring Program.

RAI 4.3.3-2

Background:

LRA Section 4.3.3 discusses the applicant's evaluation of the effects of the reactor water environment on fatigue life. The LRA states that design basis ASME Code Class 1 component fatigue evaluations will be reviewed to ensure that the most limiting components within the reactor coolant pressure boundary will be included in its environmental fatigue evaluations.

Issue:

The staff lacks sufficient information on how the applicant will identify and evaluate the plant-specific locations that may be more limiting than the locations identified in NUREG/CR-6260. It is unclear to the staff what methodology the applicant will use to ensure that the most limiting locations will be evaluated for environmental effects such that the effects of environmentally affected fatigue will be managed throughout the period of extended operation.

Request:

a) Provide the methodology that will be used to identify plant-specific component locations in the reactor coolant pressure boundary that may be more limiting than the components identified in NUREG/CR-6260.

b) Justify that each step of the methodology used to identify the plant-specific limiting locations is bounding and representative of the plant.

RAI 4.3.3-3

Background:

LRA Section 4.3.3 discusses the applicant's evaluation of the effects of the reactor water environment on fatigue life. The LRA states that the environmental effects on fatigue for a set of critical components will be evaluated using NUREG/CR-6909, "Effect of LWR Coolant Environments on the Fatigue Life of Reactor Materials."

Appendix A of NUREG/CR-6909 states the following:

> *For the case of a constant strain rate and a linear temperature response, an average temperature (i.e., average of the maximum and minimum temperatures for the transients) may be used to calculate F_{en}. In general, the "average" temperature that should be used in the calculations should produce results that are consistent with the results that would be obtained using the modified rate approach described in Section 4.2.14 of this report. The maximum temperature can be used to perform the most conservative evaluation.*

The method used to calculate the "average temperature" is dependent on whether the minimum transient temperature exceeds the temperature threshold value of the material. When the minimum temperature exceeds the threshold temperature, the maximum and minimum temperature values of the stress cycle or load set pair are used to calculate the average

temperature. When the minimum temperature is below the threshold temperature, the maximum and threshold temperature are used to calculate the average temperature. Sections 4.2.4 and 5.2.7 of NUREG/CR-6909 provide examples of determining average temperatures.

As noted above, NUREG/CR-6909 also states that the average temperature may be used to calculate the environmentally assisted fatigue correction factor (F_{en}) values for transients with a constant strain rate and a linear temperature response, which are defined as “simple” transients. Use of an average temperature may not be appropriate for more complex transients that have multiple or non-linear temperature variations. For complex transients, the modified rate approach should be used to validate F_{en} calculations.

Issue:

If the applicant will use NUREG/CR-6909 to calculate the F_{en} values, the staff needs confirmation that the average transient temperatures were calculated appropriately considering the threshold temperatures. The staff also needs confirmation that the average temperatures were limited to simple transients.

Request:

a) Identify all locations that used NUREG/CR-6909 AND the average temperature to calculate the F_{en} value. For each location, provide the following:

    i. The material of construction.
    ii. A description of how the average temperature was calculated.
    iii. A description of all transients associated with the use of average temperatures and demonstration that the transients are simple transients.

b) Justify that the guidelines of NUREG/CR-6909 were followed when calculating the F_{en} and environmentally-adjusted cumulative usage factors (CUF_{en}) and will continue to be followed for future F_{en} and CUF_{en} calculations.