FPL.

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St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389

License Renewal Commitments
Reactor Vessel Internals Aging Management Plan

References:

1. NUREG 1779, Safety Evaluation Report Related to License Renewal of St. Lucie Nuclear Plant, Units 1 and 2, September 2003.

2. Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 213 to Facility Operating License No. DPR-67, Florida Power and Light Company, St. Lucie Plant Unit No. 1, Docket No. 50-335.

3. Safety Evaluation by the Office of Nuclear Reactor Regulation related to Amendment No. 163 to Facility Operating License No. NPF-16, Florida Power and Light Company, St. Lucie Plant Unit No. 2, Docket No. 50-389.

4. FPL Letter from Joseph Jensen to U.S. Nuclear Regulatory Commission (L-2014-192) "St. Lucie Units 1 and 2 Docket Nos. 50-335 and 50-389, Reactor Vessel Internals Inspection Program Plans and Inspection Dates," June 25, 2014.

The following License Renewal (LR) commitments have been made regarding the St. Lucie Units 1 and 2 Reactor Vessel Internals (RVI) Inspection Program to be implemented during the period of extended operation (PEO). Each of these commitments and the manner in which it has been addressed is described below.

- Commitment No. 4 of NUREG 1779 (Reference 1), the Safety Evaluation Report (SER) for the renewed operating licenses of St. Lucie Units 1 and 2, requires the submission of a report summarizing the aging effects applicable to the Reactor Vessel Internals (RVI), including a description of the inspection plan prior to the end of the initial period of operation for St. Lucie Unit 1.

  FPL's response:
  As discussed in Reference 4, the RVI inspection plan for St. Lucie Unit 1 is scheduled for submittal to the NRC by September 30, 2015 and the RVI inspection plan for St. Lucie Unit 2 would be submitted at a later date. The attached RVI Aging Management Plan summarizes the St. Lucie Units 1 and 2 RVI Inspection Program and provides the age related degradation effects applicable to the RVI components, the schedule of inspections to be performed and the acceptance criteria.

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- Commitment No. 5 of NUREG 1779 requires that FPL perform a one-time inspection of the reactor vessel internals.

  FPL's response:
  Reference 4 discussed and reaffirmed FPL's adoption of MRP-227-A which requires the implementation of periodic inspections for both St. Lucie Unit 1 and 2, and supersedes the prior commitment for a one-time inspection. As also discussed in Reference 4, the first inspection of St. Lucie Unit 1 RVI is currently scheduled for the Spring Outage of 2018. The first inspection of St. Lucie Unit 2 RVI will be scheduled within 3 years after PEO.

- Commitment No. 12 of the SER for the Extended Power Uprate License Amendment of St. Lucie Unit 1 (Reference 2) and the fourth in a series of commitments of the SER for the Extended Power Uprate License Amendment of St. Lucie Unit 2 (Reference 3) require that FPL adopt MRP-227-A in place of its previously approved RVI Inspection Program.

  FPL's response:
  The attached RVI Aging Management Plan summarizes the revised St. Lucie Units 1 and 2 RVI Inspection Program which is based upon MRP-227-A.

Should you have any questions, please contact Mr. Eric Katzman, Licensing Manager, at 772-467-7734.

Very truly yours,

Christopher R Costanzo

Christopher Costanzo
Site Vice President
St. Lucie Plant

Attachments: 1) St. Lucie Units 1 and 2 RVI Aging Management Plan
2) Proposed St. Lucie Units 1 and 2 UFSAR Revisions

cc: USNRC Regional Administrator, Region II
USNRC Project Manager, St. Lucie Nuclear Plant
USNRC Senior Resident Inspector, St. Lucie Nuclear Plant

Attachment 1 to Letter L-2015-229

St. Lucie Units 1 and 2 RVI Aging Management Plan

1 SUMMARY OF RVI INSPECTION PROGRAM

The RVI Inspection Program was developed utilizing the EPRI MRP-227-A, "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." Applicability of MRP-227-A is demonstrated by the responses to the Licensee Action Items (LAI) in Section 3. The methodology of MRP-227-A is described below.

1.1 DEGRADATION MECHANISMS

A total of eight age related degradation mechanisms are considered applicable to the RVI: 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; and 8) irradiation and thermal enhanced stress relaxation/creep. A brief description of these degradation mechanisms and the associated aging effects 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 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.

1.2 COMPONENT CATEGORIZATION

The RVI components were screened for susceptibility to the eight degradation mechanisms based upon their chemical compositions, neutron fluence exposures, operating temperatures and stress levels. Functionality assessments were then performed on the screened-in components to determine the effects of the applicable degradation mechanism(s) on functionality. Each of the RVI components was then categorized as an Existing Program, Primary, Expansion or No Additional Measurements Component based upon the functionality analysis, component accessibility, operating history, existing evaluations and prior examination results. A description of the component categories follows:

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 1, CE Plants Primary Components

Expansion Components

Expansion Components are highly or moderately susceptible to at least one of the eight degradation mechanisms, but exhibit a high degree of tolerance to the associated aging effect(s). 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 2, CE Plants Expansion Components.

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 3, Existing Programs Components.

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.

1.3 INSPECTION OF RVI COMPONENTS

Inspections detailed in Table 1, CE Plants Primary Components, and Table 3, CE Plants Existing Program Components, are required to manage aging effects in Primary Components. Additionally, inspections detailed in Table 2, CE Plants Expansion Components, are required should evidence of aging degradation be detected in linked Primary Components.

Inspection Methodologies

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:

- Direct physical measurements to monitor for loss of material or preload
- VT-3 exams to monitor for general degradation associated with loss of material or preload
- EVT-1 exams to monitor for surface breaking linear discontinuities indicative of cracking
- UT exams to monitor directly for cracking
- ECT to further characterize conditions detected by visual (VT-3, VT-1 and EVT-1) exams

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 .

Inspection Frequencies

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 is specified in Tables 1 and 2, respectively. If the specified coverage cannot be obtained, the condition shall be addressed in the Corrective Action Program (CAP).

Acceptance Criteria

The acceptance criteria for Primary and Expansion Components are provided in Table 4, CE Plants Examination Acceptance and Expansion Criteria. 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, described in WCAP-17096-NP-A, "Reactor Internals Acceptance Criteria Methodology and Data Requirements". The potential loss of fracture toughness must be considered in any flaw evaluations.

Additionally, plant specific acceptance criteria have been developed for the core shroud gap measurements, should they be required. The allowable gap size to insure continued functionality is based upon design and as-built conditions, fluence, circumferential bounds of the gap (how far around the core shroud can the gap exist),stress, impact on adjacent reactor vessel internals components, impact on core and bypass flow rates, and potential effects on fuel management schemes.

Expansion Components

The criteria for expanding the scope of examination from the Primary to the linked Expansion Components are also provided in Table 4. Generally, the inspection of the Expansion Components is required in the RFO following that in which degradation of the linked Primary Component was detected.

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

2 INSPECTION PROGRAM ATTRIBUTES

The attributes of the St. Lucie RVI Inspection Program and compliance with NUREG-1801 (GALL Report), Section XI.M16, "PWR Vessel Internals" are described in this section. The GALL identifies 10 attributes for successful component aging management. The framework for assessing the effectiveness of the projected program is established by the use of the 10 elements of the GALL.

	Plan Attribute	Approach and supplemental information
1	Scope of Program	The St. Lucie RVI Inspection Program includes all Units 1 and 2 RVI components which were built to the CE NSSS design. Using the guidance provided in MRP-227-A, the St. Lucie RVI Inspection Program was developed to manage the aging of these components during the initial and extended periods of operation. Components considered for inspection under MRP-227-A include core support structures, RVI components that serve an intended license renewal safety function pursuant to criteria in 10CFR54.4(a)(1), and other RVI components whose failure could prevent satisfactory accomplishment of any other functions identified in 10 CFR 54.4(a)(i), (ii), or (iii). The program does not included consumable items, such as fuel assemblies, reactivity control assemblies, and nuclear instrumentation. 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 by the St. Lucie Reactor Vessel Integrity Program and AMP.
2	Preventive Measures	The St. Lucie Chemistry Control Program is credited for limiting the levels of corrosive chemical species (e.g. halogens, sulfur compounds, oxygen) in the RCS to extremely low levels as a preventative measure for corrosion related degradation mechanisms including pitting, crevice corrosion, SCC, PWSCC and IASCC.
3	Parameters Monitored	The St. Lucie RVI Inspection Program manages the following age-related degradation effects and mechanisms: 1) cracking induced by SCC, PWSCC, IASCC, or fatigue; 2) loss of material induced by wear; 3) loss of fracture toughness induced by either thermal aging or neutron irradiation embrittlement; 4) changes in dimension due to void swelling and irradiation growth, distortion or defection; and 5) loss of preload caused by thermal and irradiation-enhanced stress relaxation or creep. For the management of cracking, the St. Lucie RVI Inspection Program monitors for evidence of surface breaking linear discontinuities using visual (EVT-1) exams, or directly using volumetric (UT) or surface (ECT) exams. For the management of loss of material, the RVI Inspection Program monitors for surface conditions that may be indicative of wear using visual (VT-3) exams. For the management of changes in dimension and loss of preload, the RVI Inspection Program monitors for gross surface conditions using visual (VT-3) exams or direct physical measurements. The RVI Inspection Program does not directly monitor for loss of fracture toughness but relies on visual or volumetric examination techniques to monitor for cracking in components. Specifically, the St. Lucie RVI Inspection Program implements the parameters monitored/inspected criteria for CE Designed Primary Components in Table 4-2 of MRP-227-A. Additionally, the program implements the parameters monitored/inspected criteria for CE designed Expansion Components in Table 4-5 of MRP-227-A. The parameters monitored/inspected for Existing Program Components follow the bases

	Plan Attribute	Approach and supplemental information
		for the ASME Section XI Program.
4	Detection of Aging Effects	Discussion and justification of the inspection methods selected for detection of the aging effects managed by the St. Lucie RVI Inspection Program are provided in MRP-227-A and MRP-228. In all cases, well established methods described above were selected. Additionally, the RVI Inspection Program adopts the recommended guidance in MRP-227-A for defining Expansion criteria that need to be applied to inspections of Primary and Existing Program Components and for expanding the examinations to include additional Expansion Components. As a result, inspections performed on the RVI components are in conformance with the inspection criteria, sampling basis criteria and sample Expansion criteria in Section A.1.2.3.4 of NRC Branch Position RLSB-1. Specifically, the St. Lucie RVI Inspection Program implements the parameters monitored/inspected criteria and bases for inspecting the relevant parameter conditions for CE designed Primary Components in Table 4-2 of MRP-227-A and for CE designed Expansion Components in Table 4-5 of MRP-227-A. The St. Lucie RVI Inspection Program is supplemented by the addition of core support barrel expandable plugs and patches to the Primary Program Components for Unit 1 only. These components were used to repair the core barrel damage associated with the loss of the thermal shield early in plant life. The aging effects monitored for included cracking due to IASCC, SCC and fatigue. Enhanced visual examinations (EVT-1) will be performed no later than 2 refueling outages from the beginning of the PEO and every 10 years thereafter. The St. Lucie RVI Inspection Program Primary Component inspections include visual inspection (VT-3) for the presence of distortion of the core shroud due to void swelling, as evidence by separation of the assembly's upper and lower portions. If a gap exists, physical measurements are performed from the core side at the core shroud re-entrant corners. Plant specific acceptance criteria for these measurements have been developed as described below.
5	Monitoring and Trending	The methods for monitoring, recording, evaluating, and trending the data that result from the St. Lucie RVI Inspection Program inspections are given in Section 6 of MRP-227-A. The evaluation methods include recommendations for flaw depth sizing and for crack growth determinations as well 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

	Plan Attribute	Approach and supplemental information
		B-N-3 examinations for core support structures, provides a high degree of confidence in the total program.
6	Acceptance Criteria	Section 5 of MRP-227-A provides the examination acceptance criteria for the Primary and Expansion Components in the St. Lucie RVI Inspection Program. For Existing Program components referenced to ASME Section XI, the IWB-3500 acceptance criteria apply. Plant specific acceptance criteria has been developed for the core shroud gap measurements, should they be required. The allowable gap size to insure continued functionality is based upon design and as-built conditions, fluence, circumferential bounds of the gap (how far around the core shroud can the gap exist),stress, impact on adjacent reactor vessel internals components, impact on core and bypass flow rates, and potential effects on fuel management schemes.
7	Corrective Actions	Components with identified relevant conditions shall be entered into the St. Lucie Corrective action Program (CAP). The disposition may include a supplementary examination to further characterize the relevant condition, an engineering evaluation to show that the component is capable of continued operation with a known relevant condition until the next planned inspection, or repair/replacement to remediate the relevant condition. Additional inspections of expansion category components may also be required. The disposition will insure that the design basis function of the RVI will continue to be fulfilled for all licensing basis loads and events.
8	Confirmation Process and Self Assessment	The PSL quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 50, Appendix B. 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 St. Lucie RVI that are addressed in accordance the 10 CFR Part 50, Appendix B confirmation process and administrative controls.
9	Administrative Controls	The St. Lucie RVI Inspection Program is implemented by 0-ADM-17.29.
10	Operating Experience	FPL actively participates in joint industry programs addressing RVI issues including EPRI and PWROG. In accordance with 0-ADM-17.29, operating experience gained from these groups as well as INPO, WANO and international sites will be incorporated into the St. Lucie RVI Inspection Program in a timely manner.

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Core shroud bolts	Bolted plant designs **NA for PSL**	Cracking (IASCC), Fatigue Aging Management (IE and ISR) (Note 2)	Core support column bolts, Barrel-shroud bolts	Baseline volumetric (UT) examination between 25 and 35 EFPY, with subsequent examination on a ten-year interval.	100% of accessible bolts (see Note 3). Heads are accessible from the core side. UT accessibility may be affected by complexity of head and locking device designs. See Figure 4-24, MRP-227-A.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections **Applicable for PSL**	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial and horizontal weld seams at the core shroud re-entrant corners as visible from the core side of the shroud, within six inches of central flange and horizontal stiffeners. See Figures 4-12 and 4-14, MRP-227-A.
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates **NA for PSL**	Cracking (IASCC) Aging Management (IE) (Note 2)	Remaining axial welds, ribs and rings	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period and subsequent examination on a ten-year interval.	Axial weld seams at the core shroud re-entrant corners, at the core mid-plane (±three feet in height) as visible from the core side of the shroud. See Figure 4-13, MRP-227-A.

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly **(Bolted)** Assembly	Bolted plant designs **NA for PSL**	Distortion (Void Swelling), including: • Abnormal interaction with fuel assemblies • Gaps along high fluence shroud plate joints • Vertical displacement of shroud plates near high fluence joint Aging Management (IE)	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.	Core side surfaces as indicated. See Figures 4-25 and 4-26, MRP-227-A.

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Core Shroud Assembly **(Welded)** Assembly	Plant designs with core shrouds assembled in two vertical sections **Applicable for PSL**	Distortion (Void Swelling), as evidenced by separation between the upper and lower core shroud segments Aging Management (IE)	None	Visual (VT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	If a gap exists, make three to five measurements of gap opening from the core side at the core shroud re-entrant corners. Then, evaluate the swelling on a plant-specific basis to determine frequency and method for additional examinations. See Figures 4-12 and 4-14, MRP-227-A.
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants **Applicable for PSL**	Cracking (SCC)	Lower core support beams. Core support barrel assembly upper cylinder Upper core barrel flange	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the upper flange weld. See Figure 4-15, MRP-227-A.
Core Support Barrel Assembly Lower cylinder girth welds	All plants **Applicable for PSL**	Cracking (SCC, IASCC) Aging Management (IE)	Lower Cylinder Axial Welds	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval.	100% of the accessible surfaces of the lower cylinder welds. (Note 4) See Figure 4-15, MRP-227-A.

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Structure Core support column welds)	All plants **Applicable for PSL**	Cracking (SCC,9 IASCC) Aging Management (IE)	Lower Cylinder Axial Welds	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 the accessible surfaces of the core support column welds. (Note 5) See Figure 4-16 and 4-31, MRP-227-A.
Core Support Barrel Assembly Lower flange weld	All plants **No inspections required for PSL Units 1 and 2 as TLAA exists.**	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by plant-specific fatigue analysis. See Figure 4-15 and 4-16, MRP-227-A.
Core Support Barrel Assembly Expandable plugs and patches	**PSL Unit 1 Only**	Cracking (IASCC, SCC, Fatigue)	None	Enhanced visual (EVT-1) examination no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examinations on a ten-year interval	Repair region of core support barrel

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Lower Support Structure Core support plate	All plants with a core support plate **No inspections required for PSL Units 1 and 2 as TLAA exists.**	Cracking (Fatigue) Aging Management (IE)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by evaluation to determine the potential location and extent of fatigue cracking. See Figure 4-16, MRP-227-A.
Upper Internals Assembly Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates **NA for PSL**	Cracking (Fatigue)	None	If fatigue life cannot be demonstrated by time-limited aging analysis (TLAA), enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval.	Examination coverage to be defined by plant-specific fatigue analysis. See Figure 4-17, MRP-227-A.

Table 1 CE Plants Primary Components

Item	Applicability	Effect (Mechanism)	Expansion Link (Note 1)	Examination Method/Frequency (Note 1)	Examination Coverage
Control Element Assembly Instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly **Applicable for PSL**	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports	Remaining instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval. Plant-specific component integrity assessments may be required if degradation is detected and remedial action is needed.	100% of tubes in peripheral CEA shroud assemblies (i.e., those adjacent to the perimeter of the fuel alignment plate). See Figure 4-18, MRP-227-A.
Lower Support Structure Deep beams	All plants with core shrouds assembled with full-height shroud plates **NA for PSL**	Cracking (Fatigue) that results in a detectable surface-breaking indication in the welds or beams Aging Management (IE)	None	Enhanced visual (EVT-1) examination, no later than 2 refueling outages from the beginning of the license renewal period. Subsequent examination on a ten-year interval, if adequacy of remaining fatigue life cannot be demonstrated.	Examine beam-to-beam welds, in the axial elevation from the beam top surface to four inches below. See Figure 4-19, MRP-227-A.

NOTE:
1) Examination acceptance criteria and expansion criteria are in Table 4.
2) Void swelling effects on this component is managed through management of void swelling on the entire core shroud assembly.
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) A minimum of 75% of the total population of core support column welds

Table 2 CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Core Shroud Assembly (Bolted) Barrel-shroud bolts	Bolted plant designs **NA for PSL**	Cracking (IASCC, Fatigue) Aging Management (IE and ISR)	Core shroud bolts	Volumetric (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific justification; Note 2) of barrel-shroud and guide lug insert bolts with neutron fluence exposures > 3 displacements per atom (dpa). See Figure 4-23, MRP-227-A.
Core Support Barrel Assembly Lower core barrel flange	All plants **Applicable for PSL**	Cracking (SCC, Fatigue)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination Re-inspection every 10 years following the initial inspecttion. .	100% of accessible welds and adjacent base metal (Note 2). See Figure 4-15, MRP-227-A.
Core Support Barrel Assembly Upper cylinder (including welds)	All plants **Applicable for PSL**	Cracking (SCC) Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces of the welds and base metal (Note 2). See Figure 4-15, MRP-227-A.
Core Support Barrel Assembly Upper core barrel flange	All plants **Applicable for PSL**	Cracking (SCC)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination,. Re-inspection every 10 years following initial inspection.	100% of accessible bottom surface of the flange (Note 2). See Figure 4-15, MRP-227-A.
Core Support Barrel Assembly Core barrel assembly axial welds	All plants **Applicable for PSL**	Cracking (SCC)	Core barrel assembly girth welds	Enhanced visual (EVT-1) examination, with initial and subsequent examinations dependent on the results of core barrel assembly girth	100% of one side of the accessible weld and adjacent base metal surfaces for the weld with the highest calculated operating stress.

Table 2 CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
				weld examinations. .	See Figure 4-15, MRP-227-A.
Lower Support Structure Lower support column beams	All plants except those with core shrouds assembled with full-height shroud plates. **Applicable for PSL**	Cracking (SCC, fatigue) including damaged or fractured material. Aging Management (IE)	Upper (core support barrel) flange weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	100% of accessible surfaces (Note 2). See Figure 4-16 and 4-31, MRP-227-A.
Core Shroud Assembly (Bolted) Core support column bolts	Bolted plant designs **NA for PSL**	Cracking (IASCC, Fatigue) Aging Management (IE)	Core shroud bolts	Ultrasonic (UT) examination. Re-inspection every 10 years following initial inspection.	100% (or as supported by plant-specific analysis) of core support column bolts with neutron fluence exposures > 3 dpa. (Note 2) See Figures 4-16 and 4-33, MRP-227-A.
Core Shroud Assembly (Welded) Remaining axial welds	Plant designs with core shrouds assembled in two vertical sections **Applicable for PSL**	Cracking (IASCC)	Core shroud plate-former plate weld	Enhanced visual (EVT-1) examination. Re-inspection every 10 years following initial inspection.	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane. See Figure 4-12, MRP-227-A.
Core Shroud Assembly (Welded) Remaining axial welds,	Plant designs with core shrouds assembled with full-height shroud plates	Cracking (IASCC)	Shroud plates of welded core shroud assemblies	Enhanced visual (EVT-1) examination, with initial and subsequent examination frequencies dependent on the results of the core	Axial weld seams other than the core shroud re-entrant corner welds at the core mid-plane, plus ribs and rings.

Table 2 CE Plants Expansion Components

Item	Applicability	Effect (Mechanism)	Primary Link (Note 1)	Examination Method (Note 1)	Examination Coverage
Ribs and rings	**NA for PSL**			shroud weld examinations.	See Figure 4-13, MRP-227-A.
Control Element Assembly Remaining instrument guide tubes	All plants with instrument guide tubes in the CEA shroud assembly **Applicable for PSL**	Cracking (SCC, Fatigue) that results in missing supports or separation at the welded joint between the tubes and supports.	Peripheral instrument guide tubes within the CEA shroud assemblies	Visual (VT-3) examination, with initial and subsequent examinations dependent on the results of the instrument guide tubes examinations.	100% of tubes in CEA shroud assemblies. See Figure 4-18, MRP-227-A.

NOTE:
1) Examination acceptance criteria and expansion criteria 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 3 CE Plants Existing Program Components

Item	Applicability	Effect (Mechanism)	Reference	Examination Method	Examination Coverage
Core Shroud Assembly Guide lugs Guide lug inserts and bolts	All plants **Applicable for PSL**	Loss of material (Wear) Aging Management (ISR)	ASME Code Section XI	Visual (VT-3) examination, general condition examination for detection of excessive or asymmetrical wear.	First 10-year ISI after 40 years of operation, and at each subsequent inspection interval.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled with full-height shroud plates **NA for PSL**	Cracking (SCC, IASCC, Fatigue) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination to detect severed fuel alignment pins, missing locking tabs, or excessive wear on the fuel alignment pin nose or flange.	Accessible surfaces at specified frequency.
Lower Support Structure Fuel alignment pins	All plants with core shrouds assembled in two vertical sections **Applicable for PSL**	Loss of material (Wear) Aging Management (IE and ISR)	ASME Code Section XI	Visual (VT-3) examination.	Accessible surfaces at specified frequency.
Core Barrel Assembly Upper flange	All plants **Applicable for PSL**	Loss of material (Wear)	ASME Code Section XI	Visual (VT-3) examination.	Area of the upper flange potentially susceptible to wear.

Table 4 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Bolted) *Core shroud bolts*	Bolted plant designs **NA for PSL**	Volumetric (UT) examination. The examination acceptance criteria for the UT of the core shroud bolts shall be established as part of the examination technical justification.	a. Core support column bolts b. Barrel-shroud bolts	a. Confirmation that >5% of the core shroud bolts in the four plates at the largest distance from the core contain unacceptable indications shall require UT examination of the lower support column bolts barrel within the next 3 refueling cycles. b. Confirmation that >5% of the core support column bolts contain unacceptable indications shall require UT examination of the barrel-shroud bolts within the next 3 refueling cycles.	a and b. The examination acceptance criteria for the UT of the core support column bolts and barrel-shroud bolts shall be established as part of the examination technical justification.
Core Shroud Assembly (Welded) Core shroud plate-former plate weld	Plant designs with core shrouds assembled in two vertical sections **Applicable for PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Remaining axial welds	Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the core shroud plate-former plate weld at the core shroud re-entrant corners (as visible from the core side of the shroud), within 6 inches of the central flange and horizontal stiffeners, shall require EVT-1 examination of all remaining axial welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

Table 4 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Shroud plates	Plant designs with core shrouds assembled with full-height shroud plates **NA for PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	a. Remaining axial welds b. Ribs and rings	a. Confirmation that a surface-breaking indication > 2 inches in length has been detected and sized in the axial weld seams at the core shroud re-entrant corners at the core mid-plane shall require EVT-1 or UT examination of all remaining axial welds by the completion of the next refueling outage. b. If extensive cracking is detected in the remaining axial welds, an EVT-1 examination shall be required of all accessible rib and ring welds by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.
Core Shroud Assembly (Bolted) Assembly	Bolted plant designs **NA for PSL**	Visual (VT-3) examination. The specific relevant conditions are evidence of abnormal interaction with fuel assemblies, gaps along high fluence shroud plate joints, and vertical displacement of shroud plates near high fluence joints.	None	N/A	N/A

Table 4 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Shroud Assembly (Welded) Assembly	Plant designs with core shrouds assembled in two vertical sections **Applicable for PSL**	Visual (VT-1) examination. The specific relevant condition is evidence of physical separation between the upper and lower core shroud sections.	None	N/A	N/A
Core Support Barrel Assembly Upper (core support barrel) flange weld	All plants **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower core support beams Upper core barrel cylinder (including welds) Upper core barrel flange (cast)	Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the upper flange weld shall require that an EVT-1 examination of the lower core support beams, upper core barrel cylinder and upper core barrel flange be performed by the completion of the next refueling outage.	The specific relevant condition is a detectable crack-like surface indication.

Table 4 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Core Support Barrel Assembly Lower cylinder girth welds	All plants **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	Lower cylinder axial welds	a. Confirmation that a surface-breaking indication >2 inches in length has been detected and sized in the lower cylinder girth weld shall require an EVT-1 examination of all accessible lower cylinder axial welds by completion of the next refueling outage.	The specific relevant condition for the expansion lower cylinder axial welds is a detectable crack-like surface indication.
Lower Support Structure Core support column welds	All plants **Applicable to PSL**	Visual (VT-3) examination. The specific relevant condition is missing or separated welds.	None	None	
Core Support Barrel Assembly Lower flange weld	All plants **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A
Core Support Barrel Assembly Expandable plugs and patches	**PSL Unit 1 Only**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	
Lower Support Structure Core support plate	All plants with a core support plate **Applicable to PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A

Table 4 CE Plants Examination Acceptance and Expansion Criteria

Item	Applicability	Examination Acceptance Criteria (Note 1)	Expansion Link(s)	Expansion Criteria	Additional Examination Acceptance Criteria
Upper Internals Assembly Fuel alignment plate	All plants with core shrouds assembled with full-height shroud plates **NA for PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like surface indication.	None	N/A	N/A
Control Element Assembly Instrument Guide Tubes	All plants with instruments tubes in the CEA shroud assembly **Applicable to PSL**	Visual (VT-3) examination. The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.	Remaining instrument tubes within the CEA shroud assemblies	Confirmed evidence of missing supports or separation at the welded joint between the tubes and supports shall require the visual (VT-3) examination to be expanded to the remaining instrument tubes within the CEA shroud assemblies by completion of the next refueling outage.	The specific relevant conditions are missing supports and separation at the welded joint between the tubes and the supports.
Lower Support Structure Deep beams	All plants with core shrouds assembled with full-height shroud plates **NA for PSL**	Visual (EVT-1) examination. The specific relevant condition is a detectable crack-like indication.	None	N/A	N/A

NOTE
1. The examination acceptance criterion for visual examination is the absence of the specified relevant condition(s).

3 LICENSEE ACTION ITEMS

This section provides the FPL response to the eight Licensee Action Items (LAI) noted in the NRC Safety Evaluation Report (SER) issued by the NRC for the report EPRI-MRP-227-A, "Materials Reliability Program: Pressurized Water Reactor Internals Inspection and Evaluation Guidelines" dated December 2011. Additionally, the three bounding assumptions included in Section 2.4 of MRP-227-A are also addressed in the response to LAI #1.

LAI #1 Applicability of FMECA and Functionality Analysis Assumptions
As addressed in Section 3.2.5.1 of this SE, each applicant/licensee is responsible for assessing its plant's design and operating history and demonstrating that the approved version of MRP-227 is applicable to the facility. Each applicant/licensee shall refer, in particular, to the assumptions regarding plant design and operating history made in the FMECA and functionality analyses for reactors of their design (i.e., Westinghouse, CE, or B&W) which support MRP-227 and describe the process used for determining plant-specific differences in the design of their RVI components or plant operating conditions, which result in different component inspection categories. The applicant/licensee shall submit this evaluation for NRC review and approval as part of its application to implement the approved version of MRP-227. ***This is Applicant/Licensee Action Item 1.***

The response to LAI #1 is based directly upon Westinghouse Letter LTR-RIAM-13-75, Rev. 0, Final Summary Report for St. Lucie Units 1 and 2 for PWROG PA-MSC-0983 Cafeteria Task Deliverables.

FPL Response to LAI #1 and Bounding Assumption of MRP-227-A:
The process used to provide reasonable assurance that St. Lucie Units 1 and 2 are reasonably represented by the generic industry program assumptions (with regard to neutron fluence, temperature, stress values, and materials used in the development of MRP-227-A) is:

1. Identification of typical Combustion Engineering (CE)-designed pressurized water reactor (PWR) reactor vessel internals (RVI) components (Table 4-5 of MRP-191).

2. Identification of St. Lucie Units 1 and 2 PWR components.

3. Comparison of the typical CE-designed PWR RVI components to the St. Lucie Units 1 and 2 RVI components:

    a. Confirmation that no additional items were identified by this comparison (primarily supports Applicant/Licensee Action Item 2).

    b. Confirmation that the materials from Table 4-5 of MRP-191 are consistent with St. Lucie Units 1 and 2 RVI component materials.

c. Confirmation that the design and fabrication of St. Lucie Units 1 and 2 RVI components are the same as, or equivalent to, the typical CE-designed PWR RVI components.

4. Confirmation that the St. Lucie Units 1 and 2 operating history is consistent with the assumptions in MRP-227-A regarding core loading patterns and base load operation.

5. Confirmation that the St. Lucie Units 1 and 2 RVI materials operated at temperatures within the original design basis parameters.

6. Determination of stress values based on design basis documents.

7. Confirmation that any changes to the St. Lucie Units 1 and 2 RVI components do not impact the application of the MRP-227-A generic aging management strategy.

The St. Lucie Units 1 and 2 RVI components are reasonably represented by the design and operating history assumptions regarding neutron fluence, temperature, materials, and stress values in the MRP-191 generic FMECA and in the MRP-232 functionality analysis based on the following:

1. St. Lucie Units 1 and 2 operating history is consistent with the assumptions in MRP-227-A with regard to neutron fluence and fuel management.

a. FMECA and functionality analysis for MRP-227-A made the following assumption of 30 years of operation with high-leakage core loading patterns followed by 30 years of low-leakage core fuel management strategy. The St. Lucie Units 1 and 2 fuel management program changed from a high to a low leakage core loading pattern prior to 30 years of operation. Therefore, St. Lucie Units 1 and 2 meet the fluence and fuel management assumptions in MRP-191 and requirements for MRP-227-A application.

b. St. Lucie Units 1 and 2 have operated under base load conditions over the life of the plant Therefore, St. Lucie Units 1 and 2 satisfy the assumptions in Materials Reliability Program (MRP) documents regarding operational parameters affecting fluence.

2. The St. Lucie Units 1 and 2 reactor coolant system operates between T_{cold} and T_{hot}. T_{cold} is not less than 532°F and there were no changes to T_{cold} due to extended power uprate (EPU). T_{hot} was no higher than 594°F prior to EPU and no higher than 608.2°F after EPU for Unit 1. T_{hot} was no higher than 598°F prior to EPU and no higher than 607.9°F after EPU for Unit 2. The design temperature for the vessel is 650°F. Therefore, St. Lucie Units 1 and 2 operating history is within original design basis parameters and is

consistent with the assumptions used to develop the MRP- 227-A aging management strategy with regard to temperature operational parameters.

3. With the exceptions discussed below, the St. Lucie Units 1 and 2 RVI components and materials are comparable to the typical CE-designed PWR RVI components (MRP-191, Table 4-5).

a. There are two additional components for St. Lucie Unit 1 and one component for Unit 2 that are not included in MRP-191. In Unit 1, core support barrel patches and core support barrel expandable plugs were installed following the discovery of damage to the core barrel caused by fatigue of the thermal shield attachment points. CE developed and analyzed the repair method. For Unit 2, there are four specialized control element assembly (CEA) shroud assemblies that are fitted with flow bypass inserts. Other than the core support barrel patches, core support barrel expandable plugs, and flow bypass inserts, the components required for inclusion in the St. Lucie Units 1 and 2 program are consistent with those contained in MRP-191.

b. St. Lucie Units 1 and 2 RVI component materials are consistent with, or nearly equivalent to, those materials identified in Table 4-5 of MRP-191 for CE-designed plants. Where differences exist, there is no impact on the St. Lucie Units 1 and 2 RVI program or the component is already credited as being managed under an alternate St. Lucie Units 1 and 2 aging management program.

c. Design and fabrication of St. Lucie Units 1 and 2 RVI components are the same as, or equivalent to, the typical CE-designed PWR RVI components.

4. An 11.85% EPU was performed on St. Lucie Units 1 and 2. Evaluations performed by Westinghouse determined that the associated changes in temperature, fluence and loading on the RVI components did not affect the bounding assumptions or applicability of MRP-227-A. With the exception of the thermal shield removal for Unit 1, the modifications to the St. Lucie Units 1 and 2 RVI made over the lifetime of the plants are those identified in general industry practice or specifically directed by the original equipment manufacturer (OEM). The Unit 1 thermal shield removal was analyzed to be acceptable. Repairs to the core barrel, as a result of the thermal shield removal, were in accordance with recommendations and guidance of the OEM. Therefore, the design has been maintained over the lifetime of the plant as specified by the OEM and operational parameters with regard to fluence and temperature are compliant with MRP-227-A requirements. With the exception of two components for Unit 1 and one for Unit 2, the components are consistent with those considered in MRP-191. The materials for those components are also consistent with MRP-191, or where differences exist, there is no impact. The additional three components have no impact on the assumptions summarized above; therefore, the St. Lucie Units 1 and 2 RVI are represented by the

assumptions in MRP-191, MRP-227-A, and MRP-232, confirming the applicability of the generic FMECA.

Conclusion

St. Lucie Units 1 and 2 comply with LAI #1 of the Nuclear Regulatory Commission Safety Evaluation on MRP-227, Revision 0. Therefore, the requirement is met for application of MRP-227-A as a strategy for managing age-related material degradation in the RVI components.

LAI #2 PWR Vessel Internal Components Within the Scope of License Renewal

As discussed in Section 3.2.5.2 of this SE, consistent with the requirements addressed in 10 CFR 54.4, each applicant/licensee is responsible for identifying which RVI components are within the scope of LR for its facility. Applicants/licensees shall review the information in Tables 4-1 and 4-2 in MRP-189, Revision 1, and Tables 4-4 and 4-5 in MRP-191 and identify whether these tables contain all of the RVI components that are within the scope of LR for their facilities in accordance with 10 CFR 54.4. If the tables do not identify all the RVI components that are within the scope of LR for its facility, the applicant or licensee shall identify the missing component(s) and propose any necessary modifications to the program defined in MRP-227, as modified by this SE, when submitting its plant-specific AMP. The AMP shall provide assurance that the effects of aging on the missing component(s) will be managed for the period of extended operation. ***This issue is Applicant/Licensee Action Item 2.***

The response to LAI #2 is based directly upon Westinghouse Letter LTR-RIAM-13-75, Rev. 0, Final Summary Report for St. Lucie Units 1 and 2 for PWROG PA-MSC-0983 Cafeteria Task Deliverables.

FPL Response:

This Applicant/Licensee Action Item requires comparison of the St. Lucie Units 1 and 2 RVI components that are within the scope of license renewal for St. Lucie Units 1 and 2 to those components contained in Table 4-5 of MRP-191. MRP-189, Tables 4-1 and 4-2 are not applicable to St. Lucie Units 1 and 2 since those tables are applicable to a B&W-plant design, while St. Lucie Units 1 and 2 are CE-plant design. There are two additional components for St. Lucie Unit 1 and one component for Unit 2 identified in the plant-specific aging management review (AMR) that are not included in MRP-191. In Unit 1, core support barrel patches and core support barrel expandable plugs were installed following the discovery of damage to the core barrel caused by fatigue at the thermal shield attachment points, and for Unit 2, there are four specialized CEA shroud assemblies that are fitted with flow bypass inserts. Other than the core support barrel patches, core support barrel expandable plugs, and flow bypass inserts, all components in the St. Lucie Units 1 and 2 license renewal program are consistent with those contained in MRP-191.

The in-core instrumentation (ICI) guide tubes for both units have a different material than that specified in MRP-191, but the difference has no effect on the recommended MRP aging strategy or is already managed by an alternate St. Lucie Units 1 and 2 program; therefore, no modifications to the program details in MRP-227-A need to be proposed. This supports the requirement that the NRC-AMP shall provide assurance that the effects of aging on the St. Lucie Units 1 and 2 RVI components within the scope of license renewal, but not included in the generic CE-designed PWR RVI components from Table 4-5 of MRP-191, will be managed for the period of extended operation.

The generic scoping and screening of the RVI, as summarized in MRP-191 and MRP-232, to support the inspection sampling approach for aging management of the RVI specified in MRP-227- A are applicable to St. Lucie Units 1 and 2 with no modifications for the St. Lucie components that are consistent with those contained in MRP-191. For the three components that are not included in MRP-191, the aging management strategy has been determined on a plant-specific basis. FPL has conservatively categorized the Unit 1 core support barrel patches and core support barrel expandable plugs as Primary components for aging management during the period of extended operation. Plant-specific augmented inspections are required on a periodic basis to manage the associated aging effects on Primary components. St. Lucie Unit 2 has four specialized CEA shroud assemblies that are fitted with flow bypass inserts. MRP-191 categorized all components of the CEA shroud assemblies as Category A. Therefore, FPL categorized the Unit 2 flow bypass inserts consistently, making them No Additional Measures components. No Additional Measures components are either not susceptible to any degradation mechanism, or if susceptible the impact of failure on the functionality of the RVI components is insignificant. No further action is required for managing aging of these RVI components.

Conclusion

St. Lucie Units 1 and 2 comply with LAI #2 of the Nuclear Regulatory Commission Safety Evaluation on MRP-227, Revision 0. The assessment performed identified three additional components that are not identified in MRP-191. The aging management strategy for these additional components has been included in the plant-specific program to ensure aging is managed for components that are not included within the scope of MRP-227-A. Therefore, St. Lucie Units 1 and 2 meet the requirement for application of MRP-227-A as a strategy for managing age-related material degradation in reactor internals components.

<u>LAI #3 Evaluation of the Adequacy of Plant-Specific Existing Programs</u>

As addressed in Section 3.2.5.3 in this SE, applicants/licensees of CE and Westinghouse are required to perform plant-specific analysis either to justify the acceptability of an applicant's/licensee's existing programs, or to identify changes to the programs that should be implemented to manage the aging of these components for the period of extended operation. The results of this plant-specific analyses and a description of the plant-specific programs being relied on to manage aging of these components shall be submitted as part of the applicant's/licensee's AMP application. The CE and Westinghouse components identified for

this type of plant-specific evaluation include: CE thermal shield positioning pins and CE in-core instrumentation thimble tubes (Section 4.3.2 in MRP-227), and Westinghouse guide tube support pins (split pins) (Section 4.3.3 in MRP-227). ***This is Applicant/Licensee Action Item 3.***

FPL Response:

There are no thermal shields or thermal shield positioning pins installed on the core barrels of St. Lucie Units 1 and 2.

The St. Lucie Units 1 and 2 in-core instrumentation flux thimble tubes are considered out-of-scope for license renewal based upon the component screening performed in accordance with the Nuclear License Renewal Rule (10 CFR 54). All in-core instrumentation flux thimble tubes for St. Lucie Unit 1 were replaced during the Cycle 21 outage (Spring 2007) (WO 35010464), and those for St. Lucie Unit 2 were replaced during the Cycle 19 outage (Spring 2011) (WO 35010467). The replacement thimbles have been designed with sufficient margin to accommodate growth of thimbles' zircalloy sections during the PEO.

Conclusion
LAI #3 is not applicable to St. Lucie Units 1 and 2.

LAI #4 B&W Core Support Structure Upper Flange Stress Relief
As discussed in Section 3.2.5.4 of this SE, the B&W applicants/licensees shall confirm that the core support structure upper flange weld was stress relieved during the original fabrication of the Reactor Pressure Vessel in order to confirm the applicability of MRP-227, as approved by the NRC, to their facility. If the upper flange weld has not been stress relieved, then this component shall be inspected as a "Primary" inspection category component. If necessary, the examination methods and frequency for non-stress relieved B&W core support structure upper flange welds shall be consistent with the recommendations in MRP-227, as approved by the NRC, for the Westinghouse and CE upper core support barrel welds. The examination coverage for this B&W flange weld shall conform to the staff's imposed criteria as described in Sections 3.3.1 and 4.3.1 of this SE. The applicant's/licensee's resolution of this plant-specific action item shall be submitted to the NRC for review and approval. ***This is Applicant/Licensee Action Item 4.***

FPL Response: LAI #4 pertains to B&W Core Support Structure Upper Flange Stress Relief issue and is not applicable to St. Lucie Units 1 and 2 which are CE NSSS designs.

LAI #5 Application of Physical Measurements as part of I&E Guidelines for B&W, CE, and Westinghouse RVI Components
As addressed in Section 3.3.5 in this SE, applicants/licensees shall identify plant-specific acceptance criteria to be applied when performing the physical measurements required by the

NRC-approved version of MRP-227 for loss of compressibility for Westinghouse hold down springs, and for distortion in the gap between the top and bottom core shroud segments in CE units with core barrel shrouds assembled in two vertical sections. The applicant/licensee shall include its proposed acceptance criteria and an explanation of how the proposed acceptance criteria are consistent with the plants' licensing basis and the need to maintain the functionality of the component being inspected under all licensing basis conditions of operation during the period of extended operation as part of their submittal to apply the approved version of MRP-227. ***This is Applicant/Licensee Action Item 5.***

FPL Response:

The response to LAI #5 is based directly upon Westinghouse Letter LTR-RIAM-13-147, Rev. 0, Transmittal of Final Summary Letter for Acceptance Criteria for Visual Examination of Gaps between Upper and Lower Core Shroud Subassemblies at Calvert Cliffs Units 1 and 2 and St. Lucie Units 1 and 2.

FPL participated in a PWROG Project Authorization (PA) to justify a gap size for the St. Lucie Units 1 and 2 core shrouds. Basic assumptions of the PA were that the gap be measurable using the specified VT-1 inspection resolution and that it satisfy functionality requirements. The Units 1 and 2 core shrouds differ slightly in design - Unit 1 uses a mechanical attachment (via tie rods) between the upper and lower core shroud sections, whereas Unit 2 uses a welded attachment. The postulated gap would include both thermal and void swelling contributions. The thermal contribution would be present only during power operation. The void swelling contribution would be present under all conditions including plant shutdown, during which the physical examination of the core shroud will be performed.

Core shroud gap acceptance criteria have been developed for St. Lucie Units 1 and 2 that are resolvable using the specified VT-1 inspection method of MRP-227-A. Plant-specific details are proprietary and not typically released publicly. If the NRC requests additional details, the calculation can be made available for review. This satisfies the requirements of LAI #5.[F1]

<u>LAI #6 Evaluation of Inaccessible B&W Components</u>

As addressed in Section 3.3.6 in this SE, MRP-227 does not propose to inspect the following inaccessible components: the B&W core barrel cylinders (including vertical and circumferential seam welds), B&W former plates, B&W external baffle-to-baffle bolts and their locking devices, B&W core barrel-to-former bolts and their locking devices, and B&W core barrel assembly internal baffle-to-baffle bolts. The MRP also identified that although the B&W core barrel assembly internal baffle-to-baffle bolts are accessible, the bolts are non-inspectable using currently available examination techniques. Applicants/licensees shall justify the acceptability of these components for continued operation through the period of extended operation by performing an evaluation, or by proposing a scheduled replacement of the components. As part of their application to implement the approved version of MRP-227, applicants/licensees shall provide their justification for the continued operability of each of the inaccessible components

and, if necessary, provide their plan for the replacement of the components for NRC review and approval. ***This is Applicant/Licensee Action Item 6.***

FPL Response: LAI #6 pertains to B&W Inaccessible Components and is not applicable to St. Lucie Units 1 and 2 which are CE NSSS designs.

LAI #7 Plant-Specific Evaluation of CASS Materials

As discussed in Section 3.3.7 of this SE, the applicants/licensees of B&W, CE, and Westinghouse reactors are required to develop plant-specific analyses to be applied for their facilities to demonstrate that B&W IMI guide tube assembly spiders and CRGT spacer castings, CE lower support columns, and Westinghouse lower support column bodies will maintain their functionality during the period of extended operation or for additional RVI components that may be fabricated from CASS, martensitic stainless steel or precipitation hardened stainless steel materials. These analyses shall also consider the possible loss of fracture toughness in these components due to thermal and irradiation embrittlement, and may also need to consider limitations on accessibility for inspection and the resolution/sensitivity of the inspection techniques. The requirement may not apply to components that were previously evaluated as not requiring aging management during development of MRP-227. That is, the requirement would apply to components fabricated from susceptible materials for which an individual licensee has determined aging management is required, for example during their review performed in accordance with Applicant/Licensee Action Item 2. The plant-specific analysis shall be consistent with the plant's licensing basis and the need to maintain the functionality of the components being evaluated under all licensing basis conditions of operation. The applicant/licensee shall include the plant-specific analysis as part of their submittal to apply the approved version of MRP-227. ***This is Applicant/Licensee Action Item 7.***

The response to LAI #7 is based directly upon Westinghouse Letter LTR-RIAM-13-75, Rev. 0, Final Summary Report for St. Lucie Units 1 and 2 for PWROG PA-MSC-0983 Cafeteria Task Deliverables.

FPL Response:

Applicant/Licensee Action Item 7 from the NRC's final Safety Evaluation on MRP-227, Revision 0 states that, for assessment of cast austenitic stainless steel (CASS) materials, the licensees or applicant for license renewal may apply the criteria in the NRC letter of May 19, 2000, "License Renewal Issue No. 98-0030, *Thermal Aging Embrittlement of Cast Stainless Steel Components*" (NRC ADAMS Accession No.ML003717179) as the basis for determining whether the CASS materials are susceptible to the thermal aging mechanism. If the application of the screening criteria for the component material demonstrates that the components are not susceptible to either thermal embrittlement (TE) or irradiation embrittlement (IE), or the synergistic effects of TE and IE combined, then no other evaluation would be necessary.

The St. Lucie Units 1 and 2 RVI CASS components and the assessment of their susceptibility to TE are summarized in Table 1 and as follows:

- The St. Lucie Unit 1 core support columns are low molybdenum and static cast. A certified material test report (CMTR) was located for one two-legged column. Its calculated ferrite content is less than 20%; thus, it is not susceptible to TE. The remaining St. Lucie Unit 1 core support columns are potentially susceptible to TE. The support columns were previously screened in for the age-related degradation mechanism of TE, along with stress corrosion cracking (SCC) of the weld, irradiation-assisted stress corrosion cracking (IASCC), fatigue, and IE in MRP-191, Table 4-7 and the inspection and evaluation guidelines for this Primary component are in MRP-227-A . The St. Lucie Unit 2 core support columns are 304 SS; thus, A/LAI 7 is not applicable to the St. Lucie Unit 2 core support columns

- The St. Lucie Units 1 and 2 control element assembly (CEA) shroud tubes are low molybdenum and centrifugal cast; thus they are not susceptible to TE. The CEA shroud tubes were also previously screened in for the age-related degradation mechanism of SCC of the weld in MRP-191, Table 4-7.

- The St. Lucie Unit 2 flow bypass inserts are low molybdenum, static cast, and have ferrite content ≤ 20%; thus they are not susceptible to TE. The flow bypass inserts were not identified in MRP-191. FPL has categorized the St. Lucie Unit 2 flow bypass inserts as No Additional Measures Components.

CASS Component	Molybdenum Content (Wt.%)	Casting Method	Calculated Ferrite Content	Susceptibility to TE
St. Lucie Unit 1				
Core Support Columns	Low, 0.5 max	Static	≤ 20%(1)	One column not susceptible to TE
			Potentially >20%(2)	Remaining columns potentially susceptible to TE(2)
CEA Shroud Tubes	Low, 0.5 max	Centrifugal	All	Not susceptible to TE
St. Lucie Unit 2				
CEA Shroud Tubes	Low, 0.5 max	Centrifugal	All	Not susceptible to TE
Flow Bypass Inserts	Low, 0.5 max	Static	≤ 20%(1)	Not susceptible to TE

Notes:

1. Calculated ferrite content is based on CMTR data, input into Hull's formula per the guidance of NUREG/CR-4513, Rev. 1. Where molybdenum is not listed on the CMTR, a value of 0.5 percent is used. Where nitrogen is not listed on the CMTR a value of 0.04 percent is used.

2. Where component-specific CMTR data are not available, the ferrite content is calculated based on permitted variations in ASTM A351, Grade CF8 chemical requirement. Allowable variants of Grade CF8 chemical requirements may result in ferrite content greater than 20%; thus, the ferrite content is identified as potentially exceeding 20%.

The St. Lucie Units 1 and 2 martensitic stainless steel RVI components include only a 403 SS Hold-down Ring in each unit. There are no martensitic PH-SS RVI components in St. Lucie Units 1 and 2.

Conclusion

The results of this evaluation do not conflict with strategy for aging management of RVI provided in MRP-227-A. It is concluded that continued application of the strategies in MRP-227-A and the St. Lucie Units 1 and 2 RVI Inspection Program will meet the requirements for managing age-related degradation of the St. Lucie Units 1 and 2 CASS and martensitic SS RVI components.

LAI #8 Submittal of Information for Staff Review and Approval

As addressed in Section 3.5.1 in this SE, applicants/licensees shall make a submittal for NRC review and approval to credit their implementation of MRP-227, as amended by this SE, as an AMP for the RVI components at their facility. This submittal shall include the information identified in Section 3.5.1 of this SE. ***This is Applicant/Licensee Action Item 8.***

FPL Response:

During the license renewal process, St. Lucie Units 1 and 2 prepared and gained approval for RVI Inspection Program from the NRC, as documented in NUREG 1759. Subsequently, during the EPU LAR review, St. Lucie Units 1 and 2 committed to revise the RVI Inspection Program to align with MRP-227-A.

The St. Lucie RVI Inspection Program is summarized in Sections 1 and 2. It provides the following items: 1) components to be inspected; 2) the degradation mechanisms of concern; 3) the inspection methods; 4) the examination coverage; and 5) the examination acceptance criteria. And the responses to the eight Licensee Action Items of MRP-227-A are provided in Section 3. These sections satisfy the requirements of LAI #8.

Attachment 2 to Letter L-2015-229

St. Lucie Units 1 and 2 Proposed UFSAR Revisions

Unit 1 UFSAR Proposed Revision

18.1.4 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel Internals (RVI) Inspection Program manages the aging effects on the RVI during the period of extended operation. The RVI consists of three major structural assemblies, plus three other sets of major components. The three major assemblies include: 1) upper internals assembly; 2) core support barrel assembly, and 3) lower internals assembly. In addition, the three other sets of major components are the control element assembly (CEA) shroud assemblies, core shroud assembly, and in-core instrumentation support system. The RVI Inspection Program is applicable to passive RVI structural components and specifically excludes welded attachments to the reactor vessel and consumable items such as fuel assemblies, control element assemblies (CEAs) and in core instrumentation (ICI).

Aging effects and the causative degradation mechanisms addressed by the RVI Inspection Program include: 1) cracking due to stress corrosion cracking (SCC), irradiation assisted stress corrosion cracking (IASCC) or fatigue; 2) reduction in fracture toughness due to irradiation or thermal embrittlement; 3) loss of material due to wear; 4) dimensional change due to void swelling; 5) loss of mechanical closure integrity (or preload) due to irradiation and thermal enhanced stress relaxation or creep.

The RVI Inspection Program is based upon the guidance provided in EPRI MRP-227-A, "EPRI Materials Reliability Program, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The RVI Inspection Program is a living program that will be revised as necessary in response to ongoing joint industry efforts aimed at further understanding the aging effects of the RV Internals.

FPL has satisfied the following commitments concerning the RVI Inspection Program: 1) Submit an integrated report for St. Lucie Units 1 and 2 to the NRC prior to the end of the initial operating license term for St. Lucie Unit 1 that summarizes its understanding of the aging effects applicable to the reactor vessel internals and contains a description of the St. Lucie inspection plan, including methods for detection and sizing of cracks and acceptance criteria; and 2) Adopt MPR-227-A in place of the previously approved RVI Inspection Program that was included in the St. Lucie Units 1 and 2 License Renewal Applications.

Unit 2 UFSAR Proposed Revision

18.1.3 REACTOR VESSEL INTERNALS INSPECTION PROGRAM

The Reactor Vessel Internals (RVI) Inspection Program manages the aging effects on the RVI during the period of extended operation. The RVI consists of three major structural assemblies, plus three other sets of major components. The three major assemblies include: 1) upper internals assembly; 2) core support barrel assembly, and 3) lower internals assembly. In addition, the three other sets of major components are the control element assembly (CEA) shroud assemblies, core shroud assembly, and in-core instrumentation support system. The RVI Inspection Program is applicable to passive RVI structural components and specifically excludes welded attachments to the reactor vessel and consumable items such as fuel assemblies, control element assemblies (CEAs) and in core instrumentation (ICI).

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The RVI Inspection Program is based upon the guidance provided in EPRI MRP-227-A, "EPRI Materials Reliability Program, Pressurized Water Reactor Internals Inspection and Evaluation Guidelines." The RVI Inspection Program is a living program that will be revised as necessary in response to ongoing joint industry efforts aimed at further understanding the aging effects of the RV Internals.

FPL has satisfied the following commitments concerning the RVI Inspection Program: 1) Submit an integrated report for St. Lucie Units 1 and 2 to the NRC prior to the end of the initial operating license term for St. Lucie Unit 1 that summarizes its understanding of the aging effects applicable to the reactor vessel internals and contains a description of the St. Lucie inspection plan, including methods for detection and sizing of cracks and acceptance criteria; and 2) Adopt MPR-227-A in place of the previously approved RVI Inspection Program that was included in the St. Lucie Units 1 and 2 License Renewal Applications.