NRR-DMPSPEm Resource

From:	Wong, Albert
Sent:	Thursday, February 08, 2018 4:58 PM
To:	wmagui1@entergy.com
Cc:	RidsNrrDmlr Resource; RidsNrrDmlrMrpb Resource; RidsNrrPMRiverBend Resource; RidsOgcMailCenter Resource; Wilson, George; Donoghue, Joseph; Sayoc, Emmanuel; Wong, Albert; Allik, Brian; Nold, David; Medoff, James; Patel, Amrit; Mink, Aaron; Oesterle, Eric; Alley, David; Martinez Navedo, Tania; Bailey, Stewart; Wittick, Brian; Ruffin, Steve; Bloom, Steven; Regner, Lisa; Turk, Sherwin; Sowa, Jeffrey; Parks, Brian; Pick, Greg; Kozal, Jason; Young, Cale; Young, Matt; Werner, Greg; McIntyre, David; Dricks, Victor; Moreno, Angel; Burnell, Scott; 'Broussard, Thomas Ray'; Lach, David J; SCHENK, TIMOTHY A; 'Coates, Alyson'
Subject:	FINAL REQUESTS FOR ADDITIONAL INFORMATION FOR THE SAFETY REVIEW OF THE RIVER BEND STATION LICENSE RENEWAL APPLICATION (CAC NO. MF9757) – SET 10
Attachments:	RAIs Set 10 Enclosure CLEAN Final_16RAIs_020818.pdf; RAI Set 10 email ATTACHMENT 1 CLEAN 16RAIs_020818.pdf

Docket No. 50-458

Dear Mr. Maguire:

By letter dated May 25, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17153A282), Entergy Operations, Inc. (the applicant) submitted an application pursuant to Title 10 of the *Code of Federal Regulations* Part 54, “Requirements for Renewal of Operating Licenses for Nuclear Power Plants,” to renew the operating license NPF-47 for River Bend Station.

On December 19 and 20, 2017 and January 2 and 16, 2018, the U.S Nuclear Regulatory Commission (NRC) staff sent Entergy Operations, Inc. the draft Requests for Additional Information (RAIs) for various technical review packages (TRP). Entergy Operations, Inc. subsequently informed the NRC staff that clarification calls were needed to discuss the information requested. The specific dates of the RAIs clarification calls and the actions taken are summarized in Attachment 1. The final RAIs are enclosed.

David Lach of your staff agreed to provide a response to all the final RAIs within 30 days of the date of this email. The NRC staff will be placing a copy of this email in the NRC’s Agencywide Documents Access and Management System.

Sincerely,

Emmanuel Sayoc, Project Manager *Albert Wong for*
License Renewal Projects Branch (MRPB)
Division of Materials and License Renewal
Office of Nuclear Reactor Regulation

Docket No. 50-458

Enclosure:
As stated

OFFICE	PM:MRPB:DMLR	BC: MRPB:DMLR	PM: MRPB:DMLR
NAME	ESayoc	EOesterle	ESayoc *Albert Wong for*
DATE	02/01/2018	02/07/2018	02/08/2018

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Subject: FINAL REQUESTS FOR ADDITIONAL INFORMATION FOR THE SAFETY REVIEW OF THE RIVER BEND STATION LICENSE RENEWAL APPLICATION (CAC NO. MF9757) – SET 10
Sent Date: 2/8/2018 4:58:11 PM
Received Date: 2/8/2018 4:58:00 PM
From: Wong, Albert

Created By: Albert.Wong@nrc.gov

Recipients:
"RidsNrrDmlr Resource" <RidsNrrDmlr.Resource@nrc.gov>
Tracking Status: None
"RidsNrrDmlrMrpb Resource" <RidsNrrDmlrMrpb.Resource@nrc.gov>
Tracking Status: None
"RidsNrrPMRiverBend Resource" <RidsNrrPMRiverBend.Resource@nrc.gov>
Tracking Status: None
"RidsOgcMailCenter Resource" <RidsOgcMailCenter.Resource@nrc.gov>
Tracking Status: None
"Wilson, George" <George.Wilson@nrc.gov>
Tracking Status: None
"Donoghue, Joseph" <Joseph.Donoghue@nrc.gov>
Tracking Status: None
"Sayoc, Emmanuel" <Emmanuel.Sayoc@nrc.gov>
Tracking Status: None
"Wong, Albert" <Albert.Wong@nrc.gov>
Tracking Status: None
"Allik, Brian" <Brian.Allik@nrc.gov>
Tracking Status: None
"Nold, David" <David.Nold@nrc.gov>
Tracking Status: None
"Medoff, James" <James.Medoff@nrc.gov>
Tracking Status: None
"Patel, Amrit" <Amrit.Patel@nrc.gov>
Tracking Status: None
"Mink, Aaron" <Aaron.Mink@nrc.gov>
Tracking Status: None
"Oesterle, Eric" <Eric.Oesterle@nrc.gov>
Tracking Status: None
"Alley, David" <David.Alley@nrc.gov>
Tracking Status: None
"Martinez Navedo, Tania" <Tania.MartinezNavedo@nrc.gov>
Tracking Status: None
"Bailey, Stewart" <Stewart.Bailey@nrc.gov>
Tracking Status: None
"Wittick, Brian" <Brian.Wittick@nrc.gov>
Tracking Status: None
"Ruffin, Steve" <Steve.Ruffin@nrc.gov>
Tracking Status: None
"Bloom, Steven" <Steven.Bloom@nrc.gov>

Tracking Status: None
"Regner, Lisa" <Lisa.Regner@nrc.gov>
Tracking Status: None
"Turk, Sherwin" <Sherwin.Turk@nrc.gov>
Tracking Status: None
"Sowa, Jeffrey" <Jeffrey.Sowa@nrc.gov>
Tracking Status: None
"Parks, Brian" <Brian.Parks@nrc.gov>
Tracking Status: None
"Pick, Greg" <Greg.Pick@nrc.gov>
Tracking Status: None
"Kozal, Jason" <Jason.Kozal@nrc.gov>
Tracking Status: None
"Young, Cale" <Cale.Young@nrc.gov>
Tracking Status: None
"Young, Matt" <Matt.Young@nrc.gov>
Tracking Status: None
"Werner, Greg" <Greg.Werner@nrc.gov>
Tracking Status: None
"McIntyre, David" <David.McIntyre@nrc.gov>
Tracking Status: None
"Dricks, Victor" <Victor.Dricks@nrc.gov>
Tracking Status: None
"Moreno, Angel" <Angel.Moreno@nrc.gov>
Tracking Status: None
"Burnell, Scott" <Scott.Burnell@nrc.gov>
Tracking Status: None
"'Broussard, Thomas Ray'" <tbrous1@entergy.com>
Tracking Status: None
"Lach, David J" <DLach@entergy.com>
Tracking Status: None
"SCHENK, TIMOTHY A" <TSCHENK@entergy.com>
Tracking Status: None
"'Coates, Alyson'" <alacey@entergy.com>
Tracking Status: None
"wmagui1@entergy.com" <wmagui1@entergy.com>
Tracking Status: None

Post Office:

Files	**Size**	**Date & Time**
MESSAGE	1608	2/8/2018 4:58:00 PM
RAIs Set 10 Enclosure CLEAN Final_16RAIs_020818.pdf		226003
RAI Set 10 email ATTACHMENT 1 CLEAN 16RAIs_020818.pdf		101234

Options
Priority: Standard
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REQUEST FOR ADDITIONAL INFORMATION
LICENSE RENEWAL APPLICATION
RIVER BEND STATION, UNIT 1
DOCKET NO.: 50-458
CAC NO.: MF9757
Office of Nuclear Reactor Regulation
Division of Materials and License Renewal

10 CFR § 54.21(a)(3) of 10 CFR requires an 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. One of the findings that the staff must make to issue a renewed license (10 CFR § 54.29(a)) is that actions have been identified and have been or will be taken with respect to the managing the effects of aging during the period of extended operation on the functionality of structures and components that have been identified to require review under § 54.21, such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the current licensing basis (CLB). As described in SRP LR, an applicant may demonstrate compliance with 10 CFR 54.21(a)(3) by referencing the GALL Report. In order to complete its review and enable making a finding under 10 CFR § 54.29(a), the staff requires additional information in regard to the matters described below.

RAI 3.1.2.1.2-1 (Materials, Environments, Aging Effects Requiring Management, and Aging Management Programs)

Background

The regulation in 10 CFR 54.4(a)(3) requires applicants to include structures, systems, or components (SSCs) in the scope of their license renewal applications (LRAs) if the SSCs are within the scope of specific regulations, including those SSCs that are subject to the regulation in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without SCRAM Events for Light-Water Cooled Nuclear Power Plants."

For those SSCs that are scoped into an LRA in accordance with 10 CFR 54.4, the regulation in 10 CFR 54.21(a)(1) requires the applicant to subject the SSCs to an aging management review (AMR) if the SSCs are: (a) are not active or do not involve changes in configuration or moving parts, and (b) are not subject to replacement based on a qualified life or specified time period. For those SSCs that are scoped into an LRA and are required to be subjected to an AMR, the regulation in 10 CFR 54.21(a)(3) 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.

In AMR item #43 of Table 3.1-1 in the NUREG-1800, Revision 2 Report (SRP-LR) and in AMR Items IV.B1.RP-28 of NUREG-1801, Revision 2 (GALL), the staff identifies that the combined programs in GALL AMP XI.M1, "ASME Section XI, Inservice Inspection, Subsections, IWB, IWC, and IWD," and GALL AMP XI.M2, "Water Chemistry," may be used as an acceptable combination of AMPs to manage loss of material in stainless steel or nickel alloy BWR reactor vessel internal (RVI) components as a result of a pitting or crevice corrosion mechanism. In LRA AMR Item 3.1.1-43 and in the Table 2 AMR items in Table 3.1.2-2 that are linked to AMR item 3.1.1-43, the applicant proposes to use the One-Time Inspection Program (OTI) Program, LRA AMP B.1.32) as an NEI Generic Note E alternative condition program for managing loss of material due to pitting or crevice corrosion in the RVI components in lieu of the applicant's Inservice Inspection Program (LRA AMP B.1.22).

Issue

The staff has not specified in the GALL report that a combination of OTI and Water Chemistry AMPs is an acceptable basis for managing loss of material due to pitting or crevice corrosion or other corrosion-based aging effects (e.g., cracking induced by any of the stress corrosion cracking mechanisms) in BWR RVI components. Instead, the staff has always specified in GALL Table IV.B1 that a periodic condition monitoring program (e.g., the ISI Program or Vessel Internals Program) should be used in conjunction with the Water Chemistry Program to manage corrosion-based aging effects in BWR RVI components, where the implementation of the periodic condition monitoring program would be used to confirm the effectiveness of the Water Chemistry Program in managing the effects.

Request

For reactor internals, justify why the OTI Program is considered to be an acceptable basis for verifying the effectiveness of the Water Chemistry Control – BWR Program in managing loss of material due to pitting or crevice corrosion in lieu of using either LRA AMP B.1.22, "Inservice Inspection Program," or LRA AMP B.1.10, "BWR Vessel Internals Program," for this aging management objective.

RAI 3.1.2.1.2-2 (Materials, Environments, Aging Effects Requiring Management, and Aging Management Programs)

Background

The regulation in 10 CFR 54.4(a)(3) requires applicants to include structures, systems, or components (SSCs) in the scope of their license renewal applications (LRAs) if the SSCs are within the scope of specific regulations, including those SSCs that are subject to the regulation in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without SCRAM Events for Light-Water Cooled Nuclear Power Plants."

For those SSCs that are scoped into an LRA in accordance with 10 CFR 54.4, the regulation in 10 CFR 54.21(a)(1) requires the applicant to subject the SSCs to an aging management review (AMR) if the SSCs are: (a) are not active or do not involve changes in configuration or moving parts, and (b) are not subject to replacement based on a qualified life or specified time period. For those SSCs that are scoped into an LRA and are required to be subjected to an AMR, the regulation in 10 CFR 54.21(a)(3) 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.

In LRA Table 3.1.1, AMR item 3.1.1-103, the applicant addresses the AMPs that will be used to manage cracking in stainless steel incore instrument flux monitoring dry tubes (ICI dry tubes) that are exposed to a treated water > 140 ºF environment. In the basis for the AMR and the related AMR item in LRA Table 3.1.2-2, the applicant cites use of NEI Generic Note E and identifies that it is replacing the BWR Vessel Internals Program (LRA AMP B.1.10) with the Inservice Inspection (ISI) Program (LRA AMP B.1.22) to manage cracking of the tubes.

Issue

In Section 4.6 of Aging Management Program Evaluation Report RBS-EP-15-00006, Revision 0, the applicant indicates the incore instrumentation (ICI) dry tubes at the plant are within the scope of the applicant's BWR Vessel Internals Program and are inspected in accordance the inspection and evaluations guidelines in EPRI Report No. BWRVIP-47-A. The program evaluation report also indicates that the ICI dry tubes are being replaced in accordance with the additional guidelines in GE-Hitachi Services Information Letter (SIL) 409, Revision 3, with the final replacements of the dry tubes to be completed in year 2019.

Request

Clarify whether the ICI dry tube replacement activities are within the scope of the condition monitoring AMP that will be used to manage cracking in the tubes and, if so, whether the replacement activities are considered to be an enhancement of the program. Additionally, justify why LRA AMR Item 3.1.1-103 proposes to use the ISI Program for management of cracking in the ICI dry tubes when the BWR Vessel Internals Program (as the existing AMP referenced for reactor internals in the LRA) already implements inspections of the dry tubes using the methodology in EPRI Report BWRVIP-47-A.

RAI 4.2.1-1 (Reactor Vessel Fluence)

Background

The regulation in 10 CFR 54.21(c)(1)(ii) states that, for a specific time limited aging analyses (TLAA) that is dispositioned in accordance with this regulation, the applicant must demonstrate that the analysis has been projected to the end of the period of extended operation.

License renewal application (LRA) Section 4.2.1, Reactor Vessel Neutron Fluence," identifies the neutron fluence analysis in the current licensing basis as a TLAA for the LRA. Specific fracture toughness requirements for normal operation and for anticipated operational occurrences for power reactors are set forth in 10 CFR Part 50, Appendix G, "Fracture Toughness Requirements." The requirements of Appendix G are imposed by 10 CFR 50.60, "Acceptance Criteria for Fracture Prevention Measures for Light Water Nuclear Power Reactors for Normal Operation." To satisfy the requirements of Appendix G, methods for determining the fast neutron fluence (E > 1.0 MeV) are necessary to estimate the fracture toughness of the pressure vessel materials.

Regulatory Guide (RG) 1.190, "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence," describes methods and assumptions acceptable to the NRC staff for determining pressure vessel fluence. This RG is intended to ensure the accuracy and reliability of the fluence determination required by General Design Criteria 14, 30, and 31 of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50.

LRA Section 4.2.1 states: "Neutron fluence for the welds, shells and nozzles of the [reactor pressure vessel (RPV)] beltline region was determined using the General Electric Hitachi (GEH) method for neutron flux calculation documented in report NEDC-32983P-A (Ref. 4-20) and approved by the NRC." The applicant states that the fluence methodology is adherent to the staff's guidance in RG 1.190.

Issue

The water density sensitivity analyses described in Section 7.1, “Calculation Uncertainties,” of NEDC-32983P-A, Revision 2, “General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations,” are not applicable to the above core water density distribution because the generically approved calculational model does not extend above the active fuel region based on Figure 2-2, “Schematic View of (r,z) Model” in NEDC-32983P-A, Revision 2. The uncertainty in the above core water density distribution, for example, will have an impact on the overall fluence calculation uncertainty for RPV beltline components above the active fuel region (e.g., the BWR/6 low pressure coolant injection nozzle forging/weld considered in the development of pressure-temperature limits).

Also, the applicant’s existing plant records include a condition report that identifies that the current calculational methodology in GEH Report No. NEDC-32983P-A may include one or more errors in the calculational methods for calculating RPV neutron fluxes or projecting RPV neutron fluence values.

Request

1. Provide a detailed description of the above core calculational model and how this expanded model has been qualified. Describe how analytic uncertainty data, measurement benchmarking data, and calculational benchmarking data relevant to the RPV extended beltline has been incorporated into the combined uncertainty analysis applicable to the RPV extended beltline to support the qualification of an expanded model.
2. If the combined uncertainty analysis is greater than 20%, explain how the neutron fluence values used in downstream safety analyses have been augmented to account for any uncertainties associated with the calculation methods, nuclear data, or modeling accuracies of the analyses.
3. Explain how the identified error(s) included in GEH Report No. NEDC-32983P-A affect the information provided in the LRA.

RAI 4.2.6-1 (Reactor Vessel Axial Weld Failure Probability)

Background
In LRA Section 4.2.6, the applicant provides its TLAA related to the “Reactor Vessel Axial Weld Failure Probability” analysis. As part of this TLAA evaluation, the applicant provides its updated mean RT_{NDT} values for the components in LRA Table 4.2-6. The applicant reports that the limiting axial welds for the TLAA are those associated with the RPV lower-intermediate shell axial welds that were fabricated from Weld Heat No. 5P6756 and that the mean RT_{NDT} value for the welds is projected to be 65.8 ºF at 54 EPFY. The applicant’s identifies that the projected mean RT_{NDT} value meets the 114 ºF acceptance criterion set for these types of weld components in the EPRI BWRVIP-74-A and BWRVIP-05 reports.

Issue
The calculated mean RT_{NDT} values reported for 54 EPFY in LRA Table 4.2-6 were based on use of Position 1.1 in RG 1.99, Revision 2, which uses chemistry factor tables contained in the RG as part of the basis for performing RT_{NDT} value projections. However, Position 2.1 of the RG includes criteria for using RPV surveillance data to perform RT_{NDT} calculations, particularly when two or more sets of RPV surveillance data become available for incorporation into

the calculations. Implementation of the applicant's BWRVIP integrated surveillance program (ISP) has generated four sets of credible surveillance data for the RPV axial welds that are made from the Weld Heat No. 5P6756 material. However, LRA Table 4.2-6 did not include any mean RT_{NDT} calculation based on the BWRVIP ISP data that are available for these weld components. The staff's independent calculation projects that the mean RT_{NDT} value for these welds could be as high as 110.5 ºF at 54 EFPY if all credible BWRVIP ISP surveillance data for the welds were applied to the calculations and the Surveillance-Ratio procedure (as defined in Position 2.1 of RG 1.99, Revision 2) is used as the basis for the chemistry factor value used in the RT_{NDT} calculation.

<u>Request</u>

Justify the basis for not including an additional mean RT_{NDT} calculation in LRA Table 4.2-6 for the RPV lower-intermediate axial welds that is based on the use of the BWRVIP ISP surveillance data that apply to these weld components (i.e., the current set of ISP data compiled by the EPRI BWRVIP for Weld Heat No. 5P6756).

RAI 4.3.1-1 (Class 1 Fatigue)

<u>Background</u>

The regulation in 10 CFR 54.21(c)(1) requires the applicant to provide an evaluation of each analysis conforming to the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3(a) and to demonstrate that the TLAA is acceptable in accordance with one or more of three TLAA disposition bases stated in the §54.21(c)(1) requirement:

(i) demonstration that the TLAA remains valid for the period of extended operation

(ii) demonstration that the TLAA has been projected to the end of the period of extended operation

(iii) demonstration that the effects of aging (associated with the TLAA) on the intended function(s) of the component(s) will be adequately managed during the period of extended operation

LRA Section 4.3.1 and its subsections provide the applicant's metal fatigue TLAAs for ASME Code Class 1 components that have been analyzed in accordance with a cumulative usage factor (CUF) analysis. In this part of the LRA, the applicant dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) using one of the following AMPs: (a) LRA AMP B.1.18, "Fatigue Monitoring Program," for all Class 1 components with CUF type fatigue analyses, other than Class 1 reactor vessel internal (RVI) components, and (b) LRA AMP B.1.10, "BWR Vessel Internals Program," for any RVI components that have been analyzed with CUF analyses.

<u>Issue</u>

LRA Section 4.3.1.1 does not identify the components in the reactor pressure vessel that have been analyzed with a CUF analysis. The staff cannot determine which of the RPV components are within the scope of the TLAA addressed in LRA Section 4.3.1.1.

<u>Request</u>

Identify all RPV components that have been analyzed with a CUF analysis.

RAI 4.3.1-2 (Class 1 Fatigue)

Background

The regulation in 10 CFR 54.21(c)(1) requires the applicant to provide an evaluation of each analysis conforming to the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3(a) and to demonstrate that the TLAA is acceptable in accordance with one or more of three TLAA disposition bases stated in the §54.21(c)(1) requirement:

(i) demonstration that the TLAA remains valid for the period of extended operation

(ii) demonstration that the TLAA has been projected to the end of the period of extended operation

(iii) demonstration that the effects of aging (associated with the TLAA) on the intended function(s) of the component(s) will be adequately managed during the period of extended operation

LRA Section 4.3.1 and its subsections provide the applicant’s metal fatigue TLAAs for ASME Code Class 1 components that have been analyzed in accordance with a cumulative usage factor (CUF) analysis. In this part of the LRA, the applicant dispositioned the TLAAs in accordance with 10 CFR 54.21(c)(1)(iii) using one of the following AMPs: (a) LRA AMP B.1.18, “Fatigue Monitoring Program,” for all Class 1 components with CUF type fatigue analyses, other than Class 1 reactor vessel internal (RVI) components, and (b) LRA AMP B.1.10, “BWR Vessel Internals Program,” for any RVI components that have been analyzed with CUF analyses.

Issue

LRA Section 4.3.1.2 does not identify which RVI components have been analyzed with a CUF analysis or adequately describe how the BWR Vessel Internals will accomplish management of fatigue-induced cracking in those RVI components that have been analyzed with a CUF analysis in the current licensing basis (CLB). Specifically, the applicant has not identified the specific BWRVIP report or reports and BWRVIP-defined methods that will be implemented (as part of the BWR Vessel Internals Program) to manage fatigue-induced cracking in the components. Without this information, the staff does not have sufficient demonstration that implementation of the BWR Vessel Internals Program will adequately manage fatigue-induced cracking in those RVI components that have been analyzed with CUF analyses in the CLB.

Request

Identify each RVI component that has been analyzed with a CUF analysis in the CLB and the specific BWRVIP report that will be used to manage fatigue-induced cracking in the component during the period of extended operation. For each of these RVI components, clarify and explain the specific BWRVIP-defined aging management method (e.g., specific type of condition monitoring or inspection activity, performance monitoring activity, time-dependent evaluation, etc.) that will be used as the basis for managing fatigue-induced cracking in the component and justify why the specific method is considered to be capable of managing fatigue-induced cracking in the component during the period or extended operation. If the applicable BWRVIP report will apply a bounding inspection-based approach for the management of cracking, clarify the specific RVI component that will be inspected on behalf of the component with a CUF analysis, and identify the type of corrective actions that will be applied to the RVI component with the CUF analysis if the inspections of the bounding component identify any evidence of cracking in the component being inspected.

RAI 4.3.2-1 (Non-Class 1 Fatigue)

Background

The regulation in 10 CFR 54.21(c)(1) requires the applicant to provide an evaluation of each analysis conforming to the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3(a) and to demonstrate that the TLAA is acceptable in accordance with one or more of three TLAA disposition bases stated in the §54.21(c)(1) requirement:

(i) demonstration that the TLAA remains valid for the period of extended operation

(ii) demonstration that the TLAA has been projected to the end of the period of extended operation

(iii) demonstration that the effects of aging (associated with the TLAA) on the intended function(s) of the component(s) will be adequately managed during the period of extended operation

LRA Section 4.3.2.1 provides the applicant's metal fatigue TLAA for non-Class 1 piping or in-line components that have been analyzed with a time-dependent expansion stress/maximum allowable stress range reduction analysis (implicit fatigue analysis), as may have been required in accordance with applicable ASME Section III NC or ND design rules or ANSI B31.1 design rules. LRA Section 4.3.2.2 provides the applicant's metal fatigue TLAA for non-Class 1 components that have been analyzed in accordance with an ASME-defined cumulative usage factor analysis (CUF) analysis. The scope of these analyses includes components in the engineered safety feature (ESF) systems, auxiliary (AUX) systems, and steam and power conversion (SPC) systems.

The relevant aging management review (AMR) tables for non-Class 1 systems are given in LRA Tables 3.2.2-1 – 3.2.2-7 and LRA Tables 3.2.2-8-1 – 3.2.2-8-5 for ESF system components, LRA Tables 3.3.2-1 – 3.3.2-17 and LRA Tables 3.3.2-18-1 – 3.3.2-18-26 for auxiliary (AUX) system components, and LRA Table 3.4.2-1 and LRA Tables 3.4.2-2-1 – 3.4.2-2-4 for steam and power conversion (SPC) system components. In LRA Section 4.3.2.1, the applicant states that the non-Class 1 fatigue screening document in Appendix H of the EPRI Mechanical Tools was used to determine locations susceptible to fatigue cracking in non-Class 1 systems at RBS and that the first step in the screening process was to identify non-Class 1 components that may have normal or upset condition operating temperature in excess of 220ºF for carbon steel or 270ºF for stainless steel.

Issue

In LRA Section 4.3.2.1, the applicant states that, for many non-Class 1 plant systems, the implicit fatigue analyses for the systems were associated with the cumulative occurrences (i.e., cycles) of the plant's heatup and cooldown operations. The applicant also identified that the implicit fatigue analyses for the pressure relief, residual heat removal (RHR) reactor core isolation cooling (RCIC), containment penetration, control rod drive (CRD), fire protection – water, combustible gas control, standby diesel generator, high pressure core spray diesel generator, reactor water cleanup, and sampling systems may have included some additional transient considerations. Thus, with the exception of the non-Class 1 systems that were specifically referred to in LRA Section 4.3.2.1, it is not evident which of the non-class 1 systems in the LRA were within the scope of the metal fatigue TLAA in LRA Section 4.3.2.1 and which of the non-Class 1 systems were specifically screened out from the TLAA using the screening methodology for fatigue in EPRI Mechanical Tools.

Request

Identify all non-Class 1 ESF, AUX, and SPC systems that are within the scope of the metal fatigue TLAA in LRA Section 4.3.2.1 based solely on an assessment of plant heatup and cooldown operations.

RAI 4.3.3-1 (Effects of Reactor Water Environment on Fatigue Life)

Background

The regulation in 10 CFR 54.21(c)(1) requires the applicant to provide an evaluation of each analysis conforming to the definition of a time-limited aging analysis (TLAA) in 10 CFR 54.3(a) and to demonstrate that the TLAA is acceptable in accordance with one or more of three TLAA disposition bases stated in the §54.21(c)(1) requirement:

(i) demonstration that the TLAA remains valid for the period of extended operation

(ii) demonstration that the TLAA has been projected to the end of the period of extended operation

(iii) demonstration that the effects of aging (associated with the TLAA) on the intended function(s) of the component(s) will be adequately managed during the period of extended operation

LRA Section 4.3.3 provides the applicant's time-limited aging analysis (TLAA) evaluation for environmentally-assisted fatigue (EAF). In LRA AMP B.1.18, "Fatigue Monitoring Program," and Commitment No. 11 in LRA USAR Supplement Table A.4, "Commitment Tracking List," the applicant commits to perform CUF_{en} calculations of selected reactor coolant pressure boundary (RCPB) component locations prior to August 29, 2023. By letter dated August 1, 2017 (ML17213A064), the applicant supplemented the information in LRA Section 4.3.3 and provided additional details on the process that would be used to determine if additional EAF calculations (CUF_{en} calculations) would need to be performed for additional RCPB components beyond those selected for CUF_{en} analysis using the methodology in NUREG/CR-6260. The LRA supplement indicates that applicant will use a thermal zone analysis approach as its basis for determining whether additional RCPB locations will be more limiting for CUF_{en} than those locations defined in NUREG-6260 for CUF_{en} analysis.

Issue

The applicant's response to Question 2 in the LRA supplement letter of August 1, 2017, did not specifically define the details and criteria of its thermal zone analysis methodology or specifically explain how the methodology would be used to select bounding or sentinel RCPB component locations for inclusion in the CUF_{en} calculations. However, the details related to EAF analyses in the August 1, 2017, LRA supplement are not reflected in the LRA's current USAR supplement for the TLAA (i.e., in LRA Section A.2.2.3).

Request

1. Clarify whether the EAF methodology includes any methodology criteria for comparing components in different thermal zones to each other. If so, describe and justify the aspects of the methodology that will be used to make the comparison of components in

different thermal zones, including any assumptions that apply to and are used in the methodology.

2. For the aspect of the EAF methodology that compares component locations within a given thermal zone, clarify whether the methodology will be based on use of a bounding thermal transient, a set of bundled thermal transients, or all plant thermal transients that apply to the components evaluated in the thermal zone. Define and justify the relevant criteria and parameters that will be used for this thermal zone component comparison basis.

3. Provide and justify the selection criteria that will be applied to each thermal zone in order to select the bounding or sentinel component locations for the EAF analysis.

RAI 4.7.3-1 (Fluence Effects for Reactor Vessel Internals)

Background

The regulation in 10 CFR 54.21(c)(1)(ii) states that, for a specific time limited aging analyses (TLAA) that is dispositioned in accordance with this regulation, the applicant must demonstrate that the analysis has been projected to the end of the period of extended operation.

License renewal application (LRA) Section 4.7.3, "Fluence Effects for Reactor Vessel Internals," identifies the neutron fluence analysis for the plant's reactor vessel internal (RVI) components as a TLAA for the LRA. The applicant has dispositioned this TLAA in accordance with the requirement in 10 CFR 54.21(c)(1)(ii) to demonstrate that the neutron fluence values for the RVI components have been projected to the end of the period of extended operation. For the case of the TLAA evaluated in LRA Section 4.7.3, fluence is an effect that factors into TLAA assessment for the RVI components evaluated in the LRA section.

Issue

In LRA Section 4.7.3, the applicant states: "the effects of fluence for 60 years of operation (54 EFPY) were analyzed for the RVI components included in the design specification." However, the applicant did not identify the methodology that was used for estimating fluence for various RVI component locations or provide a discussion regarding how the chosen methodology for estimating fluence for various RVI component locations was qualified prior to use. In addition, fluence methods adherent to RG 1.190 may not be appropriate for estimating neutron fluence for some RVI components. For example, RG 1.190 permits representation of internal fuel assemblies in considerably less detail than peripheral assemblies because the neutron flux on the RPV is primarily due to fuel at the core periphery; this is not the case for components such as the top guide.

Request

Explain why the fluence calculational methodology used to support the disposition of the TLAA under 10 CFR 54.21(c)(1)(ii) is considered to be appropriate for projecting RVI component neutron fluence values to the end of the period of extended operation. Include the following information:

1. Demonstrate that the spatial discretization is sufficiently refined and produces a reliable neutron fluence estimate at the various RVI component locations of interest.

2. Demonstrate that the chosen fluence methods are qualified for estimating neutron fluence at the various RVI component locations of interest.

3. Explain how the neutron fluence values used in downstream safety analyses have been augmented to account for any uncertainty associated with the calculation methods, nuclear data, or modeling accuracy.

4. Quantify neutron fluence margins to RVI components meeting relevant acceptance criteria in downstream safety evaluations.

RAI 4.7.3-2 (Fluence Effects for Reactor Vessel Internals)

Background
In LRA Section 4.7.3 and in the USAR Supplement for the TLAA in LRA Section A.2.5.3, the applicant states:

> "The effects of fluence for 60 years of operation (54 EFPY) were analyzed for the reactor vessel internals components included in the design specification. Location-specific fluence levels were determined. The internal core support structure components were then evaluated against the fluence criteria in the design specification. The evaluation determined that the RBS internal core support structure components meet the design specification for operating conditions through 54 EFPY."

LRA Section 4.7.3 further references proprietary GEH report 003N9941, Rev. 0, "River Bend Nuclear Station, Fluence Effect Evaluation on RPV Internal Components."

SRP-LR Section 4.7.3.1.2 states that for a TLAA disposition pursuant to 10 CFR 54.21 (c)(1)(ii), the applicant shall provide a sufficient description of the analysis and document the results of the reanalysis to show that it is satisfactory for the 60-year period.

SRP-LR Section 4.7.2.2, which contains the FSAR Supplement acceptance criteria for plant-specific TLAAs, states, "the specific criterion for meeting 10 CFR 54.21(d) is: The summary description of the evaluation of TLAAs for the period of extended operation in the FSAR supplement is appropriate such that later changes can be controlled by 10 CFR 50.59. The description contains information associated with the TLAAs regarding the basis for determining that the applicant has made the demonstration required by 10 CFR 54.21(c)(1)."

Issue: LRA Sections 4.7.3 and A.2.5.3 do not appear to include sufficient information or analysis details to demonstrate the applicant's basis for dispositioning the TLAA in accordance with 10 CFR 54.21(c)(1)(ii) other than referencing the design specification acceptance criterion for fluence levels. Such details are necessary to satisfy SRP-LR Section 4.7.3.1.2, SRP-LR Section 4.7.2.2, and 10 CFR 54.21(d).

Request

Describe the assumptions and conservatisms that have changed from the original analysis. As part of this clarification, provide the location-specific fluence levels as projected to the end of the period of extended operation.

RAI A.1.37-1 (Reactor Vessel Surveillance)

Background

LRA Section B.1.37 provides the Reactor Vessel Surveillance Program for the LRA. The LRA indicates the program is an existing aging management program and identifies that the version of the ISP that will be implemented for the period of extended operation will be based on EPRI's updated ISP and methodology that is provided in the BWRVIP-86, Rev. 1-A report. The applicant provides its updated safety analysis report supplement for the AMP in LRA Section A.1.37.

The applicant's Reactor Vessel Surveillance Program (LRA AMP B.1.37) was approved to implement EPRI's approved ISP for boiling water reactor designs in River Bend Station (RBS) Facility License Amendment 136 (ADAMS ML0320504540). The current BWRVIP surveillance capsule reports for the program are given in EPRI Non-Proprietary Report No. BWRVIP-113NP (ML102580248, which provides the RBS 183º Capsule Report and contains the surveillance data for the specific RBS RPV shell plate [Heat No. C3054-2] and weld components [Heat No. 5P6756] represented in the ISP) and in the following additional BWRVIP supplemental surveillance program (SSP) reports that include additional data for the represented weld components: (a) BWRVIP-87NP, Revision 1 (ADAMS ML102420110, providing the data for SSP capsules D, G, and H), (b) BWRVIP-111NP, Revision 1 (ML102720220, providing the data for BWRVIP SSP capsules E, F, and I), and (c) BWRVIP-169NP (ML102590092, providing the data for BWRVIP SSP capsules A, B, and C). These reports were submitted by EPRI to the NRC's document control desk on behalf of BWR licensees participating in the BWRVIP ISP.

Issue

The USAR supplement summary description for the Reactor Vessel Surveillance Program states that the appropriate surveillance data are given in EPRI BWRVIP-135. However, EPRI does not submit EPRI Report BWRVIP-135 or any of its revisions to the NRC document control desk for inclusion in ADAMS. Therefore, the staff requests additional clarifications to verify that the relevant ISP surveillance data from the reports referenced in the background section are within the scope of the RBS RV Surveillance Program and that the data from these reports have been appropriately evaluated in the calculations of RPV adjusted reference temperature values, upper shelf energy values, and mean adjusted reference temperature values that were provided in LRA Section 4.2.

Request

Identify all BWRVIP generated RPV surveillance capsule reports that currently provide the surveillance data inputs for specific RBS RPV weld and plate materials that have been evaluated in the latest version of EPRI's BWRVIP-135 report. Clarify whether the time-limited aging analysis (TLAA) adjusted reference temperature and upper shelf energy evaluation rows included in LRA Tables 4.2-2 and 4.2-3 under the row headings "Integrated Surveillance Program for BWRVIP-135" provide the actual BWRVIP-135 surveillance data evaluations for the these materials or only a partial summary of the BWRVIP-135 surveillance evaluations for the materials. If the latter, the staff requests that the applicant submit the appropriate tables and evaluations from BWRVIP-135 report that apply to the specific RBS RPV plate and weld materials that have been included and evaluated in the BWRVIP ISP.

RAI B.1.9-1 (BWR Vessel ID Attachment Welds)

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 by demonstrating that the matter evaluated in the GALL Report applies to the plant.

Issue

Relevant information is given in LRA Section B.1.9, "BWR Vessel ID Attachment Welds Program," the applicant's program evaluation report for the AMP, and RBC Condition Report Nos. CR-RBS-2014-000253 and CR-RBS-2014-00016. The OE summaries in the referenced condition reports indicate that the applicant has been performing inspections of the feedwater sparger end brackets and their pins using the guidelines in General Electric Company (GE) SIL 658, "Feedwater Sparger End Bracket Degradation," dated July 2008. However applicant does not clarify whether further inspections in accordance with GE SIL 658 are within the scope of the AMP or whether further implementation of the augmented inspections recommended in GE SIL 658 are considered to be applicable enhancements of the "scope of program," "detection of aging effects," and "monitoring and trending" elements of the program.

Request

Clarify (with an appropriate justification) whether the guidelines in GE SIL 658 are within the scope of the BWR Vessel ID Attachment Welds Program and, if so, whether the inspections that will be performed on feedwater and core spray brackets in accordance with the SIL are considered to be enhancements of the "scope of program," "detection of aging effects," and "monitoring and trending" program elements that go beyond the program element criteria for inspecting these types of components in GALL AMP XI.M9, "BWR Vessel Internals." Otherwise, justify why inspections performed in accordance with GE SIL 658 would no longer need to be continued as part of the program during the period of extended operation.

RAI B.1.9-2 (BWR Vessel ID Attachment Welds)

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 by demonstrating that the matter evaluated in the GALL Report applies to the plant.

Issue

Relevant information is given in LRA AMP Section B.1.9, "BWR Vessel ID Attachment Welds Program," and RBS specific reports RBS-EP-13-00004, "RBS FR-17 Reactor Vessel Internals Management Program Post-Outage Report," RBS-EP-15-00014, RBS RF-18 In-Vessel Visual Inspection (IVVI) Final Report," and Attachment 3 of RBG-47362, "Entergy Facilities Reactor Vessel Internals Inspection Histories." The OE summary in the referenced plant-specific inspection history identifies that the inspections of the feedwater brackets in 2008 included inspections of both the brackets and their alignment pins. The inspection history also indicated that wear was detected in some of the feedwater bracket pins in 2008. However, the “operating experience” program summary for LRA AMP Section B.1.9 states that the inspections of the feedwater brackets in 2008 did not result in the detection of any relevant indications in the brackets.

Request

Provide justification for why the inspection history is excluded, or amend the “operating experience” program element summary of LRA AMP B.1.9, “BWR Vessel ID Attachment Welds” Program to include an OE evaluation summary of the wear that was detected in the plant’s feedwater bracket pins in 2008. As part of this summary and in light of the wear detected in 2008, 1) provide the basis why the current programmatic activities for inspecting and evaluating the brackets and their pins are still considered to be adequate for managing potential loss of material due to wear in the components during the period of extended operation, and 2) describe any adjustments of the AMP’s program element criteria and BWRVIP report methods that currently apply to inspections and evaluations of these components.

RAI B.1.10-2 (BWR Vessel Internals)

Background

The regulation in 10 CFR 54.4(a)(3) requires applicants to include structures, systems, or components (SSCs) in the scope of their license renewal applications (LRAs) if the SSCs are within the scope of specific regulations, including those SSCs that are subject to the regulation in 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without SCRAM Events for Light-Water Cooled Nuclear Power Plants.”

For those SSCs that are scoped into an LRA in accordance with 10 CFR 54.4, the regulation in 10 CFR 54.21(a)(1) requires the applicant to subject the SSCs to an aging management review (AMR) if the SSCs are: (a) are not active or do not involve changes in configuration or moving parts, and (b) are not subject to replacement based on a qualified life or specified time period. For those SSCs that are scoped into an LRA and are required to be subjected to an AMR, the regulation in 10 CFR 54.21(a)(3) 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.

The applicant provides its responses to specific applicant action items (AAIs) on specified EPRI BWRVIP reports in Appendix C of the LRA. In its response to AAI #4 on BWRVIP-27-A, the applicant states that the standby liquid control (SLC)/core ΔP lines inside the reactor vessel have no safety or license renewal intended function and are not subject to aging management review.

Issue

The standby liquid control system (SLCS) is typically included with the scope of license renewal because the system meets the scoping criterion in 10 CFR 54.4(a)(3) for systems and components that are subject to the regulation in 10 CFR 50.62, “Requirements for Reduction of Risk Associated with Anticipated Transients without SCRAM (ATWS) Events for Light-Water Cooled Nuclear Power Plants.” In Updated Safety Analysis Report (USAR) Section 9.3.5.2, the applicant states the SLCS is designed to meet the ATWS requirements in 10 CFR 50.62.

Request

Provide the basis for not including the reactor internal portions of the SLCS within the scope of the LRA based on the requirements in 10 CFR 54.4(a)(3) and the system's intended function of mitigating the consequences of ATWS events.

RAI B.1.10-4 (BWR Vessel Internals)

Background

The regulation in 10 CFR 54.4(a)(3) requires applicants to include structures, systems, or components (SSCs) in the scope of their license renewal applications (LRAs) if the SSCs are within the scope of specific regulations, including those SSCs that are subject to the regulation in 10 CFR 50.62, "Requirements for Reduction of Risk from Anticipated Transients Without SCRAM Events for Light-Water Cooled Nuclear Power Plants."

For those SSCs that are scoped into an LRA in accordance with 10 CFR 54.4, the regulation in 10 CFR 54.21(a)(1) requires the applicant to subject the SSCs to an aging management review (AMR) if the SSCs are: (a) are not active or do not involve changes in configuration or moving parts, and (b) are not subject to replacement based on a qualified life or specified time period. For those SSCs that are scoped into an LRA and are required to be subjected to an AMR, the regulation in 10 CFR 54.21(a)(3) 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.

In LRA Section B.1.10, BWR Vessel Internals Program, and the program document for the AMP identify that the applicant performs periodic inspections of the core shroud in accordance with the proprietary reactor vessel internal (RVI) inspection and evaluation (I&E) guidelines in EPRI Report No. BWRVIP-76-A. In year 2008, the applicant issued a plant-specific inspection report that identified cracking in the RBS core shroud that was approximately 9% of the total weld length of the limiting degraded horizontal weld (i.e., H4 weld) in the shroud. At the time of that inspection, the core shroud was characterized as an ERPI Category B core shroud for BWRVIP-76-A inspection purposes. Since that time, further inspections of the core shroud were performed in accordance with BWRVIOP-76-A. An updated assessment in a Year 2017 condition report (i.e., RBS record CR-RBS-2017-01066) indicates that the cumulative extent of cracking in the H4 weld is large enough to the extent that the applicant would need to re-categorize the shroud as a Category C core shroud based on the proprietary methodology in EPRI Report BWRVIP-76-A.

Issue

The "operating experience" discussion of LRA AMP B.1.10, BWR Vessel Internals Program did not discuss whether the plant's core shroud has been re-classified as a BWRVIP-76-A defined Category C shroud based on the results of the applicant's assessment in RBS Report No. CR-RBS-2017-01066.

Request

Clarify whether the BWR Vessel Internals Program has been updated to reclassify the plant's core shroud as a BWRVIP-defined Category C core shroud per the applicable criteria and methodology in the BWRVIP-76-A report. Otherwise, provide the basis why the "detection of aging effects," and "monitoring and trending" elements of the AMP would not need to be enhanced accordingly if the shroud has yet to be re-classified as a BWRVIP-76-A Category C shroud based on the results of the applicable operating experience.

RAI B.1.25-1 (Internal Surfaces in Miscellaneous Piping and Ducting Components)

Background

GALL Report AMP XI.M38, "Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components," as modified by LR-ISG-2012-02, "Aging Management of Internal Surfaces, Fire Water Systems, Atmospheric Storage Tanks, and Corrosion under Insulation," states parameters monitored or inspected include visible evidence of loss of material in metallic components.

LRA Section B.1.25, "Internal Surfaces in Miscellaneous Piping and Ducting Components," states "[f]or metallic components, visual inspection will be used to detect evidence of loss of material and reduction of heat transfer" and that this new program will be consistent with GALL Report AMP XI.M38, as modified by LR-ISG-2012-02.

The LRA (e.g., Table 3.3.2-9, "Combustible Gas Control") states that metallic components will be managed for cracking and reduction of heat transfer using the Internal Surfaces in Miscellaneous Piping and Ducting Components program.

Issue

It is not clear to the staff that the new Internal Surfaces in Miscellaneous Piping and Ducting Components program will be consistent with GALL Report AMP XI.M38 because GALL Report AMP XI.M38 does not include reduction of heat transfer or cracking in metallic components as aging effects. As a result of these apparent inconsistencies, it appears that the LRA has not included sufficient information with regard to various aging management program elements (e.g., "parameters monitored or inspected," "detection of aging effects," "acceptance criteria") to demonstrate that the reduction of heat transfer and cracking for metallic components will be adequately managed by the new Internal Surfaces in Miscellaneous Piping and Ducting Components program.

Request

Clarify whether the new Internal Surfaces in Miscellaneous Piping and Ducting Components program either:

a) will be consistent with the GALL Report AMP XI.M38 and then provide an alternate aging management program to manage reduction of heat transfer and cracking of metallic components, or
b) will not be consistent with the GALL Report AMP XI.M38 and then provide the additional information for changes to applicable program elements that demonstrate reduction of heat transfer and cracking of metallic components will be adequately managed.

ATTACHMENT 1 Summary of Set 10 RAIs Sent to Entergy Operations Inc.

No.	TRP#	RAI#	RAI Issue	Date Sent to Entergy	Date of Clarification Call	Modified After Clarification Call
1	9	3.1.2.1.2-1	BWR Vessel Internals	12/19/17	01/11/18	Yes
2	9	3.1.2.1.2-2	BWR Vessel Internals	12/19/17	01/11/18	Yes
3	59.1	4.2.1-1	Reactor Vessel Fluence	01/02/18	01/11/18	No
4	59.6	4.2.6-1	Reactor Vessel Axial Weld Failure Probability	01/16/18	01/25/18	No
5	60.1	4.3.1-1	Class 1 Fatigue	01/16/18	01/25/18	Yes
6	60.1	4.3.1-2	Class 1 Fatigue	01/16/18	01/25/18	Yes
7	60.1	4.3.2-1	Non Class 1 Fatigue	01/16/18	01/25/18	Yes
8	60.3	4.3.3-1	Effects of Reactor Water Environment on Fatigue Life	01/16/18	01/25/18	No
9	59.1	4.7.3-1	Fluence Effect for Reactor Vessel Internals	01/02/18	01/11/18	No
10	59.1	4.7.3-2	Fluence Effect for Reactor Vessel Internals	01/02/18	01/11/18	Yes
11	32	A.1.37-1	Reactor Vessel Surveillance Program	01/16/18	01/25/18	No
12	4	B.1.9-1	BWR Vessel ID Attachment Welds	12/20/17	01/11/18	No
13	4	B.1.9-2	BWR Vessel ID Attachment Welds	12/20/17	01/11/18	No
14	9	B.1.10-2	BWR Vessel Internals	12/19/17	01/11/18	Yes
15	9	B.1.10-4	BWR Vessel Internals	12/19/17	01/11/18	Yes
16	39	B.1.25-1	Internal Surfaces	01/16/18	Not Required	No