NEXTera® ENERGY SEABROOK

June 24, 2014

10 CFR 54

SBK-L-14089
Docket No. 50-443

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

Seabrook Station
Response to Request for Additional Information Related to the Review of
The Seabrook Station License Renewal Application- Set 21
(Tag No. ME4028)

References:

1. NextEra Energy Seabrook letter SBK-L-10077, "Seabrook Station Application for Renewed Operating License", May 25, 2010. (Accession Number ML101590099).

2. EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), Technical Report 1016596, December, 2008.

3. NRC Regulatory Issue Summary 2011-07, License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management, July 21, 2011.

4. Revision 0 of the Safety Evaluation Report for EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-Rev. 0), June 22, 2011.

5. EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), Technical Report 1022863, December, 2011.

6. Revision 1 to the Safety Evaluation Report for EPRI Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A), December 16, 2011.

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7. LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors", May 28, 2013.

8. NRC Letter, Requests For Additional Information Related to the Review of the Seabrook Station, License Renewal Application- Set 21 (TAC NO. ME4028), April 25, 2014, (Accession Number ML14101A324).

9. NextEra Energy Seabrook letter SBK-L-11063, "Response to Request for Additional Information, NextEra Energy Seabrook License Renewal Application, Request for Additional Information – Set 13", April 14, 2011.

In Reference 1, NextEra Energy Seabrook submitted an application for a renewed facility operating license for Seabrook Station Unit 1 in accordance with the Code of Federal Regulations, Title 10, Parts 50, 51, and 54.

Enclosure 1 provides NextEra Energy Seabrook response to the information requested in Reference 8 and provides revised response to RAI Follow-up B.2.1.27-2 requested in Reference 9.

Enclosure 2 provides the revised NextEra Energy Seabrook PWR Vessel Internals Program.

Enclosure 3 provides the revised AMR items for the Reactor Vessel Internals (Revised LRA Table 3.1.2-3).

Enclosure 4 provides the revised LRA Appendix A - Final Safety Report Supplement Table A.3, License Renewal Commitment List, updated to reflect the license renewal commitment changes made in NextEra Energy Seabrook correspondence to date. Commitment #1 has been revised as a result of response to the RAI related to the aging management of PWR Vessel Internals. This letter also makes editorial corrections to Commitments #12, #13, and #26. Corrections to commitments #12, #13 and #26 now reflect the changes that were previously submitted in SBK-L-14037, dated March 5, 2014.

The changes are explained, and where appropriate to facilitate understanding, portions of the LRA are repeated with the change highlighted by strikethroughs for deleted text and bolded italics for inserted text. In some instances the entire text of a section has been replaced or added. In these cases a note is included in the introduction indicating the replacement of the entire text of the section.

If there are any questions or additional information is needed, please contact Mr. Edward J. Carley, Engineering Supervisor - License Renewal, at (603) 773-7957.

If you have any questions regarding this correspondence, please contact Mr. Michael H. Ossing Licensing Manager, at (603) 773-7512.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on June 24, 2014.

Sincerely,

NextEra Energy Seabrook, LLC

Dean Curtland
Site Vice President

Enclosures:
Enclosure 1- NextEra Energy Seabrook Response to Request for Additional Information Related to the Review of the Seabrook Station License Renewal Application- Set 21
Enclosure 2- Revised NextEra Energy Seabrook PWR Vessel Internals Program
Enclosure 3- Revised AMR Items for the Reactor Vessel Internals (Revised LRA Table 3.1.2-3)
Enclosure 4- LRA Appendix A - Final Safety Report Supplement Table A.3, License Renewal Commitment List Updated to Reflect Changes to Date

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Enclosure 1 to SBK-L-14089

NextEra Energy Seabrook Response to Request for Additional Information Related to the Review of the Seabrook Station License Renewal Application- Set 21

And

Revised Response to RAI Follow-up B.2.1.27-2 Provided in SBK-L-11063 dated April 14, 2011

RAI LRA Appendix B-1 – Aging Management of Reactor Vessel Internals

Background:

The license renewal application (LRA) for Seabrook Station proposed aging management for the reactor vessel internal (RVI) components based on a regulatory commitment in the LRA's Updated Final Safety Analysis Report (UFSAR) Supplement. The commitment stated that the applicant will develop an aging management program (AMP) and inspection plan based on augmented inspection activities for the components developed by the EPRI Materials Reliability Project (MRP), and that the inspection plan will be submitted for NRC review and approval not later than 2 years after receipt of the renewed license or not less than 24 months prior to the period of extended operation, whichever comes first.

The NRC's recommended AMP for Pressurized Water Reactor (PWR) RVIs in Revision 2 of NUREG-1801, Generic Aging Lessons Learned (GALL) Report, is given in Chapter XI.M16A, "PWR Vessel Internals," which was issued in December 2010. On January 9, 2012, subsequent to the issuance of Revision 2 of the GALL Report, the EPRI MRP issued Technical Report No. 1022863, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A)," which included the NRC safety evaluation (SE) for the report's methodology dated December 16, 2011. On June 3, 2013, the staff revised AMP XI.M16A and the aging management review (AMR) items in GALL for PWR RVI components to be consistent with the contents of the MRP-227-A report and issued them in License Renewal Interim Staff Guidance Document No. LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors."

On July 21, 2011, the NRC issued Regulatory Information Summary (RIS) 2011-07, "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management," which provided updated NRC procedures for LRA reviews of PWR RVI AMPs. This RIS identified Category C plants as those plants that have an LRA currently under review, and stated that these applicants will be expected to revise their commitment for aging management of PWR vessel internals such that the information identified in the SE for MRP-227 would be submitted to the NRC for review and approval not later than two years after issuance of the renewed license or not later than two years before the plant enters the PEO, whichever comes first. Seabrook Station is a Category C plant in accordance with the RIS.

Issue:

The categorization of Seabrook Station and other plants in Category C of RIS 2011-07 was based on an expectation that the LRA would be reviewed and approved on a normal review schedule of 22 months, and that it would be an unreasonable burden to expect the applicant to address all aspects of the NRC's SE on MRP-227 within the LRA review. Since the completion of the staff's review of the LRA is still ongoing due to the ASR open item, the staff has concluded that the applicant should provide an LRA update or amendment that includes updated AMP and AMR items for the RVI components, including responses to the applicable Applicant/License Actions Items identified in the staff's SE for MRP-227 dated December 16, 2011.

Request: The staff requests that the applicant provide either an LRA amendment or an update that includes updated AMP and AMR items for the PWR RVI components at the Seabrook Station that are based on the guidance in LR-ISG-2011-04, including responses to applicable Applicant/License Actions Items identified in the staff's SE for MRP-227 dated December 16, 2011.

NextEra Energy Seabrook Response to RAI LRA Appendix B-1 – Aging Management of Reactor Vessel Internals

NextEra Energy Seabrook previously submitted a PWR Vessel Internals Aging Management Program (RVI AMP) as part of its License Renewal Application (Ref. 1). The RVI AMP was based on EPRI MRP-227 Rev. 0 (Pressurized Water Reactor Internals Inspection and Evaluation Guidelines), which was issued in December 2008 (Ref. 2). As part of the NextEra Energy Seabrook's License Renewal Application (LRA), a commitment was made to implement the program prior to the period of extended operation and to submit an inspection plan to the NRC not less than 24 months prior to the period of extended operation.

In SBK-L-11069 dated April 22, 2011, NextEra Energy Seabrook revised its commitment to submit the inspection plan to the NRC not later than 2 years after receipt of the renewed license or not less than 24 months prior to the period of extended operation, whichever came first. This revised commitment was consistent with the Regulatory Issue Summary (RIS 2011-07), "License Renewal Submittal Information for Pressurized Water Reactor Internals Aging Management", which was issued by the NRC on July 21, 2011 (Ref. 3). NextEra Energy Seabrook fell into Category C (plants that have a license renewal application currently under review) in accordance with the guidance provided in RIS-2011-07.

On January 12, 2009, MRP-227 Rev.0 was submitted to the NRC for review and approval through the Nuclear Energy Institute. On June 22, 2011, the NRC issued Revision 0 of the Safety Evaluation Report (SER) for MRP-227 (Ref. 4). On December 16, 2011, the NRC issued Revision 1 of the Safety Evaluation Report for the final version of MRP-227 [(i.e. MRP-227-A)(Ref. 6)].

In the SER, the NRC staff determined that MRP-227-A is acceptable for use in PWR Vessel Internals aging management program in License Renewal Applications. The SER includes eight plant-specific Applicant/Licensee Action Items. Applicant/Licensee Action Items 4 and 6 are only applicable to B&W plants. NextEra Energy Seabrook is a Westinghouse NSSS plant.

Therefore, Applicant/Licensee Action Items 4 and 6 are not applicable to NextEra Energy Seabrook.

On May 28, 2013, the NRC issued License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors" (Ref. 5). This LR-ISG revised the guidance provided in NUREG-1801, Revision 2, "Generic Aging Lessons Learned Report" for the aging management of PWR Reactor Vessel Internal components exposed to reactor coolant environment.

In response to RAI B.2.1.7, the following changes are made to NextEra Energy Seabrook's LRA:

1. PWR Vessel Internals Program described in LRA Section B.2.1.7 is revised as described in Enclosure 2.
   Note: The entire text of the PWR Vessel Internals Program has been replaced.

2. The PWR Vessel Internals Inspection Plan for NextEra Energy Seabrook is provided in Tables 1 through 4 of the PWR Vessel Internals Aging Management Program.

   a. Table 1 describes the RVI components that are classified as Existing Program components.
   b. Table 2 describes the RVI components classified as Primary components, which is based on MRP-227-A, Table 4.3.
   c. Table 3 describes the RVI components classified as Expansion components, which is based on MRP-227-A, Table 4.6.
   d. Table 4 describes the Examination Acceptance and Expansion criteria for Westinghouse plants, which is based on MRP-227-A, Table 5.3.

3. AMR items for Reactor Vessel Internals (LRA Table 3.1.2-3) have been revised as described in Enclosure 3.
   Note: The entire LRA Table 3.1.2-3 has been replaced.

4. In LRA Appendix A, Section A.2.1.7, PWR Vessel Internals, has been revised as follows.
   Note: The entire text of LRA Section A.2.1.7 has been replaced.

   The PWR Vessel Internals Program implements the guidance provided in EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A, Technical Report 1022863) and EPRI Inspection Standard for PWR Internals (MRP-228, Technical Report 10166609).

   The program is a condition monitoring program designed to manage the aging effects on the PWR vessel internal components. The recommended activities provided in MRP-227-A and additional plant-specific activities not defined in MRP-227-A are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues."

This program is used to manage the effects of age-related degradation mechanisms that are applicable to the PWR vessel internal components. These aging effects include: a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclic loading, b) loss of material induced by wear, c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement, d) changes in dimensions due to void swelling or distortion, and e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

5. Commitment # 1 is revised as follows to provide confirmation and acceptability of the implementation of MRP-227-A by addressing the plant-specific Applicant/Licensee Action Items outlined in section 4.2 of the NRC SER. NextEra Energy Seabrook is currently working with the NSSS supplier to establish a submittal schedule to address the plant-specific Applicant/Licensee Action Items outlined in section 4.2 of the NRC SER.

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
1.	PWR Vessel Internals	~~An inspection plan for Reactor Vessel Internals will be submitted for NRC review and approval.~~ ***Provide confirmation and acceptability of the implementation of MRP-227-A by addressing the plant-specific Applicant/Licensee Action Items outlined in section 4.2 of the NRC SER.***	A.2.1.7	~~Program to be implemented prior to the period of extended operation. Inspection plan to be submitted to NRC not later than 2 years after receipt of the renewed license or not less than 24 months prior to the period of extended operation, whichever comes first.~~ ***NextEra Energy Seabrook will provide a submittal schedule by October 15, 2014***

<u>RAI B.2.1.22-6</u>

<u>Background</u>:

The Buried Piping Inspection Locations table provided in a letter dated July 2, 2013, states the number of inspections based on the material type, interval of inspection, and status of cathodic protection, backfill, and soil sample results.

Issue:

The staff has concluded that the referenced table is consistent with LR-ISG-2011-03 Table 4a, "Inspections of Buried Pipe," with the following exceptions:

1. The not to exceed number of inspections for polymeric piping in the 40–50 year period is three in the referenced table and four in Table 4a of LR-ISG-2011-03. The fire protection system is the only in-scope system with polymeric piping. The Buried Piping and Tanks Inspection Program states that it may conduct jockey pump monitoring in lieu of excavated direct visual inspections of fire protection piping. However, if the program were changed to use excavated direct visual inspections of the fire protection system in lieu of jockey pump monitoring, the number of inspections would not be consistent with LR-ISG-2011-03.

2. Footnote 5 of the referenced table states, "[i]f cathodic protection does not meet Category C and backfill has been determined to be inadequate, buried steel piping will be inspected as Category F [the highest inspection quantity category]." Footnote 2 of the referenced table states, "[t]he adequacy of backfill will be determined by the condition of coatings and base materials noted during inspections. If damage to the coatings or base materials is determined to have been caused by the backfill, the backfill will be considered to be 'inadequate' (for the purpose of this program)." The applicant's table states that Category F inspections will be performed if the soil is determined to be corrosive.

LR-ISG-2011-03 Table 4a states that the conditions that result in conducting the highest number of inspections (Category F) are: (a) coatings and backfill have not been provided in accordance with the "preventive actions" program element, (b) a leak has occurred in buried piping due to external corrosion, (c) significant coating degradation or metal loss in more than 10 percent of inspections conducted, and (d) the soil has been demonstrated to be corrosive for steel piping.

The staff concludes the following in relation to each of the Table 4a conditions:

a. The referenced table is not consistent with condition (b) because it has not stated leaks as a criterion for entry into Category F.
b. The referenced table is not consistent with condition (c) because, although footnote 2 states that if damage to the coatings or base materials are determined to have been caused by the backfill, the backfill will be considered to be inadequate (resulting in use of Category F to determine the number of inspections), the Table 4a condition is not solely based on damage to coatings by backfill.

Request:

1. State the basis why conducting three excavated direct visual inspections of polymeric piping in the 40–50 year period; in lieu of the jockey pump monitoring; is adequate to provide reasonable assurance that the piping will meet its current licensing basis intended function(s).

Alternatively, revise the table to be consistent with the number of recommended inspections for buried polymeric piping in LR-ISG-2011-03.

2. State the basis for why the proposed criteria for invoking Category F inspections will be adequate to provide reasonable assurance that the piping will meet its current licensing basis intended function(s). Alternatively, revise the table to be consistent with LR-ISG-2011-03.

NextEra Energy Seabrook Response to RAI B.2.1.22-6

In response to RAI B.2.1.22-6, the "Buried Piping Inspection Locations Table" provided in SBK-L-13115 (dated July 2, 2013) and SBK-L-13183 (dated October 21, 2013) has been revised as follows:

Buried Piping Inspection Locations

Material Type	Status of Cathodic Protection	GALL Category	Inspections each 10-Year Period [1] 30-40	40-50	50-60	Systems Currently in this Category
AL6XN	N/A	N/A	0	0	0	None
Stainless Steel	N/A	N/A	1	1	1	CO, DG
Polymeric	N/A	A	Adequate Backfill[2]			FP[8]
			1	1	1	
		B	Inadequate Backfill[2,3]			
			1% NTE 2	2 % NTE ~~3~~ *4*	3% NTE 6	
Steel[5]	Installed, available and effective[4]	C	1	1	1	CBA, IA, FP[8], SW
	External corrosion control not required	D	1% NTE 2	1% NTE 2	1% NTE 2	None
	Not practical, not installed, or installed but not meeting Cat C; non-corrosive soil[6]	E	5% NTE 7	6% NTE 10	7.5% NTE 12	AB[7], CBA, CO, DF, DG, FW, FP[8]
	Not installed or installed but not meeting Cat C; corrosive soil[6]	F	10% NTE 15	12% NTE 20	15% NTE 25	

GENERAL NOTES:

1. Each inspection will examine a minimum of 10 feet of pipe or the entire length of a run, whichever is less.
2. The adequacy of backfill will be determined by the condition of coatings and base materials noted during inspections. If damage to the coatings or base materials are determined to have been caused by the backfill, the backfill will be considered to be "inadequate" (for the purpose of this program).
3. If all polymeric pipe in-scope is non-safety related, the inspection quantities may be reduced by half.
4. Cathodic protection is available and effective if it

- was installed or refurbished 5 years prior to the end of the inspection period of interest; and
- has been operational (available) at least 85 percent of the time since 10 years prior to the PEO or since installation or refurbishment (exclusive of time off-line for testing), whichever is shorter; and
- has met the acceptance criteria of Section 6 at least 80 percent of the time since 10 years prior to the PEO or since installation or refurbishment, whichever is shorter.

5. ~~If cathodic protection does not meet Category C, and backfill has been determined to be inadequate, buried steel piping will be inspected as Category F.~~ ***Buried piping will be inspected as category F if; a) cathodic protection does not meet Category C, or b) significant coating degradation or metal loss in more than 10 percent of inspections conducted, or c) a leak has occurred in buried piping due to external corrosion, or d) the soil has been demonstrated to be corrosive for steel piping.***
6. Soil corrosivity is determined by soil analysis using a demonstrated methodology such as EPRI report 1021470, Table 8-1. A soil corrosivity value of 10 or greater using this method is considered corrosive.
7. This line is not is use. It has been drained and flushed and is awaiting replacement per a design change. The inspection criteria for the replacement piping will be determined based on material selection, coating, cathodic protection, and quality of backfill.
8. If Fire Protection piping is inspected by excavation in lieu of by alternative testing (e.g., flow test, jockey pump monitoring), and the extent of examinations is not based on the percentage of piping in the material group, the Not-to-Exceed (NTE) value will be increased by 1 inspection, if normally less than 10, or 2 inspections, if normally 10 or greater.

RAI A.1-2, License Renewal Commitments and the UFSAR

Background:

By letter dated May 25, 2010, NextEra Energy Seabrook, LLC submitted an application pursuant to Title 10 of the Code of Federal Regulations (CFR) Part 54 to renew the operating license, NPF-86 for Seabrook Station, for review by the U.S. Nuclear Regulatory Commission (NRC) staff. The NRC staff is reviewing this application in accordance with the guidance in NUREG-1800, "Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants." During the review of the Seabrook license renewal application (LRA) by the NRC staff, NextEra Energy Seabrook made commitments related to aging management programs (AMPs), aging management reviews (AMRs), and time-limited aging analyses, as applicable, related to managing the aging effects of structures and components prior to the PEO. The list of these commitments, as well as the implementation schedules and the sources for each commitment, will be included as a Table in Appendix A to the LRA and the SER.

In Section 1.7, "Summary of Proposed License Conditions," of the SER with Open Items, the staff stated that following its review of the LRA, including subsequent information and clarifications provided by the applicant, it identified proposed license conditions. The first license condition requires the information in the UFSAR supplement, submitted pursuant to

10 CFR 54.21 (d), as revised during the LRA review process, be made a part of the UFSAR. The second license condition in part states that the new programs and enhancements to existing programs listed in Appendix A of the SER and the applicant's UFSAR supplement be implemented no later than 6 months prior to the PEO. This license condition also states, in part, that activities in certain other commitments shall be completed by 6 months prior to the PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.

The NRC plans to revise Appendix A of the SER to align with this guidance, and to reformat the license condition to be as follows:

> The UFSAR supplement submitted pursuant to 10 CFR 54.21(d), as revised during the license renewal application review process, and as supplemented by Appendix A of NUREG [XXXX], "Safety Evaluation Report Related to the License Renewal of Seabrook Station" dated [Month Year], describes certain programs to be implemented and activities to be completed prior to the PEO.
>
> a. The licensee shall implement those new programs and enhancements to existing programs no later than 6 months prior to the PEO.
> b. The licensee shall complete those inspection and testing activities, as noted in Commitment Nos. [x] through [xx] of Appendix A of NUREG [XXXX], by the 6 month date prior to PEO or the end of the last refueling outage prior to the PEO, whichever occurs later.
>
> The licensee shall notify the NRC in writing within 30 days after having accomplished item (a) above and include the status of those activities that have been or remain to be completed in item (b) above.

The staff also notes that in the course of its evaluating multiple commitments to be implemented in the future in order to arrive at a conclusion of reasonable assurance that requirements of 10 CFR 54.29(a) have been met, these license renewal commitments must be incorporated either into a license condition or into a mandated licensing basis document, such as the UFSAR. Those commitments that are incorporated into the UFSAR are typically done so by incorporating each one verbatim (or by a summary and a commitment reference number) into the respective UFSAR summaries in the applicant's LRA Appendix A.

Issue:

As reflected in the SER Appendix A, the implementation schedule for some commitments may conflict with the implementation schedule intended by the generic license condition. In addition, these licensing commitments need to be incorporated either into a license condition or into the applicant's UFSAR summary in such a manner as discussed above.

Request:

1. Identify those commitments to implement new programs and enhancements to existing programs. Indicate the expected date for completing the implementation of each of these programs and enhancements.

2. Identify those commitments to complete inspection or testing activities prior to the PEO. Indicate the expected dates for the completion of each of these inspection and testing activities.

3. For each commitment provided by the applicant in the SER Appendix A, identify where and how NextEra Energy Seabrook, LLC proposes that it be incorporated: into either a license condition or into the Seabrook UFSAR.

NextEra Energy Seabrook Response to RAI A.1-2, License Renewal Commitments and the UFSAR

In response to RAI A.1-2, Requests 1, 2 & 3, the folowing has been added to the end of LRA Appendix A, A.1 Introduction.

> ***Commitments for implementing new programs and enhancements to existing programs with schedule dates as prior to the period of extended operation shall be completed no later than 6 months prior to the period of extended operation.***
>
> ***Commitments for completion of inspection and testing activities shall be completed no later than 6 months prior to the period of extended operation or the last refueling outage prior to the period of extended operation, whichever is later.***
>
> ***The final version of the License Renewal Commitment List will be included in the NextEra Energy Seabrook UFSAR Supplement (LRA Appendix A) before incorporation into the NextEra Energy Seabrook UFSAR (after NRC approval of the LRA). After incorporation into the NextEra Energy Seabrook UFSAR, changes to information within the UFSAR Supplement will be made in accordance with 10 CFR 50.59.***

NextEra Energy Seabrook will notify the NRC in writing within 30 days after having accomplished items listed in the License Renewal Commitment List and include the status of those activities that have been or remain to be completed.

Revised NextEra Energy Seabrook Response to RAI Follow-up B.2.1.27-2 Provided in SBK-L-11063 dated April 14, 2011

In SBK-L-11063 dated April 14, 2011 (Reference 9), NextEra Energy Seabrook provided a response to RAI Follow-up B.2.1.27-2 describing the planned actions related to monitoring liner plate thickness around the fuel transfer tube and efforts to address leakage of borated water.

At the time the response to the request for additional information was submitted, the liner plate was identified as being Category E-C in accordance with IWE-2420(b) which states in part "when examination results require evaluation of flaws or degradation in accordance with IWE-3000, and the component is acceptable for continued service or when the examination results in performance of a repair/replacement activity, the areas containing such flaws or area of degradation, or areas subject to a repair/replacement activity, shall be reexamined during the next inspection period listed in the schedule of the inspection program of IWE-2411 or IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C." Upon subsequent review, it was determined that this area did not meet the requirements for Category E-C augmented inspections and was reclassified. The original coating indications and minor surface corrosion will not reoccur since the leakage into the vault was successfully remediated during OR11 in October 2006. The areas of concern on the containment liner plate that were originally identified were examined and accepted in October 2009 (via IWE-VT-3 Examination and UT thickness measurements).

Based on the revised classification, NextEra Energy Seabrook has revised the original response to RAI B.2.1.27-2 as follows:

<u>Request for Additional Information (RAI) Follow-up B.2.1.27-2</u>

Background:

By letter dated December 17, 2010, the applicant responded to RAI B.2.1.27-2 and stated that the liner plate around the fuel transfer tube has been identified in the ISI program for augmented inspection in accordance with the 1995 Edition with 1996 Addenda of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, Subsection IWE-2420(b) and (c).

Issue:

The ASME 1995 Edition with 1996 Addenda, Section XI, Subsection IWE-2420(b) and (c) states that reexamination of degraded areas is no longer required if these areas remains essentially unchanged for three consecutive inspection periods. However, it is not clear from the applicant's response if the containment liner plate around the fuel transfer tube is still exposed to the borated water leakage. Exposure to borated water can promote corrosion of the liner plate and adversely affect the ability of the liner to perform its intended function.

Request:

Describe steps that are being taken to monitor the liner plate thickness around the transfer tube and/or efforts to address the leakage of borated water.

Revised: NextEra Energy Seabrook Response to RAI Follow-up B.2.1.27-2:

~~The leak path into the fuel transfer tube vault has been repaired and the borated water leakage stopped. The areas of the containment liner plate that had showed signs of deficiency (loss of material) have been examined and accepted. The areas are subject to IWE required augmented UT examinations for the next three exam cycles. If no further degradation (loss of material) is observed during those three cycles, the subject area will return to normal visual IWE inspections. These visual inspections would be able to identify any further leakage of borated water.~~

In SBK-L-11063 dated April 14, 2011, NextEra Energy Seabrook previously provided a response to RAI Follow-up B.2.1.27-2 describing the planned actions related to monitoring liner plate thickness around the fuel transfer tube and efforts to address leakage of borated water.

At the time the response to the request for additional information was submitted, the liner plate was identified as being Category E-C in accordance with IWE-2420(b) which states in part "when examination results require evaluation of flaws or degradation in accordance with IWE-3000, and the component is acceptable for continued service or when the examination results in performance of a repair/replacement activity, the areas containing such flaws or area of degradation, or areas subject to a repair/replacement activity, shall be reexamined during the next inspection period listed in the schedule of the inspection program of IWE-2411 or IWE-2412, in accordance with Table IWE-2500-1, Examination Category E-C." Upon subsequent review, it was determined that this area did not meet the requirements for Category E-C augmented inspections and was reclassified. The original coating indications and minor surface corrosion will not reoccur since the leakage into the vault was successfully remediated during OR11 in October 2006. The areas of concern on the containment liner plate that were originally identified were examined and accepted in October 2009 (via IWE-VT-3 Examination and UT thickness measurements).

Enclosure 2 to SBK-L-14089

Revised NextEra Energy Seabrook PWR Vessel Internals Program

B.2.1.7 PWR VESSEL INTERNALS PROGRAM

Program Description

The PWR Vessel Internals Program implements the guidance provided in EPRI Pressurized Water Reactor Internals Inspection and Evaluation Guidelines (MRP-227-A, Technical Report 1022863) and EPRI Inspection Standard for PWR Internals (MRP-228, Technical Report 10166609).

The program is a condition monitoring program designed to manage the aging effects on the PWR vessel internal components. The recommended activities provided in MRP-227-A and additional plant-specific activities not defined in MRP-227-A are implemented in accordance with Nuclear Energy Institute (NEI) 03-08, "Guideline for the Management of Materials Issues."

This program is used to manage the effects of age-related degradation mechanisms that are applicable to the PWR vessel internal components. These aging effects include: a) cracking, including stress corrosion cracking (SCC), primary water stress corrosion cracking (PWSCC), irradiation-assisted stress corrosion cracking (IASCC), and cracking due to fatigue/cyclic loading, b) loss of material induced by wear, c) loss of fracture toughness due to either thermal aging or neutron irradiation embrittlement, d) changes in dimensions due to void swelling or distortion, and e) loss of preload due to thermal and irradiation-enhanced stress relaxation or creep.

The program applies the guidance provided in MRP-227-A, for inspecting, evaluating, and, if applicable, dispositioning non-conforming PWR vessel internal components. These examinations provide reasonable assurance that the effects of age-related degradation mechanisms will be managed during the period of extended operation. The program includes expanding periodic examinations and other inspections, if the extent of the degradation identified exceeds the expected levels.

MRP-227-A guidance for selecting RVI components for inclusion in the inspection sample is based on a four-step ranking process. Through this process, the PWR vessel internal components are categorized into four groups; 1) Primary, 2) Expansion, 3) Existing Programs, or 4) No Additional Measurements.

The result of this four-step sample selection process is a set of "Primary" PWR vessel internal component locations that are inspected because they are expected to show the leading indications of the degradation effects, with another set of "Expansion" internals component locations that are specified to expand the sample should the indications be more severe than anticipated.

The degradation effects in a third set of internals locations are deemed to be adequately managed by "Existing Programs," such as ASME Code, Section XI, B-N-3 examinations of core support structures. A fourth set of internals locations are deemed to require "No Additional Measures."

Program Elements

The following provides the results of the evaluation of each program element against the 10 elements described in License Renewal Interim Staff Guidance LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors".

Element 1 - Scope of Program

The scope of the RVI AMP includes RVI components at NextEra Energy Seabrook, which is built to a Westinghouse design. The scope of the program applies the methodology and guidance in MRP-227-A, which provides augmented inspection and flaw evaluation methodology for assuring the functional integrity of safety-related internals in commercial operating U.S. PWR nuclear power plants designed by B&W, CE, and Westinghouse. The scope of components considered for inspection under MRP-227-A guidance includes core support structures (typically denoted as Examination Category B-N-3 by the ASME Code, Section XI), those RVI components that serve an intended license renewal safety function pursuant to criteria in 10 CFR 54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any of the functions identified in 10 CFR 54.4(a)(1)(i), (ii), or (iii).

The scope of the program does not include consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation, because these components are not typically within the scope of the components that are required to be subject to an aging management review. The scope of the program also does not include welded attachments to the internal surface of the reactor vessel because these components are considered to be ASME Code Class 1 appurtenances to the reactor vessel and are adequately managed in accordance with NextEra Energy Seabrook's ASME Code, Section Xl, Inservice Inspection, Subsections IWB, IWC, and IWD Program as described in Section B.2.1.1 of the LRA.

Element 2 – Preventive Actions

The guidance in MRP-227-A relies on PWR water chemistry control to prevent or mitigate aging effects that can be induced by corrosive aging mechanisms such as loss of material induced by general corrosion, pitting corrosion, crevice corrosion, or stress corrosion cracking or any of its forms (SCC, PWSCC, or IASCC). NextEra Energy Seabrook's reactor coolant water chemistry is monitored and maintained in accordance with the Water Chemistry Program as described in Section B.2.1.2 of the LRA.

Additionally, NextEra Energy Seabrook replaced the original X-750 Guide Tube Assembly (GTA) support pins with cold worked (CW) 316 SS support pins during Refueling Outage 11 (Fall of 2006). This modification is considered a preventative action as the CW 316 SS support pins are considered to be much more resistant to the degradation mechanisms of concern for the RVI components than the X-750 material. Therefore, there is no requirement for augmented inspections of the CW 316 SS GTA support pins under the PWR Vessel Internals Aging Management Program. However, appropriate actions will be taken upon receiving further recommendations from Westinghouse.

Element 3 – Parameters Monitored/Inspected

A total of eight age related degradation mechanisms are considered applicable to the RVI components: 1) stress corrosion cracking (SCC), 2) irradiation assisted stress corrosion cracking (IASCC), 3) fatigue, 4) irradiation embrittlement (IE), 5) thermal embrittlement (TE), 6) wear, 7) void swelling (VS), and 8) irradiation enhanced stress relaxation/creep (ISR/IC). A brief description of these degradation mechanisms and the associated aging effects are as follows:

Stress Corrosion Cracking (SCC)
SCC is a localized, non-ductile failure caused by a combination of stress, susceptible material, and an aggressive environment. The fracture path of SCC can be either transgranular or intergranular in nature. The aggressive contaminants most commonly associated with SCC of austenitic stainless steels are dissolved chlorides and oxygen. Nickel base alloys such as Alloy 600 and X-750 have exhibited susceptibility to intergranular SCC in primary water without the presence of aggressive contaminants, commonly referred to as primary water stress corrosion cracking (PWSCC). SCC of Stainless Steel (SS) in primary water is also considered feasible at high stress levels. The aging effect of SCC is cracking.

Irradiation Assisted SCC (IASCC)
IASCC is a form of intergranular SCC that results from the combined influence of neutron irradiation and an aggressive environment. A limited number of IASCC failures of RVI components, specifically fasteners, constructed of austenitic stainless steels and nickel base alloys have been observed. The aging effect of IASCC is cracking.

Fatigue
Fatigue is defined as the structural deterioration that can occur as a result of the periodic application of stress by mechanical, thermal, or combined effects. High cycle fatigue results from relatively low cyclic stress (<yield strength) applied for many ($>10^5$) cycles. Low cycle fatigue results from relatively high cyclic stress (≥yield strength) applied for low number of cycles. The aging effect of fatigue is cracking.

Irradiation Embrittlement (IE)
IE refers to a gradual and progressive change in mechanical properties of a material resulting from exposure to high levels of neutron irradiation. These changes include an increase in yield and tensile strengths, and a corresponding decrease in ductility and toughness. The aging effect

of IE is loss of fracture toughness.

Thermal Embrittlement (TE)
Thermal embrittlement refers to the same gradual and progressive change in mechanical properties of a material as IE except it results from exposure to elevated temperatures rather than neutron irradiation. For the RVI components, TE is only a concern for SS castings and welds with duplex microstructures containing both ferrite and austenite. The aging effect of TE is loss of fracture toughness.

Wear
Wear is caused by the relative motion between adjacent surfaces, with the extent determined by the relative properties of the adjacent materials and their surface condition. The aging effect of wear is loss of material.

Void Swelling (VS)
Void swelling is the gradual increase in volume of a component caused by the formation of microscopic cavities. These cavities result from the nucleation and growth of vacancies created by exposure to high levels of neutron irradiation. During the initial licensing periods of domestic PWRs, field experience has not revealed any evidence of VS in RVI components; however it is postulated as a possibility during periods of extended operation based upon accelerated laboratory testing. The aging effect of VS is dimensional change.

Irradiation and Thermally Enhanced Stress Relaxation/Creep (SR/C)
Stress relaxation involves the short term unloading of preloaded components upon exposure to elevated temperatures or high levels of neutron irradiation. Creep is a longer term process in which plastic deformation occurs within a loaded component. The temperatures of RVI are typically not high enough to support creep; however it can develop upon exposure to high levels of neutron irradiation over an extended period. The aging effect of stress relaxation and creep is loss of preload.

For the management of cracking, the program monitors for evidence of surface breaking linear discontinuities if a visual inspection technique is used as the non-destructive examination (NDE) method, or for relevant flaw presentation signals if a volumetric UT method is used as the NDE method. For the management of loss of material, the program monitors for gross or abnormal surface conditions that may be indicative of loss of material occurring in the components. For the management of loss of preload, the program monitors for gross surface conditions that may be indicative of loosening in applicable bolted, fastened, keyed, or pinned connections. The program does not directly monitor for loss of fracture toughness that is induced by thermal aging or neutron irradiation embrittlement. Instead, the impact of loss of fracture toughness on component integrity is indirectly managed by using visual or volumetric examination techniques to monitor for cracking in the components and by applying applicable reduced fracture toughness properties in the flaw evaluations if cracking is detected in the components and is extensive enough to warrant a supplemental flaw growth or flaw tolerance evaluation under the MRP-227-A guidance or ASME Code, Section XI requirements. The program uses physical measurements to monitor for any dimensional changes due to void swelling, irradiation growth, distortion, or deflection.

Specifically, the program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for Westinghouse designed Primary Components in Table 4-3 of MRP-227-A and for Westinghouse designed Expansion Components in Table 4-6 of MRP-227-A.

Element 4 - Detection of Aging Effects

The RVI components have been categorized as Existing Program, Primary, Expansion, or No Additional Measurements, based on the guidance provided in MRP-227-A. A description of the component categories is as follows:

Existing Program Components:
Existing Program Components are susceptible to at least one of the eight degradation mechanisms, for which existing plant programs are capable of managing the associated aging effect(s). Details of the required inspections for Existing Program Components are provided in Table 1, Westinghouse Plants Existing Program Components Applicable to NextEra Energy Seabrook.

Primary Components:
Primary Components are highly susceptible to at least one of the eight degradation mechanisms, for which augmented inspections are required on a periodic basis to manage the associated aging effect(s). Primary Components are considered lead indicators for the onset of the applicable degradation mechanism(s). Details of the required inspections for Primary Components are provided in Table 2, Westinghouse Plants Primary Components Applicable to NextEra Energy Seabrook. The Primary group also includes components which have shown a degree of tolerance to a specific aging degradation effect, but for which no highly susceptible component exists or for which no highly susceptible component is accessible.

Expansion Components:
Expansion Components are highly or moderately susceptible to at least one of the eight degradation mechanisms, but for which functionality assessment has shown a degree of tolerance to those aging effects. Augmented inspections are required once a specified level of degradation is detected in a linked Primary Component. Details of the required inspections for Expansion Components are provided in Table 3, Westinghouse Plants Expansion Components Applicable to

NextEra Energy Seabrook.

No Additional Measures Components:
No Additional Measures Components are either not susceptible to any of the eight degradation mechanisms, or if susceptible, the impact of failure on the functionality of the RVI components is insignificant. No further action is required for managing the aging of these RVI components. Proven inspection methodologies are utilized to detect evidence of the relevant aging mechanism(s) for the Existing Programs, Primary, and Expansion Components. These include the following:

a) Direct physical measurements to monitor for loss of material or preload.

b) VT-3 (visual) exams to monitor for general degradation associated with loss of material or preload.

c) EVT-1 (enhanced visual) exams to monitor for surface breaking linear discontinuities indicative of cracking.

d) UT (ultrasonic) exams to monitor directly for cracking.

e) ECT (eddy current testing) to further characterize conditions detected by VT-3 and EVT-1 examination.

The requirements for the inspection methodologies and qualification of NDE systems used to perform those inspections are provided in EPRI MRP-228, "Inspection Standard for PWR Internals".

In some cases (as defined in MRP-227-A), physical measurements are used as supplemental techniques to manage for the gross effects of wear, loss of preload due to stress relaxation, or for changes in dimension due to void swelling, deflection or distortion. The physical measurement methods applied in accordance with this program include measurement of hold down spring height.

Inspection coverage for "Primary" and "Expansion" RVI components is implemented consistent with Sections 3.3.1 and 3.3.2 of the NRC SE, Revision 1, on MRP-227.

<u>Element 5 – Monitoring and Trending</u>

The methods for monitoring, recording, evaluating, and trending the data that result from the program's inspections are given in Section 6 of MRP-227-A and its subsections. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well as for performing applicable limit load, linear elastic and elastic-plastic fracture analyses of relevant flaw indications. The examinations and re-examinations required by the MRP-227-A guidance, together with the requirements specified in MRP-228 for inspection methodologies, inspection procedures, and inspection personnel, provide timely detection, reporting, and corrective actions with respect to the effects of the age-related degradation mechanisms within the scope of the program.

The extent of the examinations, beginning with the sample of susceptible PWR internals component locations identified as Primary Component locations, with the potential for inclusion of Expansion Component locations if the effects are greater than anticipated, plus the continuation of the Existing Programs activities, such as the ASME Code, Section XI, Examination Category B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.

Inspection Frequencies
The required inspection frequencies for Existing, Primary and Expansion Components are specified in Tables 1, 2 and 3, respectively. Specified inspection frequencies are considered adequate to manage aging effects. However more frequent inspections may be warranted based upon an internal and external OE.

Inspection Coverage
The required inspection coverage for Primary and Expansion Components are specified in Tables 2 and 3, respectively. The required inspection coverage for the Existing Program Components is as specified in the applicable program document (e.g. ASME Section XI). If the specified coverage for any of these components cannot be obtained, the condition shall be addressed in the Corrective Action Program (CAP).

<u>Element 6 – Acceptance Criteria</u>

The acceptance criteria for Primary and Expansion Components are provided in Table 4, Westinghouse Plants Examination Acceptance and Expansion Criteria Applicable to NextEra Energy Seabrook. These criteria are based upon the requirements of ASME Section XI. All detected relevant conditions must be addressed in the CAP prior to plant start-up. Possible disposition options include: 1) supplemental exams to further characterize a detected condition, 2) engineering evaluation for continued service until the next inspection, 3) repair, or 4) replacement.

Engineering evaluations for continued service shall be conducted in accordance with NRC approved methodologies. WCAP-17096, "Reactor Internals Acceptance Criteria Methodology and Data Requirements", is currently under NRC review for this purpose. The potential loss of fracture toughness must be considered in any flaw evaluations.

Expansion Criteria
The expansion criteria for expanding the scope of examination from the Primary to the linked Expansion Components, including the timing of inspections, are provided in Table 4,

Westinghouse Plants Examination Acceptance and Expansion Criteria.

It should be noted that the categorizations and associated inspection requirements described above do not replace or relieve any of the current ASME Section XI inspection requirements for the RVI components.

Element 7 – Corrective Actions

Corrective actions following the detection of unacceptable conditions are fundamentally provided for in the NexEra Energy Seabrook Corrective Action Program. Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the plant corrective action program, which may require repair, replacement, or analytical evaluation for continued service until the next inspection. The disposition will ensure that design basis functions of the reactor internals components will continue to be fulfilled for all licensing basis loads and events. Examples of methodologies that can be used to analytically disposition unacceptable conditions are found in the ASME Code, Section XI or in Section 6 of MRP-227-A. Section 6 of MRP-227-A describes the options that are available for disposition of detected conditions that exceed the examination acceptance criteria of Section 5 of the report. These include engineering evaluation methods conducted in accordance with WCAP-17096, as well as supplementary examinations to further characterize the detected condition, or the alternative of component repair and replacement procedures. The latter are subject to the requirements of the ASME Code, Section XI. The implementation of the guidance in MRP-227-A, plus the implementation of any ASME Code requirements, provides an acceptable level of aging management of safety-related components addressed in accordance with the corrective actions of 10 CFR Part 50, Appendix B, as applicable.

Element 8 – Confirmation Process

NextEra Energy Seabrook quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B, as applicable. It is expected that the implementation of the guidance in MRP-227-A will provide an acceptable level of quality for inspection, flaw evaluation, and other elements of aging management of the PWR internals that are addressed in accordance with the 10 CFR Part 50, Appendix B (as applicable), confirmation process, and administrative controls.

Element 9 – Administrative Controls

The administrative controls for License Renewal RVI AMP including the implementing procedures, review and approval processes, are under existing station 10 CFR 50 Appendix B Quality Assurance Programs. The PWR Vessel Internals AMP is established with a sufficient level of documentation and administrative controls to ensure effective long-term implementation. The implementing procedure for NextEra Energy Seabrook is Chapter 3 of SASR (Seabrook Station RCS Materials Degradation Management Reference Manual), "Reactor Vessel Internals

Aging Management Program".

Element 10 – Operating Experience

Few incidents of PWR internals aging degradation have been reported in operating U.S. commercial PWR plants. A summary of observations to date is provided in Appendix A of MRP-227-A. Operating experience gained through Industry groups such as the EPRI MRP, the PWROG, INPO, WANO and International Sites shall be evaluated and incorporated into this program as needed in a timeframe consistent with the significance. Operating experience (OE) reports are continuously reviewed by NextEra Energy Seabrook personnel to ensure relevant OE is reviewed for impact on aging effects and/or aging management programs.

NUREG-1801 Consistency

The NextEra Energy Seabrook PWR Vessel Internal Program is consistent with NUREG-1801 XI.M16A as modified by LR-ISG-2011-04, "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors".

Exceptions to NUREG-1801

None

Enhancements

None

Conclusion

The NextEra Energy Seabrook PWR Vessel Internals Program provides reasonable assurance that the aging effects will be adequately managed such that applicable components will continue to perform their intended functions consistent with the current licensing basis for the period of extended operation.

Tables

Table 1 - Westinghouse Plants Existing Programs Components Applicable to Seabrook

Table 2 - Westinghouse Plants Primary Components Applicable to Seabrook

Table 3 - Westinghouse Plants Expansion Components Applicable to Seabrook

Table 4 - Westinghouse Plants Examination, Acceptance, and Expansion Criteria

Table 1 - Westinghouse Plants Existing Program Components Applicable to NextEra Energy Seabrook

(Sheet 1 of 1)

Component (Note 2)	Applicability	Effect (Mechanism)	Reference Document	Generic Requirement Description	Examination Method and Frequency
Core Barrel Assembly Core barrel flange	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination to determine general condition for excessive wear	All accessible surfaces; one time per interval
Upper Internals Assembly Upper support ring or skirt	All plants	Cracking (SCC, fatigue)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces; one time per interval
Lower Internals Assembly Lower core plate	All plants	Cracking (SCC, IASCC, fatigue)	ASME Code Section XI	Visual (VT-3) examination of the lower core plates to detect evidence of distortion and/or loss of bolt integrity	All accessible surfaces; one time per interval
Lower Internals Assembly Lower core plate	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces; one time per interval
Bottom Mounted Instrumentation System Flux thimble tubes	Not Applicable (Note 3)	Loss of material (wear)	BMI-FTT-IP	Surface (ET) examination	ET surface examination of full length tubes at frequency specified in BMI-FTT-IP. Tube selection and frequency based upon engineering evaluation of previous examination results
Alignment and Interfacing Components Clevis insert bolt	All plants	Loss of material (wear) (Note 1)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces; one time per interval
Alignment and Interfacing Components Upper core plate alignment pins	All plants	Loss of material (wear)	ASME Code Section XI	Visual (VT-3) examination	All accessible surfaces; one time per interval

Notes:
1. Bolt was screened in because of stress relaxation and associated cracking; however, wear of the clevis/insert is the issue. Note doesn't apply to lower core plates.
2. There is no formal Program document for the GTA Support Pin Replacement Program; however, appropriate actions will be taken upon receiving further recommendation from Westinghouse. This program is not listed in this table because it does not include inspections of any RVI components.
3. NextEra Energy Seabrook does not utilize a Flux Thimble Tube Inspection Program because of the unique double wall design of the flux thimble tubes.

Table 2 - Westinghouse Plants Primary Components Applicable to NextEra Energy Seabrook

(Sheet 1 of 4)

Component	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method	Examination Coverage
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Loss of material (wear)	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period, and no earlier than two refueling outages prior to the start of the license renewal period. Subsequent examinations are required on a ten-year interval	20% examination of the number of CRGT assemblies, with all guide cards within each selected CRGT assembly examined
Control Rod Guide Tube Assembly Lower flange Welds	All plants	Cracking (SCC, fatigue) Aging Management (IE and TE)	Bottom-mounted instrumentation column bodies, Lower support column bodies (cast), Upper core plate, Lower support casting/forging	Enhanced visual (EVT-1) examination to determine the presence of crack-like surface flaws in flange welds no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of outer (accessible) lower flange weld surfaces and adjacent base metal on the individual peripheral assemblies (Note 2)
Core Barrel Assembly Upper core barrel flange weld	All plants	Cracking (SCC)	Lower support column bodies (non-cast) Core barrel outlet nozzle welds	Periodic enhanced visual (EVT-1) examination, with 10-year intervals, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)
Core Barrel Assembly Upper and lower core barrel cylinder girth welds	All plants	Cracking (SCC, IASCC, Fatigue)	Upper and lower core barrel cylinder axial welds	Periodic enhanced visual (EVT-1) examination, with 10-year intervals, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)

Table 2 - Westinghouse Plants Primary Components Applicable to NextEra Energy Seabrook

(Sheet 2 of 4)

Component	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method	Examination Coverage
Core Barrel Assembly Lower core barrel flange weld (Note 5)	All plants	Cracking (SCC, Fatigue)	None	Periodic enhanced visual (EVT-1) examination, with 10-year intervals, no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 4)
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Cracking (IASCC, fatigue) that results in • Lost or broken locking devices • Failed or missing bolts • Protrusion of bolt heads Aging Management (IE and ISR) (Note 6)	None	Visual (VT-3) examination, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval	Bolts and locking devices on high fluence seams. 100% of components accessible from core side (Note 3)
Baffle-Former Assembly Baffle-former bolts	All plants	Cracking (IASCC, fatigue) Aging Management (IE and ISR) (Note 6)	Lower support column bolts, Barrel-former bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval	100% of accessible bolts (Note 3). Heads accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs

Table 2 - Westinghouse Plants Primary Components Applicable to NextEra Energy Seabrook

(Sheet 3 of 4)

Component	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method	Examination Coverage
Baffle-Former Assembly Assembly (Includes baffle plates, baffle edge bolts and indirect effects of void swelling in former plates)	All plants	Distortion (void swelling), or cracking (IASCC) that results in • Abnormal interaction with fuel assemblies • Gaps along high fluence baffle joint • Vertical displacement of baffle plates near high fluence joint • Broken or damaged edge bolt locking systems along high fluence baffle joint	None	Visual (VT-3) examination to check for evidence of distortion, with baseline examination between 20 and 40 EFPY and subsequent examinations on a ten-year interval	Core side surface as indicated
Alignment and Interfacing Components Internals hold down spring	All plants with Type 304 stainless steel hold down springs	Distortion (loss of load) Note: This mechanism was not strictly identified in the original list of age-related degradation mechanisms	None	Direct measurement of spring height within three cycles of the beginning of the license renewal period. If the first set of measurements is not sufficient to determine life, spring height measurements must be taken during the next two outages, in order to extrapolate the expected spring height to 60 years	Measurements should be taken at several points around the circumference of the spring, with a statistically adequate number of measurements at each point to eliminate uncertainty. Replacement of Type 304 springs by Type 403 springs is required when the spring stiffness is determined to relax beyond design tolerance

Table 2 - Westinghouse Plants Primary Components Applicable to NextEra Energy Seabrook

(Sheet 4 of 4)

Component	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method	Examination Coverage
Thermal Shield Flexures	All plants with thermal shields	Cracking (fatigue) Loss of material (wear) that results in thermal shield flexures excessive wear, fracture, or complete separation	None	Visual (VT-3) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval	100% of thermal shield flexures

Notes:

1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 4. (General note applies to entire table).
2. A minimum of 75% of the total identified sample population must be examined.
3. A minimum of 75% of the total population (examined + unexamined), including coverage consistent with the Expansion criteria in Table 4, must be examined for inspection credit.
4. A minimum of 75% of the total weld length (examined + unexamined), including coverage consistent with the Expansion criteria in Table 4, must be examined from either the inner or outer diameter for inspection credit.
5. The lower core barrel flange weld may be alternatively designated as the core barrel-to-support plate weld in some Westinghouse plant designs.
6. Void swelling effects on this component is managed through management of void swelling on the entire baffle-former assembly.

Table 3 - Westinghouse Plants Expansion Components Applicable to NextEra Energy Seabrook

(Sheet 1 of 2)

Component	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Upper Internals Assembly Upper core plate	All plants	Cracking (fatigue, wear) Aging Management (IE)	CRGT lower flange weld	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection	100% of accessible surfaces (Note 2)
Lower Internals Assembly Lower support forging or casting	All plants	Cracking Aging Management (TE in casting)	CRGT lower flange weld	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection	100% of accessible surfaces (Note 2)
Core Barrel Assembly Barrel-former bolts	All plants	Cracking (IASCC, fatigue) Aging Management (IE, void swelling, and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection	100% of accessible bolts. Accessibility is limited by presence of thermal shields or neutron pads (Note 2)
Lower Support Assembly Lower support column bolts	All plants	Cracking (IASCC, fatigue) Aging Management (IE and ISR)	Baffle-former bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection	100% of accessible bolts or as supported by plant specific justification (Note 2)
Core Barrel Assembly Core barrel outlet nozzle welds	All plants	Cracking (SCC, fatigue) Aging Management (IE of lower sections)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2)
Core Barrel Assembly Upper and lower core barrel cylinder axial welds	All plants	Cracking (SCC, fatigue) Aging Management (IE)	Upper and lower core barrel cylinder girth welds	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection	100% of one side of the accessible surfaces of the selected weld and adjacent base metal (Note 2)

Table 3 - Westinghouse Plants Expansion Components Applicable to NextEra Energy Seabrook

(Sheet 2 of 2)

Component	Applicability	Effect (Mechanism)	Primary Link	Examination Method	Examination Coverage
Lower Support Assembly Lower support column bodies (non-cast)	All plants	Cracking (IASCC) Aging Management (IE)	Upper core barrel flange weld	Enhanced visual (EVT-1) examination Re-inspection every 10 years following initial inspection	100% of accessible surfaces (Note 2)
Lower Support Assembly Lower support column bodies (cast)	All plants	Cracking (IASCC) including the detection of fractured support columns Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (EVT-1) examination. Re-inspection every 10 years following initial inspection	100% of accessible support columns (Note 2)
Bottom Mounted Instrumentation System Bottom-mounted instrumentation (BMI) column bodies	All plants	Cracking (fatigue) including detection of completely fractured column bodies. Aging Management (IE)	Control rod guide tube (CRGT) lower flanges	Visual (VT-3) examination of BMI column bodies as indicated by difficulty of insertion/withdrawal of flux thimbles Re-inspection every 10 years following initial inspection Flux thimble insertion/withdrawal to be monitored at each inspection interval	100% of BMI column bodies for which difficulty is detected during flux thimble insertion/withdrawal

Notes:
1. Examination acceptance criteria and expansion criteria for the Westinghouse components are in Table 4.
2. A minimum of 75% coverage of the entire examination area or volume, or a minimum sample size of 75% of the total population of like components of the examination is required (including both the accessible and inaccessible portions).

Table 4 – Westinghouse Plants Examination, Acceptance, and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 1 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Control Rod Guide Tube Assembly Guide plates (cards)	All plants	Visual (VT-3) examination The specific relevant condition is wear that could lead to loss of control rod alignment and impede control assembly insertion	None	N/A	N/A
Control Rod Guide Tube Assembly Lower flange welds	All plants	Enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication	a. Bottom-mounted instrumentation (BMI) column bodies b. Lower support column bodies (cast), upper core plate and lower support forging or casting	a. Confirmation of surface-breaking indications in two or more CRGT lower flange welds, combined with flux thimble insertion/withdrawal difficulty, shall require visual (VT-3) examination of BMI column bodies by the completion of the next refueling outage b. Confirmation of surface-breaking indications in two or more CRGT lower flange welds shall require EVT-1 examination of cast lower support column bodies, upper core plate and lower support forging/casting within three fuel cycles following the initial observation	a. For BMI column bodies, the specific relevant condition for the VT-3 examination is completely fractured column bodies b. For cast lower support column bodies, upper core plate and lower support forging/casting, the specific relevant condition is a detectable crack-like surface indication

Table 4 – Westinghouse Plants Examination, Acceptance, and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 2 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Upper core barrel flange weld	All plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication	a. Core barrel outlet nozzle welds b. Lower support column bodies (non-cast)	a. The confirmed detection and sizing of a surface-breaking indication with a length greater than two inches in the upper core barrel flange weld shall require that the EVT-1 examination, and any supplementary UT examination, be expanded to include the core barrel outlet nozzle welds by the completion of the next refueling outage b. If extensive cracking in the core barrel outlet nozzle welds is detected, EVT-1 examination shall be expanded to include the upper six inches of the accessible surfaces of the non-cast lower support column bodies within three fuel cycles following the initial observation	a and b. The specific relevant condition for the core barrel outlet nozzle weld and lower support column body examination is a detectable crack-like surface indication

Table 4 – Westinghouse Plants Examination, Acceptance, and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 3 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Barrel Assembly Lower core barrel flange weld **(Note 2)**	All Plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication	None	None	None
Core Barrel Assembly Upper core barrel cylinder girth welds	All Plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication	Upper core barrel cylinder axial welds	The confirmed detection and sizing of a surface breaking indication with a length greater than two inches in the upper core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the upper core barrel cylinder axial welds by the completion of the next refueling outage	The specific relevant condition for the expansion upper core barrel cylinder axial weld examination is a detectable crack-like surface indication
Core Barrel Assembly Lower core barrel cylinder girth welds	All Plants	Periodic enhanced visual (EVT-1) examination The specific relevant condition is a detectable crack-like surface indication	Lower core barrel cylinder axial weld	The confirmed detection and sizing of a surface breaking indication with a length greater than two inches in the lower core barrel cylinder girth welds shall require that the EVT-1 examination be expanded to include the lower core barrel cylinder axial welds by the completion of the next refueling outage	The specific relevant condition for the expansion lower core barrel cylinder axial weld examination is a detectable crack-like surface indication

Table 4 – Westinghouse Plants Examination, Acceptance, and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 4 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Baffle-edge bolts	All plants with baffle-edge bolts	Visual (VT-3) examination The specific relevant conditions are missing or broken locking devices, failed or missing bolts, and protrusion of bolt heads	None	N/A	N/A
Baffle-Former Assembly Baffle-former bolts	All plants	Volumetric (UT) examination The examination acceptance criteria for the UT of the baffle-former bolts shall be established as part of the examination technical justification	a. Lower support column bolts b. Barrel-former bolts	a. Confirmation that more than 5% of the baffle-former bolts actually examined on the four baffle plates at the largest distance from the core (presumed to be the lowest dose locations) contain unacceptable indications shall require UT examination of the lower support column bolts within the next three fuel cycles b. Confirmation that more than 5% of the lower support column bolts actually examined contain unacceptable indications shall require UT examination of the barrel-former bolts	a and b. The examination acceptance criteria for the UT of the lower support column bolts and the barrel-former bolts shall be established as part of the examination technical justification

Table 4 – Westinghouse Plants Examination Acceptance and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 5 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Baffle-Former Assembly Assembly	All plants	Visual (VT-3) examination The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, vertical displacement of shroud plates near high fluence joints, and broken or damaged edge bolt locking systems along high fluence baffle plate joints	None	N/A	N/A
Alignment and Interfacing Components Internals hold down spring	All plants with 304 stainless steel hold down springs	Direct physical measurement of spring height The examination acceptance criterion for this measurement is that the remaining compressible height of the spring shall provide hold-down forces within the plant-specific design tolerance	None	N/A	N/A

Table 4 – Westinghouse Plants Examination Acceptance and Expansion Criteria Applicable to NextEra Energy Seabrook

(Sheet 6 of 6)

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Thermal Shield Assembly Thermal shield flexures	All plants with thermal shields	Visual (VT-3) examination The specific relevant conditions for thermal shield flexures are excessive wear, fracture, or complete separation	None	N/A	N/A

Notes:

1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).
2. The lower core barrel flange weld may alternatively be designated as the core barrel-to-support plate weld in some Westinghouse plant designs.

Enclosure 3 to SBK-L-14089

Revised AMR Items for the Reactor Vessel Internals (Revised LRA Table 3.1.2-3)

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Alignment and Interfacing components: internals hold down spring	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-33 (R-108)	3.1.1-27	A, 1
Alignment and Interfacing components: internals hold down spring	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Changes in Dimensions	PWR Vessel Internals	None	None	A, 1
Alignment and Interfacing components: internals hold down spring	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Alignment and Interfacing components: upper core plate alignment pins	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-40 (R-112)	3.1.1-37	A, 1 A, 1
Alignment and Interfacing components: upper core plate alignment pins	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	IV.B2-34 (R-115)	3.1.1-63	A, 1
Baffle-to- Former Assembly: baffle-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-10 (R-125)	3.1.1-30	A, 1 A, 1
Baffle-to- Former Assembly: baffle-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-6 (R-128)	3.1.1-22	A, 1
Baffle-to- Former Assembly: baffle-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Change in Dimensions	PWR Vessel Internals	IV.B2-4 (R-126)	3.1.1-33	A, 1
Baffle-to- Former Assembly: baffle-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-5 (R-129)	3.1.1-27	A, 1
Baffle-to- Former Assembly: baffle and former plates	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Change in Dimensions	PWR Vessel Internals	IV.B2-1 (R-124)	3.1.1-33	A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Baffle-to- Former Assembly: baffle and former plates	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-2 (R-123)	3.1.1-30	A, 1 A, 1
Baffle-to- Former Assembly: baffle-edge bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-2 (R-123)	3.1.1-30	A, 1 A, 1
Baffle-to- Former Assembly: baffle-edge bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-6 (R-128)	3.1.1-22	A, 1
Baffle-to- Former Assembly: baffle-edge bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Change in Dimensions	PWR Vessel Internals	IV.B2-4 (R-126)	3.1.1-33	A, 1
Baffle-to- Former Assembly: baffle-edge bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-5 (R-129)	3.1.1-27	A, 1
Baffle-to- Former Assembly: barrel-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-10 (R-125)	3.1.1-30	A, 1 A, 1
Baffle-to- Former Assembly: barrel-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-6 (R-128)	3.1.1-22	A, 1
Baffle-to- Former Assembly: barrel-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Change in Dimensions	PWR Vessel Internals	IV.B2-4 (R-126)	3.1.1-33	A, 1
Baffle-to- Former Assembly: barrel-to-former bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-5 (R-129)	3.1.1-27	A, 1
Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals	None	None	A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Bottom-mounted instrumentation system: bottom-mounted instrumentation (BMI) column bodies	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-22 (R-141)	3.1.1-22	A, 1
Control rod guide tube (CRGT) assemblies: CRGT guide plates (cards)	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Control rod guide tube (CRGT) assemblies: CRGT lower flange welds	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-28 (R-118)	3.1.1-37	A, 1 A, 1
Control rod guide tube (CRGT) assemblies: CRGT lower flange welds	Structural Support	Stainless Steel (including CASS)	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-22 (R-141)	3.1.1-22	A, 1
Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	Structural Support	Stainless Steel; Nickel alloy	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-28 (R-118)	3.1.1-37	A, 1 A, 1
Control rod guide tube (CRGT) assemblies: guide tube support pins (split pins)	Structural Support	Stainless Steel; Nickel alloy	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Core barrel assembly: upper core barrel and lower core barrel circumferential (girth) welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-8 (R-120)	3.1.1-30	A, 1 A, 1
Core barrel assembly: upper core barrel and lower core barrel circumferential (girth) welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-9 (R-122)	3.1.1-22	A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Core barrel assembly: upper core barrel and lower core barrel vertical (axial) welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-8 (R-120)	3.1.1-30	A, 1 A, 1
Core barrel assembly: upper core barrel and lower core barrel vertical (axial) welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-9 (R-122)	3.1.1-22	A, 1
Core barrel assembly: core barrel flange	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Core barrel assembly: core barrel outlet nozzle welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-8 (R-120)	3.1.1-30	A, 1 A, 1
Core barrel assembly: core barrel outlet nozzle welds	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-9 (R-122)	3.1.1-22	A, 1
Core barrel assembly: lower core barrel flange weld	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-8 (R-120)	3.1.1-30	A, 1 A, 1
Core barrel assembly: upper core barrel flange weld	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-8 (R-120)	3.1.1-30	A, 1 A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Lower internals assembly: clevis insert bolts or screws	Structural Support	Nickel Alloy	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Lower internals assembly: clevis insert bolts or screws	Structural Support	Nickel Alloy	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-14 (R-137)	3.1.1-27	A, 1
Lower internals assembly: clevis insert bolts or screws	Structural Support	Nickel Alloy	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-16 (R-133)	3.1.1-37	A, 1 A, 1
Lower internals assembly: lower core plate	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-20 (R-130)	3.1.1-30	A, 1 A, 1
Lower internals assembly: lower core plate	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-18 (R-132)	3.1.1-22	A, 1
Lower support assembly: lower support column bodies (cast)	Structural Support	CASS	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-24 (R-138)	3.1.1-30	A, 1 A, 1
Lower support assembly: lower support column bodies (cast)	Structural Support	CASS	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	None	None	
Lower support assembly: lower support forging or casting	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-24 (R-138)	3.1.1-30	A, 1 A, 1
Lower support assembly: lower support forging or casting	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-17 (R-135)	3.1.1-22	A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Lower support assembly: lower support column bodies (non-cast)	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-24 (R-138)	3.1.1-30	A, 1 A, 1
Lower support assembly: lower support column bodies (non-cast)	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-9 (R-122)	3.1.1-22	A, 1
Lower support assembly: lower support column bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-16 (R-133)	3.1.1-30	A, 1 A, 1
Lower support assembly: lower support column bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Fracture Toughness	PWR Vessel Internals	IV.B2-17 (R-135)	3.1.1-22	A, 1
Lower support assembly: lower support column bolts	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Preload	PWR Vessel Internals	IV.B2-25 (R-136)	3.1.1-27	A, 1
Reactor vessel internal components	Direct Flow Structural Support	Stainless Steel; Nickel Alloy	Reactor Coolant and Neutron Flux	Cumulative fatigue damage	TLAA	IV.B2-31 (R-53)	3.1.1-5	A, 1
Reactor vessel internal components	Direct Flow Structural Support	Stainless Steel; Nickel Alloy	Reactor Coolant and Neutron Flux	Loss of Material	Water Chemistry	IV.B2-32 (RP-24)	3.1.1-83	A, 1
Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components (not already identified as "Existing Programs" components in MRP-227-A)	Direct Flow Structural Support	Stainless Steel; Nickel Alloy	Reactor Coolant and Neutron Flux	Cracking	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	IV.B2-26 (R-142)	None	A, 1

Table 3.1.2-3

REACTOR VESSEL INTERNALS

Summary of Aging Management Evaluation

Component Type	Intended Function	Material	Environment	Aging Effect Requiring Management	Aging Management Program	NUREG 1801 Vol. 2 Item	Table 3.X.1 Item	Note
Reactor vessel internals: ASME Section XI, Examination Category B-N-3 core support structure components (not already identified as "Existing Programs" components in MRP-227-A)	Direct Flow Structural Support	Stainless Steel; Nickel Alloy	Reactor Coolant and Neutron Flux	Loss of Material	ASME Section XI Inservice Inspection, Subsections IWB, IWC, and IWD	IV.B2-26 (R-142)	None	A, 1
Thermal shield assembly: thermal shield flexures	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals	None	None	A, 1
Thermal shield assembly: thermal shield flexures	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Reactor internal "No Additional Measures" components	Direct Flow Structural Support	Stainless steel; Nickel alloy	Reactor Coolant and Neutron Flux	No additional aging management for reactor internal "No Additional Measures" components unless required by ASME Section XI, Examination Category B-N-3 or relevant operating experience exists	PWR Vessel Internals	None	None	A, 1
Upper Internals Assembly; upper core plate	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals	None	None	A, 1
Upper Internals Assembly; upper core plate	Direct Flow Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Loss of Material	PWR Vessel Internals	None	None	A, 1
Upper Internals Assembly: upper support ring or skirt	Structural Support	Stainless Steel	Reactor Coolant and Neutron Flux	Cracking	PWR Vessel Internals Water Chemistry	IV.B2-42 (R-106)	3.1.1-30	A, 1 A, 1

Standard Notes:

A	Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.
B	Consistent with NUREG-1801 item for component, material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP.
C	Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP is consistent with NUREG-1801 AMP.
D	Component is different, but consistent with NUREG-1801 item for material, environment, and aging effect. AMP takes some exceptions to NUREG-1801 AMP
E	Consistent with NUREG-1801 for material, environment and aging effect, but a different aging management program is credited or NUREG-1801 identifies a plant-specific aging management program
F	Material not in NUREG-1801 for this component.
G	Environment not in NUREG-1801 for this component and material.
H	Aging effect not in NUREG-1801 for this component, material and environment combination.
I	Aging effect in NUREG-1801 for this component, material and environment combination is not applicable.
J	Neither the component nor the material and environment combination is evaluated in NUREG-1801.

Plant Specific Notes:

1	Consistent with NUREG-1801 as modified by LR-ISG-2011-04.

<u>Enclosure 4 to SBK-L-14089</u>

LRA Appendix A - Final Safety Report Supplement Table A.3,

License Renewal Commitment List Updated to Reflect Changes to Date

A.3 LICENSE RENEWAL COMMITMENT LIST

No.	PROGRAM or TOPIC	COMMITMENT	UFSAR LOCATION	SCHEDULE
1.	PWR Vessel Internals	~~An inspection plan for Reactor Vessel Internals will be submitted for NRC review and approval.~~ ***Provide confirmation and acceptability of the implementation of MRP-227-A by addressing the plant-specific Applicant/Licensee Action Items outlined in section 4.2 of the NRC SER.***	A.2.1.7	~~Program to be implemented prior to the period of extended operation. Inspection plan to be submitted to NRC not later than 2 years after receipt of the renewed license or not less than 24 months prior to the period of extended operation, whichever comes first.~~ ***NextEra Energy Seabrook will provide a submittal schedule by October 15, 2014.***
2.	Closed-Cycle Cooling Water	Enhance the program to include visual inspection for cracking, loss of material and fouling when the in-scope systems are opened for maintenance.	A.2.1.12	Prior to the period of extended operation.
3.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Enhance the program to monitor general corrosion on the crane and trolley structural components and the effects of wear on the rails in the rail system.	A.2.1.13	Prior to the period of extended operation.
4.	Inspection of Overhead Heavy Load and Light Load (Related to Refueling) Handling Systems	Enhance the program to list additional cranes for monitoring.	A.2.1.13	Prior to the period of extended operation.
5.	Compressed Air Monitoring	Enhance the program to include an annual air quality test requirement for the Diesel Generator compressed air sub system.	A.2.1.14	Prior to the period of extended operation.
6.	Fire Protection	Enhance the program to perform visual inspection of penetration seals by a fire protection qualified inspector.	A.2.1.15	Prior to the period of extended operation.

7.	Fire Protection	Enhance the program to add inspection requirements such as spalling, and loss of material caused by freeze-thaw, chemical attack, and reaction with aggregates by qualified inspector.	A.2.1.15	Prior to the period of extended operation.
8.	Fire Protection	Enhance the program to include the performance of visual inspection of fire-rated doors by a fire protection qualified inspector.	A.2.1.15	Prior to the period of extended operation.
9.	Fire Water System	Enhance the program to include NFPA 25 (2011 Edition) guidance for "where sprinklers have been in place for 50 years, they shall be replaced or representative samples from one or more sample areas shall be submitted to a recognized testing laboratory for field service testing".	A.2.1.16	Prior to the period of extended operation.
10.	Fire Water System	Enhance the program to include the performance of periodic flow testing of the fire water system in accordance with the guidance of NFPA 25 (2011 Edition).	A.2.1.16	Prior to the period of extended operation.
11.	Fire Water System	Enhance the program to include the performance of periodic visual or volumetric inspection of the internal surface of the fire protection system upon each entry to the system for routine or corrective maintenance to evaluate wall thickness and inner diameter of the fire protection piping ensuring that corrosion product buildup will not result in flow blockage due to fouling. Where surface irregularities are detected, follow-up volumetric examinations are performed. These inspections will be documented and trended to determine if a representative number of inspections have been performed prior to the period of extended operation. If a representative number of inspections have not been performed prior to the period of extended operation, focused inspections will be conducted. These inspections will be performed within ten years prior to the period of extended operation.	A.2.1.16	Within ten years prior to the period of extended operation.
12.	Aboveground Steel Tanks	Enhance the program to include components and aging effects required by the Aboveground Steel Tanks and to perform visual, surface, and volumetric examinations of the outside and inside surfaces for managing the aging effects of loss of material and cracking.	A.2.1.17	~~Prior to the period of extended operation.~~ Within 10 years prior to the period of extended operation.

13.	~~Aboveground Steel Tanks~~ Fire Water System	Enhance the program to perform exterior inspection of the fire water storage tanks annually for signs of degradation and include an ultrasonic inspection and evaluation of the internal bottom surface of the two Fire Protection Water Storage Tanks per the guidance provided in NFPA 25 (2011 Edition).	~~A.2.1.17~~ A.2.1.16	Within ten years prior to the period of extended operation.
14.	Fuel Oil Chemistry	Enhance program to add requirements to 1) sample and analyze new fuel deliveries for biodiesel prior to offloading to the Auxiliary Boiler fuel oil storage tank and 2) periodically sample stored fuel in the Auxiliary Boiler fuel oil storage tank.	A.2.1.18	Prior to the period of extended operation.
15.	Fuel Oil Chemistry	Enhance the program to add requirements to check for the presence of water in the Auxiliary Boiler fuel oil storage tank at least once per quarter and to remove water as necessary.	A.2.1.18	Prior to the period of extended operation.
16.	Fuel Oil Chemistry	Enhance the program to require draining, cleaning and inspection of the diesel fire pump fuel oil day tanks on a frequency of at least once every ten years.	A.2.1.18	Prior to the period of extended operation.
17.	Fuel Oil Chemistry	Enhance the program to require ultrasonic thickness measurement of the tank bottom during the 10-year draining, cleaning and inspection of the Diesel Generator fuel oil storage tanks, Diesel Generator fuel oil day tanks, diesel fire pump fuel oil day tanks and auxiliary boiler fuel oil storage tank.	A.2.1.18	Prior to the period of extended operation.
18.	Reactor Vessel Surveillance	Enhance the program to specify that all pulled and tested capsules, unless discarded before August 31, 2000, are placed in storage.	A.2.1.19	Prior to the period of extended operation.
19.	Reactor Vessel Surveillance	Enhance the program to specify that if plant operations exceed the limitations or bounds defined by the Reactor Vessel Surveillance Program, such as operating at a lower cold leg temperature or higher fluence, the impact of plant operation changes on the extent of Reactor Vessel embrittlement will be evaluated and the NRC will be notified.	A.2.1.19	Prior to the period of extended operation.
20.	Reactor Vessel Surveillance	Enhance the program as necessary to ensure the appropriate withdrawal schedule for capsules remaining in the vessel such that one capsule will be withdrawn at an outage in which the capsule receives a neutron fluence that meets the schedule requirements of 10 CFR 50 Appendix H and ASTM E185-82 and that bounds the 60-year fluence, and the remaining capsule(s) will be removed from the vessel unless determined to provide meaningful metallurgical data.	A.2.1.19	Prior to the period of extended operation.

21.	Reactor Vessel Surveillance	Enhance the program to ensure that any capsule removed, without the intent to test it, is stored in a manner which maintains it in a condition which would permit its future use, including during the period of extended operation.	A.2.1.19	Prior to the period of extended operation.
22.	One-Time Inspection	Implement the One Time Inspection Program.	A.2.1.20	Within ten years prior to the period of extended operation.
23.	Selective Leaching of Materials	Implement the Selective Leaching of Materials Program. The program will include a one-time inspection of selected components where selective leaching has not been identified and periodic inspections of selected components where selective leaching has been identified.	A.2.1.21	Within five years prior to the period of extended operation.
24.	Buried Piping And Tanks Inspection	Implement the Buried Piping And Tanks Inspection Program.	A.2.1.22	Within ten years prior to entering the period of extended operation
25.	One-Time Inspection of ASME Code Class 1 Small Bore-Piping	Implement the One-Time Inspection of ASME Code Class 1 Small Bore-Piping Program.	A.2.1.23	Within ten years prior to the period of extended operation.
26.	External Surfaces Monitoring	Enhance the program to specifically address the scope of the program, relevant degradation mechanisms and effects of interest, the refueling outage inspection frequency, ~~the inspections of opportunity for possible corrosion under insulation,~~ the training requirements for inspectors and the required periodic reviews to determine program effectiveness.	A.2.1.24	Prior to the period of extended operation.
27.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Implement the Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components Program.	A.2.1.25	Prior to the period of extended operation.
28.	Lubricating Oil Analysis	Enhance the program to add required equipment, lube oil analysis required, sampling frequency, and periodic oil changes.	A.2.1.26	Prior to the period of extended operation.
29.	Lubricating Oil Analysis	Enhance the program to sample the oil for the Reactor Coolant pump oil collection tanks.	A.2.1.26	Prior to the period of extended operation.
30.	Lubricating Oil Analysis	Enhance the program to require the performance of a one-time ultrasonic thickness measurement of the lower portion of the Reactor Coolant pump oil collection tanks prior to the period of extended operation.	A.2.1.26	Prior to the period of extended operation.

31.	ASME Section XI, Subsection IWL	Enhance procedure to include the definition of "Responsible Engineer".	A.2.1.28	Prior to the period of extended operation.
32.	Structures Monitoring Program	Enhance procedure to add the aging effects, additional locations, inspection frequency and ultrasonic test requirements.	A.2.1.31	Prior to the period of extended operation.
33.	Structures Monitoring Program	Enhance procedure to include inspection of opportunity when planning excavation work that would expose inaccessible concrete.	A.2.1.31	Prior to the period of extended operation.
34.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	A.2.1.32	Prior to the period of extended operation.
35.	Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits	Implement the Electrical Cables and Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements Used in Instrumentation Circuits program.	A.2.1.33	Prior to the period of extended operation.
36.	Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Inaccessible Power Cables Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	A.2.1.34	Prior to the period of extended operation.
37.	Metal Enclosed Bus	Implement the Metal Enclosed Bus program.	A.2.1.35	Prior to the period of extended operation.
38.	Fuse Holders	Implement the Fuse Holders program.	A.2.1.36	Prior to the period of extended operation.

39.	Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements	Implement the Electrical Cable Connections Not Subject to 10 CFR 50.49 Environmental Qualification Requirements program.	A.2.1.37	Prior to the period of extended operation.
40.	345 KV SF6 Bus	Implement the 345 KV SF6 Bus program.	A.2.2.1	Prior to the period of extended operation.
41.	Metal Fatigue of Reactor Coolant Pressure Boundary	Enhance the program to include additional transients beyond those defined in the Technical Specifications and UFSAR.	A.2.3.1	Prior to the period of extended operation.
42.	Metal Fatigue of Reactor Coolant Pressure Boundary	Enhance the program to implement a software program, to count transients to monitor cumulative usage on selected components.	A.2.3.1	Prior to the period of extended operation.
43.	Pressure –Temperature Limits, including Low Temperature Overpressure Protection Limits	Seabrook Station will submit updates to the P-T curves and LTOP limits to the NRC at the appropriate time to comply with 10 CFR 50 Appendix G.	A.2.4.1.4	The updated analyses will be submitted at the appropriate time to comply with 10 CFR 50 Appendix G, Fracture Toughness Requirements.

44.	Environmentally-Assisted Fatigue Analyses (TLAA)	NextEra Seabrook will perform a review of design basis ASME Class 1 component fatigue evaluations to determine whether the NUREG/CR-6260-based components that have been evaluated for the effects of the reactor coolant environment on fatigue usage are the limiting components for the Seabrook plant configuration. If more limiting components are identified, the most limiting component will be evaluated for the effects of the reactor coolant environment on fatigue usage. If the limiting location identified consists of nickel alloy, the environmentally-assisted fatigue calculation for nickel alloy will be performed using the rules of NUREG/CR-6909. (1) Consistent with the Metal Fatigue of Reactor Coolant Pressure Boundary Program Seabrook Station will update the fatigue usage calculations using refined fatigue analyses, if necessary, to determine acceptable CUFs (i.e., less than 1.0) when accounting for the effects of the reactor water environment. This includes applying the appropriate Fen factors to valid CUFs determined from an existing fatigue analysis valid for the period of extended operation or from an analysis using an NRC-approved version of the ASME code or NRC-approved alternative (e.g., NRC-approved code case). (2) If acceptable CUFs cannot be demonstrated for all the selected locations, then additional plant-specific locations will be evaluated. For the additional plant-specific locations, if CUF, including environmental effects is greater than 1.0, then Corrective Actions will be initiated, in accordance with the Metal Fatigue of Reactor Coolant Pressure Boundary Program, B.2.3.1. Corrective Actions will include inspection, repair, or replacement of the affected locations before exceeding a CUF of 1.0 or the effects of fatigue will be managed by an inspection program that has been reviewed and approved by the NRC (e.g., periodic non-destructive examination of the affected locations at inspection intervals to be determined by a method accepted by the NRC).	A.2.4.2.3	At least two years prior to entering the period of extended operation.

45.	Number Not Used			
46.	Protective Coating Monitoring and Maintenance	Enhance the program by designating and qualifying an Inspector Coordinator and an Inspection Results Evaluator.	A.2.1.38	Prior to the period of extended operation.
47.	Protective Coating Monitoring and Maintenance	Enhance the program by including, "Instruments and Equipment needed for inspection may include, but not be limited to, flashlight, spotlights, marker pen, mirror, measuring tape, magnifier, binoculars, camera with or without wide angle lens, and self sealing polyethylene sample bags."	A.2.1.38	Prior to the period of extended operation.
48.	Protective Coating Monitoring and Maintenance	Enhance the program to include a review of the previous two monitoring reports.	A.2.1.38	Prior to the period of extended operation.
49.	Protective Coating Monitoring and Maintenance	Enhance the program to require that the inspection report is to be evaluated by the responsible evaluation personnel, who is to prepare a summary of findings and recommendations for future surveillance or repair.	A.2.1.38	Prior to the period of extended operation.
50.	ASME Section XI, Subsection IWE	Perform UT testing of the containment liner plate in the vicinity of the moisture barrier for loss of material.	A.2.1.27	Within the next two refueling outages, OR15 or OR16, and repeated at intervals of no more than five refueling outages.
51.	Number Not Used			
52.	ASME Section XI, Subsection IWL	Implement measures to maintain the exterior surface of the Containment Structure, from elevation -30 feet to +20 feet, in a dewatered state.	A.2.1.28	Ongoing
53.	Reactor Head Closure Studs	Replace the spare reactor head closure stud(s) manufactured from the bar that has a yield strength > 150 ksi with ones that do not exceed 150 ksi.	A.2.1.3	Prior to the period of extended operation.

54.	Steam Generator Tube Integrity	NextEra will address the potential for cracking of the primary to secondary pressure boundary due to PWSCC of tube-to-tubesheet welds using one of the following two options: 1) Perform a one-time inspection of a representative sample of tube-to-tubesheet welds in all steam generators to determine if PWSCC cracking is present and, if cracking is identified, resolve the condition through engineering evaluation justifying continued operation or repair the condition, as appropriate, and establish an ongoing monitoring program to perform routine tube-to-tubesheet weld inspections for the remaining life of the steam generators, or 2) Perform an analytical evaluation showing that the structural integrity of the steam generator tube-to-tubesheet interface is adequately maintaining the pressure boundary in the presence of tube-to-tubesheet weld cracking, or redefining the pressure boundary in which the tube-to-tubesheet weld is no longer included and, therefore, is not required for reactor coolant pressure boundary function. The redefinition of the reactor coolant pressure boundary must be approved by the NRC as part of a license amendment request.	A.2.1.10	Complete
55.	Steam Generator Tube Integrity	Seabrook will perform an inspection of each steam generator to assess the condition of the divider plate assembly.	A.2.1.10	Within five years prior to entering the period of extended operation.
56.	Closed-Cycle Cooling Water System	Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for hydrazine and sulfates.	A.2.1.12	Prior to entering the period of extended operation.
57.	Closed-Cycle Cooling Water System	Revise the station program documents to reflect the EPRI Guideline operating ranges and Action Level values for Diesel Generator Cooling Water Jacket pH.	A.2.1.12	Prior to entering the period of extended operation.
58.	Fuel Oil Chemistry	Update Technical Requirement Program 5.1, (Diesel Fuel Oil Testing Program) ASTM standards to ASTM D2709-96 and ASTM D4057-95 required by the GALL XI.M30 Rev 1	A.2.1.18	Prior to the period of extended operation.
59.	Nickel Alloy Nozzles and Penetrations	The Nickel Alloy Aging Nozzles and Penetrations program will implement applicable Bulletins, Generic Letters, and staff accepted industry guidelines.	A.2.2.3	Prior to the period of extended operation.
60.	Buried Piping and Tanks Inspection	Implement the design change replacing the buried Auxiliary Boiler supply piping with a pipe-within-pipe configuration with leak detection capability.	A.2.1.22	Prior to entering the period of extended operation.

61.	Compressed Air Monitoring Program	Replace the flexible hoses associated with the Diesel Generator air compressors on a frequency of every 10 years.	A.2.1.14	Within ten years prior to entering the period of extended operation.
62.	Water Chemistry	Enhance the program to include a statement that sampling frequencies are increased when chemistry action levels are exceeded.	A.2.1.2	Prior to the period of extended operation.
63.	Flow Induced Erosion	Ensure that the quarterly CVCS Charging Pump testing is continued during the PEO. Additionally, add a precaution to the test procedure to state that an increase in the CVCS Charging Pump mini flow above the acceptance criteria may be indicative of erosion of the mini flow orifice as described in LER 50-275/94-023.	N/A	Prior to the period of extended operation.
64.	Buried Piping and Tanks Inspection	Soil analysis shall be performed prior to entering the period of extended operation to determine the corrosivity of the soil in the vicinity of non-cathodically protected steel pipe within the scope of this program. If the initial analysis shows the soil to be non-corrosive, this analysis will be re-performed every ten years thereafter.	A.2.1.22	Prior to entering the period of extended operation.
65.	Flux Thimble Tube	Implement measures to ensure that the movable incore detectors are not returned to service during the period of extended operation.	N/A	Prior to entering the period of extended operation.
66.	Number Not Used			
67.	Structures Monitoring Program	Perform one shallow core bore in an area that was continuously wetted from borated water to be examined for concrete degradation and also expose rebar to detect any degradation such as loss of material. The removed core will also be subjected to petrographic examination for concrete degradation due to ASR per ASTM Standard Practice C856.	A.2.1.31	No later than December 31, 2015.
68.	Structures Monitoring Program	Perform sampling at the leakoff collection points for chlorides, sulfates, pH and iron once every three months.	A.2.1.31	Quarterly Preventive Maintenance Activity Implemented
69.	Open-Cycle Cooling Water System	Replace the Diesel Generator Heat Exchanger Plastisol PVC lined Service Water piping with piping fabricated from AL6XN material.	A.2.1.11	Prior to the period of extended operation.
70.	Closed-Cycle Cooling Water System	Inspect the piping downstream of CC-V-444 and CC-V-446 to determine whether the loss of material due to cavitation induced erosion has been eliminated or whether this remains an issue in the primary component cooling water system.	A.2.1.12	Within ten years prior to the period of extended operation.

71.	Alkali-Silica Reaction (ASR) Monitoring Program	Implement the Alkali-Silica Reaction (ASR) Monitoring Program. Testing will be performed to confirm that parameters being monitored and acceptance criteria used are appropriate to manage the effects of ASR.	A.2.1.31A	Prior to entering the period of extended operation.
72.	Flow-Accelerated Corrosion	Enhance the program to include management of wall thinning caused by mechanisms other than FAC.	A.2.1.8	Prior to entering the period of extended operation.
73.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Enhance the program to include performance of focused examinations to provide a representative sample of 20%, or a maximum of 25, of each identified material, environment, and aging effect combinations during each 10 year period in the period of extended operation.	A.2.1.25	Prior to entering the period of extended operation.
74.	Fire Water System	Enhance the program to perform sprinkler inspections annually per the guidance provided in NFPA 25 (2011 Edition). Inspection will ensure that sprinklers are free of corrosion, foreign materials, paint, and physical damage and installed in the proper orientation (e.g., upright, pendant, or sidewall). Any sprinkler that is painted, corroded, damaged, loaded, or in the improper orientation, and any glass bulb sprinkler where the bulb has emptied, will be evaluated for replacement.	A.2.1.16	Within ten years prior to the period of extended operation.
75.	Fire Water System	Enhance the program to conduct an inspection of piping and branch line conditions every 5 years by opening a flushing connection at the end of one main and by removing a sprinkler toward the end of one branch line for the purpose of inspecting for the presence of foreign organic and inorganic material per the guidance provided in NFPA 25 (2011 Edition).	A.2.1.16	Within ten years prior to the period of extended operation.
76.	Fire Water System	Enhance the Program to conduct the following activities annually per the guidance provided in NFPA 25 (2011 Edition). • main drain tests • deluge valve trip tests • fire water storage tank exterior surface inspections	A.2.1.16	Within ten years prior to the period of extended operation.

77.	Fire Water System	The Fire Water System Program will be enhanced to include the following requirements related to the main drain testing per the guidance provided in NFPA 25 (2011 Edition). • The requirement that if there is a 10 percent reduction in full flow pressure when compared to the original acceptance tests or previously performed tests, the cause of the reduction shall be identified and corrected if necessary. • Recording the time taken for the supply water pressure to return to the original static (nonflowing) pressure.	A.2.1.16	Within ten years prior to the period of extended operation.
78.	External Surfaces Monitoring	Enhance the program to include periodic inspections of in-scope insulated components for possible corrosion under insulation.	A.2.1.8	Prior to the period of extended operation.
79.	Open-Cycle Cooling Water System	Enhance the program to include visual inspection of Service Level III (augmented) internal coatings for loss of coating integrity.	A.2.1.11	Within 10 years prior to the period of extended operation.
80.	Fire Water System	Enhance the program to include visual inspection of Service Level III (augmented) internal coatings for loss of coating integrity.	A.2.1.16	Within 10 years prior to the period of extended operation.
81.	Fuel Oil Chemistry	Enhance the program to include visual inspection of Service Level III (augmented) internal coatings for loss of coating integrity.	A.2.1.18	Within 10 years prior to the period of extended operation.
82.	Inspection of Internal Surfaces in Miscellaneous Piping and Ducting Components	Enhance the program to include visual inspection of Service Level III (augmented) internal coatings for loss of coating integrity.	A.2.1.25	Within 10 years prior to the period of extended operation.

83.	Alkali-Silica Reaction Monitoring	Install instrumentation in representative sample areas of structures to monitor expansion due to alkali-silica reaction in the out-of-plane direction. Evaluate instrument and pin expansion data under the Operating Experience Element of the Alkali-Silica Reaction Monitoring Program to determine whether there is a need to enhance the program to monitor expansion in the out-of-plane direction. If the evaluation concludes that out-of-plane monitoring is necessary, establish acceptance criteria and monitoring frequencies for expansion in the out-of-plane direction using the instrument and pin expansion data.	A.2.1.31A	Prior to the period of extended operation.
84.	ASME Section XI, Subsection IWL	Evaluate the acceptability of inaccessible areas for structures within the scope of ASME Section XI, Subsection IWL Program.	A.2.1.28	Prior to the period of extended operation.