FPL

February 26, 2016

L-2016-040
10 CFR 54

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

St. Lucie Units 1 and 2
Docket Nos. 50-335 and 50-389
License Renewal Commitments
Reactor Vessel Internals Aging Management Plan
Response to Request for Additional Information

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. Electric Power Research Institute (EPRI) Materials Reliability Program Report 1022863 (MRP-227-A), "Pressurized Water Reactor Internals Inspection and Evaluation Guidelines," ADAMS Accession Nos. ML12017A194, ML12017A196, ML12017A197, ML12017A191, ML12017A192, ML12017A195, and ML12017A199.

5. 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.

6. FPL Letter from FPL Letter from Christopher Costanzo to U.S. Nuclear Regulatory Commission (L-2015-229) "St. Lucie Units 1 and 2 Docket Nos. 50-335 and 50-389, License Renewal Commitments - Reactor Vessel Internals Aging Management Plan," Dated September 28, 2015.

7. NRC e-Mail from Perry Buckberg to Ken Frehafer, Request for Additional Information, St. Lucie Plant Units 1 and 2, Reactor Vessel Internals Aging Management Plan, Docket Nos. 50-335 and 50-389, TAC Nos. MF6777 and MF6778.

By letter to the U.S. Nuclear Regulatory Commission (NRC) dated September 28, 2015 (Reference 5), Florida Power & Light Company (FPL) submitted its License Renewal Reactor

A089
NRR

Implementation of Materials Reliability Program MRP-227-A (Reference 4) at St. Lucie Nuclear Plant Units 1 and 2 for NRC staff review.

The NRC staff reviewed the information provided by FPL in its submittal and requested additional information to complete their review (Reference 7). FPL responses to the RAIs are provided in Attachment No. 1.

For the response to RAI-3 for addressing the 20% cold work issue for non-weld or bolting austenitic stainless steel, FPL will be crediting the efforts of the PWROG, which is addressing this issue on a generic basis. FPL is a participant in this PWROG Program. A final report is expected by April 2017. FPL will forward the final report to the NRC.

The responses to RAI-5 and RAI-6 are deferred for a later time. Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in RAI-5 and RAI-6 by March 31, 2016. FPL will advise the NRC of the schedule once it has been received.

In response to RAI-8, a non-proprietary version of the Core Shroud Gap calculation is enclosed as Attachment No. 2.

In response to RAI-9, St. Lucie plant is actively participating in two joint industry programs for addressing the functionality of CASS RVI components during period of extended operation for loss of fracture toughness due to thermal embrittlement and irradiation embrittlement. The estimated completion date for these two projects is June 2017.

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 FPL Responses to RAIs
2) PWROG Report No. PWROG-16012-NP, Rev. 0

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-2016-040

St. Lucie Units 1 and 2

RVI Aging Management Plan

FPL Responses to the NRC RAIs

FPL Responses to the NRC Request for Additional Information:

RAI-MF6777/MF6778-EVIB-01

Table 1 of the RVI AMP notes in the "Applicability" column for the Core Support Barrel Assembly - Lower flange weld, and the Lower Support Structure - Core support plate, that no inspections are required for these components in St. Lucie Plant, Units 1 and 2 as a time-limited aging analysis (TLAA) exists. The staff notes that for these components, MRP-227-A, Table 4-2 states under "Examination Method/Frequency" that if fatigue life cannot be demonstrated by TLAA, enhanced visual examination is required no later than 2 refueling outages from the beginning of the license renewal period, with subsequent examinations on a 10-year interval. TLAA's are analyses that must meet six criteria as defined in Title 10 of the Code of Federal Regulation (10 CFR) 54.3, one of which is that the analyses are "contained or incorporated by reference in the current licensing basis." However, the license renewal application and NUREG-1779 do not identify TLAA's related to fatigue of the lower support structure – lower flange weld and lower support structure - core support plate. Therefore, these analyses are apparently new analyses or did not meet the criteria for a TLAA at the time FPL applied to renew the St. Lucie Plant licenses. The staff therefore needs more information in order to review the licensee's determination that the fatigue analyses adequately manage the aging effect of cracking due to fatigue. The staff therefore requests the licensee:

1. Clarify whether these analyses were previously part of the current licensing basis for St. Lucie Plant, or whether they are new analyses,

Response

Fatigue of the Reactor Vessel Internals (RVI) was identified as a time-limited aging analysis (TLAA) in the St. Lucie License Renewal Application (LRA), Section 4.3. It was also identified in NUREG-1779.

2. Describe the methodology and results of the fatigue analyses, including the cumulative usage factor (CUF) obtained from these calculations.

Response

The RVI fatigue evaluation was conducted in accordance with Paragraph NG-3228.3 of ASME Section III which requires a simplified elastic-plastic analysis for any component in which the primary-plus-secondary stress intensity exceeds 3Sm. This condition occurs in the core shroud and the instrument tube supports. Cumulative usage factors (CUF) of <1 were calculated for both of these components. For the remaining RVI components a general scoping fatigue evaluation was performed using the same simplified elastic-plastic analysis with the maximum primary-plus-secondary stress intensity range set at 3Sm. The calculated CUF for the remaining RVI components was also <1.

3. Did the fatigue analyses consider the effects of the reactor water environment of the CUF? If so, describe how the effects of the environment were considered.

Response

Section 4.3.3 of the St. Lucie LRA addresses Environmental Fatigue for Class 1 components. Environmental fatigue calculations were performed for the fatigue sensitive components identified in NUREG/CR-6260. Based upon these results, all component locations were determined to be acceptable for the period of extended operation, with the exception of the pressurizer surge lines. A proposed aging management program (AMP) for the pressurizer surge line has been submitted to the NRC for review/approval (Ref. L-2015-272). FPL's commitment to inspect the surge line provides reasonable assurance that potential environmental effects of fatigue will be managed such that other Class 1 components within the scope of license renewal will continue to perform their intended functions consistent with the St. Lucie Units 1 and 2 CLBs for the PEO. This approach meets the requirements specified in the NRC closure of GSI-190.

4. Describe how these analyses are documented at St. Lucie Plant (for example, in the UFSAR, design calculation, engineering report, appendix to the RVI AMP, etc.).

Response
Metal fatigue is identified as a TLAA in the LR Application, NUREG-1779 and Chapter 18 of the UFSAR.

RAI-MF6777/MF6778-EVIB-02

The staff evaluated the licensee's description of the ten elements of its RVI AMP using the criteria of License Renewal Interim Staff Guidance LR-ISG-2011-04: "Updated Aging Management Criteria for Reactor Vessel Internal Components for Pressurized Water Reactors (Ref. 6)," which represents the most current U.S. Nuclear Regulatory Commission (NRC) guidance on aging management of RVI. The staff found a few instances in which the licensee's descriptions of the AMP elements did not address certain items from the guidance of LR-ISG-2011-04. Therefore, the staff requests the licensee:

1. Confirm that the Administrative Controls element of the RVI AMP is governed by the site's 10 CFR 50, Appendix B quality assurance program.

Response
The St. Lucie RVI Inspection Program Administrative Control element has been revised to better align with LR-ISG-2011-04. See revisions below.

2. With respect to the Confirmation Process, Administrative Controls, and Operating Experience elements of the AMP, discuss how the RVI AMP meets the NEI 03-08 implementation requirements for MRP-227-A.

Response
The St. Lucie RVI Inspection Program Confirmation Process, Administrative Controls, and Operating Experience elements have been revised to better align with LR-ISG-2011-04. See revisions below.

8	Confirmation Process and Self Assessment	The PSL quality assurance procedures, review and approval processes, and administrative controls are implemented in accordance with the recommendations of NEI 03-08 and the requirements of 10 CFR Part 50, Appendix B. The implementation of the guidance in MRP-227-A, in conjunction with NEI 03-08, and other guidance documents, reports or methodologies referenced in this AMP, provide an acceptable level of quality and basis for confirming the quality of inspections, flaw evaluations and corrective actions.
9	Administrative Controls	The administrative controls for the St. Lucie RVI Inspection Program, including its implementing procedure and review and approval processes, are implemented in accordance with the recommended industry guidelines and criteria in NEI 03-08, and are under the site 10 CFR 50 Appendix B, Quality Assurance Program.
10	Operating Experience	The review and assessment of relevant operating experience for impact on the St. Lucie RVI Inspection Program are governed by NEI 03-08 and Appendix A of MRP-227-A. The reporting of inspection results and operating experience is treated as a "Needed" category item under NEI 03-08.

<u>RAI-MF6777/MF6778-EVIB-03</u>

Applicant/Licensee Action Item (A/LAI) 1 essentially requires an applicant or licensee to verify the applicability of the MRP-227-A guidelines to its plant. One of the issues for plant-specific applicability is assurance that the assessment of susceptibility to stress corrosion cracking (SCC) in MRP-227-A is bounding for the plant. This assessment is predicated on the stainless steel components meeting certain criteria for cold work and stress.

As discussed in the enclosure to MRP Letter 2013-025 (Ref. 7) entitled "MRP-227-A Applicability Guidelines for Combustion Engineering and Westinghouse Pressurized Water Reactor Designs," do St. Lucie Plant Units 1 and 2 have non-weld or bolting austenitic stainless steel components with 20 percent cold work or greater, and, if so, do the affected components have operating stresses greater than 30 kilopounds-per-square inch? If St. Lucie Plant has such components, provide a plant-specific aging management recommendation for SCC of these components.

<u>Response</u>
St. Lucie is actively participating in a joint industry program under the PWROG aimed at addressing the 20% cold work issue for non-weld or bolting austenitic stainless steel components on a generic rather than plant-specific basis. A discussion of this ongoing program (PA) follows:

<u>PA-MSC-1288</u>, PWR Materials Assessment, was discussed with the NRC at the June 2-4, 2015 Annual Materials Programs Technical Information Exchange Public Meeting (Ref. ML15155B431). This PA utilizes a statistical approach for determining and assessing material or fabrication factors for PWR internals components. To date, plant-specific component manufacturing records have been gathered for over 50% of the domestic PWRs. A review of these records in accordance with the guidance provided in MRP 2013-025 (ML1322A454) has revealed the following:

- 20% cold work limitation was already recognized at the time of plant construction, i.e. from 1970's
- Plant fabricators quality programs were in place to adhere to limitations in cold work in austenitic stainless steels in these times
- Plant specific assessments conducted to date confirm that no non-fastener materials contain cold work greater than 20%
- Correlation of data based on searches to date demonstrates consistency across the PWR fleet - B&W, CE and, W show no cold worked non-fastener materials used in reactor vessel internals

A final report for this PA is scheduled to be issued in April 2017. FPL will continue to participate in and follow the progress of PA-MSC-1288, including interactions with the NRC. If necessary, plant specific information will be provided to the NRC to supplement this joint industry program.

<u>RAI-MF6777/MF6778-EVIB-04.</u>

In the licensee's response to A/LAI 1, the licensee stated that St. Lucie Plant RVI component materials are consistent, or nearly equivalent to the materials identified in MRP-191, Table 4-5. The licensee also stated that where differences exist, that either there is no impact due to the differences, or the components are being managed by an alternate AMP. The staff requests that the licensee:

1. Identify the components fabricated from different materials than assumed in MRP-191, Table 4-5. Identify the material type/grade (e.g., Type 316 stainless steel) used for these components at St. Lucie Plant.

Response
The PSL Units 1 and 2 ICI guide tubes are constructed of 304 SS, whereas MRP-191 indicates these components are constructed of 316 SS.

2. Provide a justification for the determination that there is no impact on the categorization of these components.

Response
Both 304 SS and 316 SS are wrought austenitic stainless steels. The screening criteria for all eight age related degradation mechanisms addressed by MRP-227-A and MRP-191 are the same for both alloys. Therefore, no change to the categorization of the 304 SS ICI guide tubes is required.

3. Identify the alternate AMP(s) that will be used to manage aging of certain RVI components with materials that differ from MRP-191. Explain how these components are adequately managed by the alternate program(s).

Response
No alternate AMP is required to manage aging of the St. Lucie Units 1 and 2 ICI guide tubes based upon the response to item 2 above.

RAI-MF6777/MF6778-EVIB-05

In the licensee's response to A/LAI 1, the licensee stated that an 11.85% EPU was performed on St. Lucie Plant, and that 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. For St. Lucie Plant Unit 1, the response to RAI CVIB-5 related to the EPU (Ref. 8) stated that a detailed fluence analysis of the reactor pressure vessel (from the interior of the core shroud plates through the vessel wall around the mid-plane) was used to determine fluence through the various RVI components, and that the fluence calculation adhered to the requirements of Regulatory Guide 1.190 with regard to method and uncertainty.

For St. Lucie Plant Unit 2, the EPU Licensing Report (Ref. 9) also implies that a detailed neutron fluence analysis was performed similar to that for St. Lucie Plant Unit 1. The staff therefore requests that the licensee describe how the fluence analysis of the St. Lucie Plant Unit 2 RVI was performed in support of the EPU, or confirm the methodology used was the same as for St. Lucie Plant Unit 1.

Response
Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

RAI-MF6777/MF6778-EVIB-06

In the staff's safety evaluation related to the EPU for St. Lucie Plant Unit 1 (Ref. 2), the staff concluded that it has reviewed the licensee's evaluation of the effects of the proposed EPU on the susceptibility of RVI to known degradation mechanisms and concludes that the licensee has identified appropriate degradation management programs to address the effects of changes in operating temperature and neutron fluence on the integrity of these components. The staff reached a similar conclusion in its safety evaluation related to the EPU for St. Lucie Plant Unit 2. However, the staff *notes that in its evaluation of RVI aging considering EPU, the licensee determined that some components are susceptible to certain aging mechanisms, which were screened out in the development process of MRP-227-A. For example, the EPU Licensing Reports for St. Lucie Plant Unit 1 (Ref. 10) and St. Lucie Plant Unit 2 (Ref. 9) list the fuel alignment plate, upper guide structure support plate, control element assembly (CEA) shroud tubes, and CEA shroud bolts and locking bars as susceptible to loss of fracture toughness due to irradiation embrittlement (IE), while MRP-191 screened out these components for IE. The EPU licensing reports also identified the CEA flow channel parts as susceptible to IE, which are a plant-specific component. There is no equivalent generic component in MRP-191. Similarly, the EPU Licensing Report for St. Lucie Plant Unit 1, and St. Lucie Plant Unit 2, list the fuel alignment plate, upper guide structure support plate, CEA shrouds (lower part), and CEA shroud bolts and locking bars as components susceptible to irradiation assisted stress corrosion cracking (IASCC), while MRP-191 screened out these components for IASCC. The staff therefore requests the licensee:*

1. Provide the fluence screening criteria it used for IE and IASCC, if different than the screening criteria of MRP-191.

Response

Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

2. Confirm whether the components listed above actually exceed the MRP-191 fluence screening criteria.

Response

Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

3. If any of the components listed above exceed the MRP-191 fluence screening criteria, provide the estimated fluence for those components considering EPU at the end of life.

Response

Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

4. If the components do exceed the screening criteria, explain how MRP-227-A is bounding (provides for appropriate aging management) for St. Lucie Plant Units 1 and 2, considering that the fluences for these components exceed the MRP-191 screening limits.

Response

Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

5. Finally, if MRP-227-A is not bounding for any specific components, provide a plant-specific aging management recommendation for such components.

Response
Based on several discussions between FPL and Westinghouse, it is anticipated that FPL will receive a proposal to perform the work (scope and schedule) requested in items 5 & 6 by March 31, 2016. FPL will advise the NRC of the schedule once we receive the same.

RAI-MF6777/MF6778-EVIB-07

A/LAI 2 essentially requires an applicant or licensee to identify any plant-specific RVI components and modify its program as necessary to manage aging of such components. In its response to A/LAI 2, the licensee identified the CEA shroud flow bypass inserts as a plant-specific component for St. Lucie Plant Unit 2. The licensee indicated that it categorized the St. Lucie Plant Unit 2 CEA shroud flow bypass inserts consistently with the categorization of the generic CEA shroud components in MRP-191 as Category A. The licensee stated that it therefore categorized the Unit 2 flow bypass inserts consistently, making them "No Additional Measures Components." Thus, the licensee stated no further action is required for managing aging of these RVI components.

The staff needs to verify that the CEA flow bypass inserts are categorized consistently with the generic CEA shroud components.

The staff therefore requests the licensee provide details on the failure modes, effects, and consequences analysis of the CEA flow bypass inserts, including the component functions, material, screened-in degradation mechanisms, consequences of failure, likelihood of failure, and likelihood of damage (conditional core damage likelihood).

Response

Failure modes, effects, and criticality analysis (FMECA) results for Combustion Engineering (CE) nuclear steam supply system reactor vessel internals components were summarized in Materials Reliability Program (MRP) report MRP-191 [4] (U.S.NRC ADAMS ML12335A503). A detailed discussion for the U.S. NRC staff regarding the MRP-227-A process, including the FMECA process results and MRP-191 results, was provided on November 28, 2012 (U.S. NRC ADAMS ML12335A040). Further investigation by FPL and Westinghouse has determined that the control element assembly (CEA) flow bypass inserts were indeed included as part of the CEA in the MRP-191 generic industry activity and appropriately assigned a "No Additional Measures" categorization. FPL's initial response to A/LAI 2 was incorrect concerning the CEA flow bypass inserts' exclusion from MRP-191.

RAI-MF6777/MF6778-EVIB-08

A/LAI 5 requires applicants/licensees to identify plant-specific acceptance criteria to be applied when performing the physical measurements required by MRP-227-A for several components, including 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 (such as St. Lucie Plant Units 1 and 2). A/LAI 5 further requires that 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 as part of their submittal to apply MRP-227-A.

In its response to A/LAI 5, the licensee stated that core shroud gap acceptance criteria have been developed for St. Lucie Plant Units 1 and 2 that are resolvable using the specified VT-1 inspection method of MRP-227-A. The licensee further stated that plant-specific details are proprietary and not *typically released publicly, but if the NRC requests additional details, the calculation can be made available for review. The licensee concluded that this satisfies the requirements of A/LAI 5.*

The licensee provided no detail on the methodology or results of the analysis used to develop the core shroud gap acceptance criteria for St. Lucie Plant Units 1 and 2. The staff therefore requests that the licensee make the calculation available for review by the staff, either by submitting it on the docket, or making it available for an audit. If made available for audit only, it is possible that the staff will need to issue a follow-up RAI to request specific information from the calculation be submitted on the docket.

Response
A non-proprietary version of the calculation is enclosed.

RAI-MF6777/MF6778-EVIB-09

A/LAI 7 requires an applicant or licensee to provide an evaluation demonstrating that cast austenitic stainless steel (CASS) RVI components will maintain their functionality throughout the period of extended operation (PEO), considering the potential loss of fracture toughness due to both thermal embrittlement (TE) and IE.

In its response to A/LAI 7, the licensee identified the RVI components that are fabricated from CASS as St. Lucie Plant Unit 1 core support columns, the CEA shroud tubes for both units, and the St. Lucie Plant Unit 2 flow bypass inserts. The licensee indicated that all but one of the Unit 1 core support columns screen in for TE based on the assumption that the columns have ferrite > 20%, since certified material test reports could not be located for these columns.

The licensee then concluded that the results of this evaluation do not conflict with strategy for aging management of RVI provided in MRP-227-A. The licensee stated that it is concluded that continued application of the strategies in MRP-227-A and the St. Lucie Plant Units 1 and 2 RVI Inspection Program will meet the requirements for managing age-related degradation of St. Lucie Plant Units 1 and 2, CASS and martensitic stainless steel RVI components. However, the licensee did not provide any justification for its position that the MRP-227-A aging management requirements (which require no inspections of the core support columns) are sufficient, considering the potential for loss of fracture toughness due to two mechanisms, and the susceptibility to cracking of the columns. The staff notes that the core support column welds, which are visible from above the core support plate, are inspected as Primary components, but MRP-227-A and the St. Lucie Plant Unit 1 RVI AMP require no expansion to the columns if degradation is detected in the welds.

Since the St. Lucie Plant Unit 1 core support columns (except one) are screened in for TE, and are also susceptible to IE and several cracking mechanisms, the staff requests the licensee provide an evaluation for St. Lucie Plant Unit 1, demonstrating that the core support columns will remain functional during the PEO considering the potential combined loss of fracture toughness due to TE plus IE, along with the potential for cracking in the columns.

Response

St. Lucie is actively participating in two joint industry programs under the PWROG aimed at addressing the functionality of CASS RVI components during the period of extended operation, considering the loss of fracture toughness due to both thermal embrittlement (TE) and irradiation embrittlement (TE). A discussion of each of these ongoing programs (PAs) follows.

PA-MSC-1288, PWR Materials Assessment, was discussed with the NRC at the June 2-4, 2015 Annual Materials Programs Technical Information Exchange Public Meeting (Ref. ML15155B431). This PA utilizes a statistical approach for determining and assessing material or fabrication factors for PWR internals components. A complete survey of the estimations of the ferrite content and saturation toughness after thermal embrittlement (TE) was performed for all the CASS components that had been evaluated by the industry to date. Estimated ferrite contents of the CASS components were calculated from alloy chemical compositions reported in the plant-specific certified material test reports (CMTR), using the Hull's Factors method as described in NUREG/CR-4513. Saturation fracture toughness values after TE were also calculated using procedures outlined in that same document. The data were then analyzed statistically, both on an overall basis and with respect to parameters that were expected to affect the ferrite content and the saturation toughness of the CASS; specifically, manufacturer, plant, vintage of production, and component.

The initial results of PA-MSC-1288 were published in PWROG-15032-NP, Rev. 0, and discussed with the NRC in a meeting between the NRC and PWROG on September 16, 2015, Presentation WAAP-9551, "PA-MSC-1288R0: PWR Materials Assessment Results. (See page 78 of PWROG-15032-NP for record meeting with the NRC). A lognormal distribution was found to provide a reasonable fit to the overall distribution of ferrite content in static-cast Grade CF8 RV internals components. This allowed a 95/95 ferrite content upper bound of 17.5 percent to be calculated, below the 20 percent ferrite screening criteria for thermal embrittlement (TE) established by the Grimes letter [NUREG/CR-4513]. A 95/95 upper bound based on the mean ferrite content of material produced by each individual manufacturer was also calculated, and agreed well (17.4 percent) with the value calculated from the lognormal distribution. Further, for the Grade CF8 materials, it was demonstrated that for all compositions (including those with over 20 percent ferrite) fracture toughness greater than the screening criterion value was retained after TE effects had saturated.

Although, the CMTRs for the majority of the St. Lucie Unit 1 core support columns could not be located, the likely manufacturer was Kearsage based upon the vintage of these components. To date a review of 498 individual heats of static cast materials produced by Kearsage under PA-MSC-1288 has revealed a lognormal distribution, with average ferrite content of 6.2% and standard deviation of 1.7%. The 95/95 limit for ferrite content for static cast Kearsage material was 9.7%. Therefore, it is reasonable to conclude that the ferrite contents of the St. Lucie Unit 1 core support columns fall well below the TE threshold.

PA-MSC-1103, Functionality Analysis of Westinghouse Lower Support Columns, is another PWROG project that ultimately strives to develop of a generic functionality analysis for CASS lower support columns that will bound all Westinghouse and CE designed plants, thereby satisfying LAI 7. The initial task of the PA focused on the development of generic methodology for performing a functionality analysis of the lower support columns. The initial report, PWROG-14048-P, was submitted to the NRC for review in March 2015 (ML15077A113). Subsequently, the NRC issued its Summary Assessment of Report PWROG-14048-P in December 2015 (ML15334A462). The NRC generally concurred with the methodology of PWROG-14048-P, provided that plant specific design parameters and that buckling and changes in modal characteristics in the lower support structure are considered in the redundancy analysis.

Ongoing tasks of PA-MSC-1103 will attempt to address NRC concerns and permit fleet wide applicability of a bounding functionality analysis. Major work will include: 1) a comprehensive review of fleet-wide design parameters related to lower support column functionality; 2) the establishment of representative plant designs needed to cover all fleet-wide design consideration; and 3) conducting a comprehensive loading parameter review. The estimated completion date of these tasks is mid-2017.

FPL will continue to participate in and follow the progress of PA-MSC-1288 and PA-MSC-1103, including interactions with the NRC. It is hoped that the results of these joint industry programs will provide adequate details to demonstrate the functionality of St. Lucie Unit 1 CASS core support columns during the period of extended operation, considering the loss of fracture toughness due to both thermal embrittlement (TE) and irradiation embrittlement (TE). If necessary, plant specific information will be provided to the NRC to supplement these joint industry programs

RAI-MF6777/MF6778-EVIB-10

The Expansion Link column in Table 1 of the RVI AMP, "CE Plants Primary Components," lists the lower cylinder axial welds as the expansion link for the core support column welds. However, this appears to be an error because Table 4 of the AMP, "CE Plant Examination Acceptance and Expansion Criteria," has "none" in the Expansion Link column for the same component, and MRP-227-A specifies no expansion link for the core support column welds for CE plants. The staff requests the licensee correct this error. If not an error, justify the plant-specific expansion link.

Response

The error noted in Table 1 of the St. Lucie RVI Inspection Program has been corrected as shown below:

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)	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 the accessible surfaces of the core support column welds. (Note 5) See Figure 4-16 and 4-31, MRP-227-A.