October 20, 2014

L-2014-318
10 CFR 50.4
10 CFR 50.55a

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

Re: St. Lucie Unit 2
Docket Nos. 50-389
In-Service Inspection Plans
Fourth Ten-Year Interval
Unit 2 Relief Request 7

Pursuant to 10CFR50.55a(a)(3)(i), Florida Power & Light (FPL) requests to revise the St. Lucie Unit 2 ISI Program for Class 1 and 2 piping through the use of the Risk-Informed Inservice Inspection Program (RI-ISI), Attachment 1, as an alternative to the current requirements of Class 1 and 2 examination Categories B-F, B-J, C-F-1, and C-F-2 as specified in Table IWB-2500-1 and Table IWC-2500-1 of the 2007 Edition with 2008 Addenda of ASME Section XI.

The justification for this relief is contained in the Attachment.

Please contact Ken Frehafer at 772-467-7748 if there are any questions about this submittal.

Very truly yours,

Eric Katzman
Licensing Manager
St. Lucie Plant

Attachment

ESK/KWF

cc: USNRC Regional Administrator, Region II
USNRC Senior Resident Inspector, St. Lucie Units 1 and 2

A047
NRR

St. Lucie Unit 2
FOURTH INSPECTION INTERVAL
10CFR50.55a RELIEF REQUEST NUMBER 7

Proposed Alternative
In Accordance with 10 CFR 50.55a(a)(3)(i)

--Alternative Provides Acceptable Level of Quality and Safety--

1. ASME Code Components Affected

Class 1 and 2 pressure retaining similar and dissimilar metal piping welds

Exam Cat.	Item No.	Examination Description
B-F	B5.40	Pressurizer- NPS 4 or larger, Nozzle-to-Safe End Butt Welds
	B5.50	Pressurizer- Less than NPS 4, Nozzle-to-Safe End Butt Welds
B-J	B9.11	Piping- NPS 4 or Larger, Circumferential Welds
	B9.21	Piping- Less than NPS 4, Circumferential Welds
	B9.31	Piping- Branch Pipe Connection Welds, NPS 4 or Larger
	B9.32	Piping- Branch Pipe Connection Welds, Less than NPS 4
	B9.40	Piping- Socket Welds
C-F-1	C5.11	Circumferential Welds ≥ 3/8 in. on piping > NPS 4
	C5.21	Circumferential Welds > 1/5 in. on piping ≤ NPS 4 and ≥ NPS 2
	C5.30	Piping- Socket Welds
	C5.41	Branch Pipe Connection Circumferential Welds ≥ NPS 2
C-F-2	C5.51	Circumferential Welds ≥ 3/8 in. on piping > NPS 4
	C5.61	Circumferential Welds > 1/5 in. on piping ≥ NPS 2 and ≤ NPS 4
	C5.81	Branch Pipe Connection Circumferential Welds ≥ NPS 2

2. Applicable Code Edition and Addenda

Inservice inspections (ISI) are performed on piping to the requirements of the ASME Boiler and Pressure Vessel Code Section XI, 2007 Edition with 2008 Addenda as required by 10CFR50.55a.

3. Applicable Code Requirement

Pursuant to 10CFR50.55a(a)(3)(i), FPL requests to revise the St. Lucie Unit 2 ISI Program for Class 1 and 2 piping through the use of the Risk-Informed Inservice Inspection Program (RI-ISI), Attachment 1, as an alternative to the current requirements of Class 1 and 2

examination Categories B-F, B-J, C-F-1, and C-F-2 as specified in Table IWB-2500-1 and Table IWC-2500-1 of the 2007 Edition with 2008 Addenda of ASME Section XI.

4. Reason for Request

The objective of this submittal is to request a conversion of the St. Lucie 2 risk-informed inservice inspection (RI-ISI) Program for Class 1 piping welds during the fourth interval to a different process than that used during the previous interval. In addition, this RI-ISI process will be extended to include Class 2 piping welds. The RI-ISI process used in this submittal is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A, "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578-1 "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B," and ASME Section XI Nonmandatory Appendix R, "Risk-Informed Inspection Requirements for Piping."

5. Proposed Alternatives and Basis for Use

Proposed Alternative

ASME Section XI Examination Categories B-F, B-J, C-F-1, and C-F-2 contain the requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components. St. Lucie 2 had previously implemented a RI-ISI Program for Class 1 piping welds in accordance with WCAP-14572. The proposed alternative RI-ISI program for piping is described in EPRI TR-112657. The RI-ISI program will be substituted for Class 1 and 2 piping welds (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10CFR50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. EPRI TR-112657 provides the requirements for defining the relationship between the RI-ISI program and the remaining unaffected portions of ASME Section XI.

Basis for Use

The attached Risk-Informed Inservice Inspection Program Plan (see Attachment 1) establishes the basis for use and supports the conclusion that the proposed alternative provides an acceptable level of quality and safety. This Program Plan is provided in a standardized format that has been used and accepted throughout the industry for over ten years to document initial RI-ISI applications.

Additionally, this submittal meets the intent and principles of Regulatory Guides 1.174 and 1.178.

6. Duration of Proposed Alternative

This relief is requested for the duration of the Fourth Inservice Inspection Interval, which begins on August 8, 2013 and is scheduled to end on August 7, 2023.

St. Lucie Unit 2
FOURTH INSPECTION INTERVAL
10CFR50.55a RELIEF REQUEST NUMBER 7

7. Precedents

The NRC has approved similar requests for alternatives at numerous other nuclear facilities, including the following:

- Relief Request No. A-1 was approved for the Comanche Peak Nuclear Power Plant in a letter from the NRC dated October 5, 2006, ADAMS Accession No. ML062750371.

- Relief Request No. IR-3-01 was approved for the Callaway Energy Center in a letter from the NRC dated January 3, 2007, ADAMS Accession No. ML063520007.

- Relief Request No. RR-III-02 was approved for the V. C. Summer Nuclear Station in a letter from the NRC dated September 6, 2005, ADAMS Accession No. ML052300616.

- Relief Request No. RI-ISI-INT3 was approved for the Diablo Canyon Power Plant in a letter from the NRC dated January 16, 2013, ADAMS Accession No. ML12353A130.

8. Attachments to the Relief

Attachment 1 - "Risk-Informed Inservice Inspection Program Plan for St. Lucie Nuclear Power Plant, Unit 2"

RISK-INFORMED INSERVICE INSPECTION PROGRAM PLAN
ST. LUCIE NUCLEAR POWER PLANT, UNIT 2

Table of Contents

1. INTRODUCTION

The St. Lucie Nuclear Power Plant Unit 2 (PSL-2) is currently in the fourth inservice inspection (ISI) interval as defined by the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Section XI Code. The fourth ISI interval for St. Lucie 2 began on August 8, 2013 and is scheduled to end on August 7, 2023. Pursuant to 10CFR50.55a(g)(4)(ii), the applicable ASME Section XI Code for the fourth ISI interval is the 2007 Edition through 2008 Addenda.

By letter to the NRC on August 6, 2003, supplemented by letters dated September 17 and December 28, 2004, Florida Power & Light (FPL) requested relief for St. Lucie 2 from the ASME Section XI Code examination requirements of Class 1 piping weld (Examination Categories B-F and B-J) inservice inspections by implementing a Risk-Informed Inservice Inspection (RI-ISI) Program. The alternative was authorized February 23, 2005 for the third ISI interval ending August 7, 2013.

The St. Lucie 2 RI-ISI Program was initially developed based on WCAP-14572, "Westinghouse Owners Group Application of Risk-Informed Methods to Piping Inservice Inspection Topical Report," Rev. 1-NP-A, with identified differences, and with additional guidance taken from ASME Code Case N-577, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method A, Section XI, Division 1".

The objective of this submittal is to request a conversion of the St. Lucie 2 RI-ISI Program for Class 1 piping welds during the fourth interval to a different process than that used during the previous interval. In addition, this RI-ISI process will be applied to Class 2 piping welds. The RI-ISI process used in this submittal is described in Electric Power Research Institute (EPRI) Topical Report (TR) 112657 Rev. B-A "Revised Risk-Informed Inservice Inspection Evaluation Procedure." The RI-ISI application was also conducted in a manner consistent with ASME Code Case N-578-1 "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B," and ASME Section XI Nonmandatory Appendix R, "Risk-Informed Inspection Requirements for Piping."

1.1 Relation to NRC Regulatory Guides 1.174 and 1.178

As a risk-informed application, this submittal meets the intent and principles of Regulatory Guide 1.174 "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis" and Regulatory Guide 1.178 "An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping". Further information is provided in Section 3.6.2 relative to defense-in-depth.

1.2 PRA Quality

An evaluation of the St. Lucie Probabilistic Risk Assessment (PRA) capabilities as measured against the current ASME/ANS PRA Standard (RA-Sa-2009) as endorsed by U.S. NRC Regulatory Guide 1.200 is provided in Attachment A to this document. This evaluation establishes the technical adequacy of the St. Lucie PRA with respect to the RI-ISI Program. The evaluation is based on a series of formal pier reviews and internal self-assessments. See Attachment A for details.

2. PROPOSED ALTERNATIVE TO CURRENT ISI PROGRAM REQUIREMENTS

2.1 ASME Section XI

ASME Section XI Examination Categories B-F, B-J, C-F-1 and C-F-2 contain the requirements for the nondestructive examination (NDE) of Class 1 and 2 piping components. In addition, St. Lucie 2 had previously implemented a RI-ISI Program for Class 1 piping welds in accordance with WCAP-14572. The proposed alternative RI-ISI program for piping is described in EPRI TR-112657. The RI-ISI program will be substituted for Class 1 and 2 piping welds (Examination Categories B-F, B-J, C-F-1 and C-F-2) in accordance with 10CFR50.55a(a)(3)(i) by alternatively providing an acceptable level of quality and safety. Other non-related portions of the ASME Section XI Code will be unaffected. EPRI TR-112657 provides the requirements for defining the relationship between the RI-ISI program and the remaining unaffected portions of ASME Section XI.

2.2 Augmented Programs

The plant augmented inspection programs listed below were considered during the RI-ISI application. It should be noted that this section documents only those plant augmented inspection programs that address common piping with the RI-ISI application scope (i.e., Class 1 and 2 piping).

- St. Lucie 2 currently maintains an augmented examination program on the pressurizer surge line as a result of NRC Bulletin 88-11. Visual, VT-1 examinations are performed on this piping to monitor for the potential effects of thermal stratification. Since the RI-ISI Program does not address visual examinations, this augmented examination program is not changed by the RI-ISI application and will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program for alloy 600/82/182 butt welds to examine for Primary Water Stress Corrosion Cracking (PWSCC). In accordance with 10CFR50.55a(g)(6)(ii)(F), these welds are subject to the criteria of ASME Section XI Code Case N-770-1. The examination of welds due to PWSCC is considered administratively during the RI-ISI application, but the Code Case N-770-1 program takes precedence. Therefore, welds subject to PWSCC are selected for examination per Code Case N-770-1 and examined under that program. Welds for which no other degradation mechanism has been postulated will be examined solely under the Code Case N-770-1 Program and were removed from consideration during the RI-ISI element selection process. Welds for which another degradation mechanism other than PWSCC has been postulated were considered for further examination in the RI-ISI application in the same population as those subject to the additional degradation mechanism. However, the Code Case N-770-1 augmented examination program is not changed by the RI-ISI application and will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program to manage thermal fatigue in normally non-isolable reactor coolant system branch lines in accordance with MRP-146. The RI-ISI application considers this degradation mechanism during the application, but does not include all the criteria of MRP-146. Therefore, this augmented examination program will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program for alloy 600/82/182 butt welds to examine for Primary Water Stress Corrosion Cracking (PWSCC). In accordance with 10CFR50.55a(g)(6)(ii)(E), these welds are subject to the criteria of ASME Section XI Code Case N-722-1. Visual, VE examinations are performed on these piping welds to monitor for the potential effects of PWSCC. Since the RI-ISI Program does not address visual examinations, this augmented examination program is not changed by the RI-ISI application and will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program for piping welds in high energy main steam and main Feedwater piping in accordance with NUREG-0800, Standard Review Plan 3.6.2, Branch Technical Position MEB 3-1. The examination of welds due to high energy line break concerns is not addressed in a standard RI-ISI application. Therefore, this augmented examination program is not changed by the RI-ISI application and will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program on Feedwater piping as a result of the continuation of NRC Bulletin 79-13 and NRC Information Notice 93-20. Enhanced ultrasonic examinations are performed starting at the Feedwater nozzle ramp and extending out onto the elbow. The RI-ISI application considers this degradation mechanism during the application, but does not include all the criteria of this program. Therefore, this augmented examination program will remain in effect.

- St. Lucie 2 currently maintains an augmented examination program for piping in accordance with USNRC Generic Letter 89-08 to examine for Flow Accelerated Corrosion (FAC). The examination of piping due to FAC is considered administratively during the RI-ISI application, but the existing FAC program takes precedence. Therefore, this augmented examination program is not changed by the RI-ISI application and will remain in effect.

3. RISK-INFORMED ISI PROCESS

The process used to develop the RI-ISI program conformed to the methodology described in EPRI TR-112657 and consisted of the following steps:

- Scope Definition
- Consequence Evaluation
- Failure Potential Assessment
- Risk Characterization
- Element and NDE Selection
- Risk Impact Assessment
- Implementation Program
- Feedback Loop

A deviation to the EPRI RI-ISI methodology has been implemented in the failure potential assessment for St. Lucie 2. Table 3-16 contains the criteria for thermal stratification, cycling and

striping (TASCS). Key attributes for horizontal or slightly sloped piping greater than 1" nominal pipe size (NPS) include:

1. Potential exists for low flow in a pipe section connected to a component allowing mixing of hot and cold fluids, or
2. Potential exists for leakage flow past a valve, including in-leakage, out-leakage and cross-leakage allowing mixing of hot and cold fluids, or
3. Potential exists for convective heating in dead-ended pipe sections connected to a source of hot fluid, or
4. Potential exists for two phase (steam/water) flow, or
5. Potential exists for turbulent penetration into a relatively colder branch pipe connected to header piping containing hot fluid with turbulent flow,

AND

$\Delta T > 50^{o}F$,

AND

Richardson Number > 4 (This value predicts the potential buoyancy of stratified flow.)

These criteria, based on meeting a high cycle fatigue endurance limit with the actual ΔT assumed equal to the greatest potential ΔT for the transient, will identify all locations where stratification is likely to occur, but allows for no assessment of severity. As such, many locations will be identified as subject to TASCS where no significant potential for thermal fatigue exists. The critical attribute missing from the existing methodology that would allow consideration of fatigue severity is a criterion that addresses the potential for fluid cycling. The impact of this additional consideration on the existing TASCS criteria is presented below.

- **Turbulent penetration TASCS**

  Turbulent penetration typically occurs in lines connected to piping containing hot flowing fluid. In the case of downward sloping lines that then turn horizontal, significant top-to-bottom cyclic ΔTs can develop in the horizontal sections if the horizontal section is less than about 25 pipe diameters from the reactor coolant piping. Therefore, TASCS is considered for this configuration.

  For upward sloping branch lines connected to the hot fluid source that turn horizontal or in horizontal branch lines, natural convective effects combined with effects of turbulence penetration will keep the line filled with hot water. If there is no potential for in-leakage towards the hot fluid source from the outboard end of the line, this will result in a well-mixed fluid condition where significant top-to-bottom ΔTs will not occur. Therefore TASCS is not considered for these configurations. Even in fairly long lines, where some heat loss from the outside of the piping will tend to occur and some fluid stratification may be present, there is no significant potential for cycling as has been observed for the in-leakage case. The effect of TASCS will not be significant under these conditions and can be neglected.

- **Low flow TASCS**

  In some situations, the transient startup of a system (e.g., SDC suction piping) creates the potential for fluid stratification as flow is established. In cases where no cold fluid source exists, the hot flowing fluid will fairly rapidly displace the cold fluid in stagnant lines, while fluid mixing will occur in the piping further removed from the hot source and stratified conditions will exist only briefly as the line fills with hot fluid. As such, since the situation is transient in nature, it can be assumed that the criteria for thermal transients (TT) will govern.

- **Valve leakage TASCS**

  Sometimes a very small leakage flow can occur outward past a valve into a line with a significant temperature difference. However, since this is a generally a "steady-state" phenomenon with no potential for cyclic temperature changes, the effect of TASCS is not significant and can be neglected.

- **Convection heating TASCS**

  Similarly, there sometimes exists the potential for heat transfer across a valve to an isolated section beyond the valve, resulting in fluid stratification due to natural convection. However, since there is no potential for cyclic temperature changes in this case, the effect of TASCS is not significant and can be neglected.

In summary, these additional considerations for determining the potential for thermal fatigue as a result of the effects of TASCS provide an allowance for the consideration of cycle severity in assessing the potential for TASCS effects. The above criteria have previously been submitted by EPRI for generic approval (Letters dated February 28, 2001, and March 28, 2001, P.J. O'Regan (EPRI) to Dr. B. Sheron (USNRC), "Extension of Risk-Informed Inservice Inspection Methodology").

3.1 Scope of Program

The systems included in the RI-ISI program are provided in Tables 3.1. The piping and instrumentation diagrams and additional plant information including the existing plant ISI program were used to define the Class 1 and 2 piping system boundaries. In Table 3.1, consequence segments were defined as continuous runs of piping whose failure would result in the same consequence.

3.2 Consequence Evaluation

The consequence(s) of pressure boundary failures were evaluated and ranked based on their impact on core damage and containment performance (isolation, bypass and large, early release). The impact on these measures due to both direct and indirect effects was considered using the guidance provided in EPRI TR-112657.

3.3 Failure Potential Assessment

Failure potential estimates were generated utilizing industry failure history, plant specific failure history and other relevant information. These failure estimates were determined using the guidance provided in EPRI TR-112657, with the exception of the previously stated deviation.

Table 3.3 summarizes the failure potential assessment by system for each degradation mechanism that was identified as potentially operative for St. Lucie 2.

3.4 Risk Characterization

In the preceding steps, each run of piping within the scope of the program was evaluated to determine its impact on core damage and containment performance (isolation, bypass and large, early release) as well as its potential for failure. Given the results of these steps, risk groups are then defined as welds within a single system potentially susceptible to the same degradation mechanism and whose failure would result in the same consequence. Risk groups are then ranked based upon their risk significance as defined in EPRI TR-112657.

The results of these calculations are presented in Table 3.4.

3.5 Element and NDE Selection

In general, EPRI TR-112657 requires that 25% of the locations in the high risk region and 10% of the locations in the medium risk region be selected for inspection using appropriate NDE methods tailored to the applicable degradation mechanism. In addition, per Section 3.6.4.2 of EPRI TR-112657, if the percentage of Class 1 piping locations selected for examination falls substantially below 10%, then the basis for selection needs to be investigated.

For St. Lucie 2, the percentage of Class 1 welds selected per the RI-ISI process was 8.7% (49 of 562 welds), which is not a significant departure from 10%.

One additional factor that was considered during the St. Lucie 2 evaluation was that the overall percentage of Class 1 selections included both socket and non-socket welds. Therefore, the final percentage of Class 1 selections was 8.7% when both socket and non-socket piping welds were considered. This percentage increases to 11% (41 of 373 welds) when considering only those piping welds that are non-socket welded. It should be noted that non-socket welds are subject to volumetric examination, so this percentage does not rely upon welds that are solely subject to a VT-2 visual examination.

As stated in TR-112657, the existing FAC augmented inspection program provides the means to effectively manage this mechanism. No additional credit was taken for any FAC augmented examinations. In addition, no credit was taken for any PWSCC examinations performed in accordance with Code Case N-770-1.

A brief summary is provided in the following table, and the results of the selections are presented in Tables 3.5-1 and 3.5-2. Section 4 of EPRI TR-112657 was used as guidance in determining the examination requirements for these locations.

St. Lucie 2 Element Selection									
Unit	Class 1 Piping Welds(1)(2)			Class 2 Piping Welds(3)			All Piping Welds(4)		
	Population	Selected	% (2)	Population	Selected	%	Population	Selected	%
2	562	49	8.7%	1774	56	3.2%	2336	105	4.5%

Notes

1. Includes all Category B-F and B-J locations.
2. Includes both butt and socket welds
3. Includes all Category C-F-1 and C-F-2 locations.
4. All in-scope piping components, regardless of risk classification, will continue to receive Code required pressure testing, as part of the current ASME Section XI program. VT-2 visual examinations are scheduled in accordance with the station's pressure test program that remains unaffected by the RI-ISI program.

3.5.1 Additional Examinations

The RI-ISI program in all cases will determine through an engineering evaluation the root cause of any unacceptable flaw or relevant condition found during examination. The evaluation will include the applicable service conditions and degradation mechanisms to establish that the element(s) will still perform their intended safety function during subsequent operation. Elements not meeting this requirement will be repaired or replaced.

The evaluation will include whether other elements in the segment or additional segments are subject to the same root cause conditions. Additional examinations will be performed on those elements with the same root cause conditions or degradation mechanisms. The additional examinations will include high risk significant elements and medium risk significant elements, if needed, up to a number equivalent to the number of elements required to be inspected on the segment or segments during the current outage. If unacceptable flaws or relevant conditions are again found similar to the initial problem, the remaining elements identified as susceptible will be examined. No additional examinations will be performed if there are no additional elements identified as being susceptible to the same root cause conditions.

3.5.2 Program Relief Requests

Since the RI-ISI application is being implemented near the start of the 4th Interval, there are no Relief Requests that will need to be withdrawn or modified as a result of the RI-ISI application.

3.6 Risk Impact Assessment

The RI-ISI program has been conducted in accordance with Regulatory Guide 1.174 and the requirements of EPRI TR-112657, and the risk from implementation of this program is expected to remain neutral or decrease when compared to that estimated from current requirements.

This evaluation identified the allocation of segments into High, Medium, and Low risk regions of the EPRI TR-112657 and ASME Code Case N-578-1 risk ranking matrix, and then determined for each of these risk classes what inspection changes are proposed for each of the locations in each segment. The changes include changing the number and location of inspections within the segment and in many cases improving the effectiveness of the inspection to account for the findings of the RI-ISI degradation mechanism assessment. For example, for locations subject to thermal fatigue, examinations will be conducted on an expanded volume and will be focused to enhance the probability of detection (POD) during the inspection process.

3.6.1 Quantitative Analysis

Limits are imposed by the EPRI methodology to ensure that the change in risk of implementing the RI-ISI program meets the requirements of Regulatory Guides 1.174 and 1.178. The EPRI criterion requires that the cumulative change in core damage frequency (CDF) and large early release frequency (LERF) be less than 1E-07 and 1E-08 per year per system, respectively.

St. Lucie 2 conducted a risk impact analysis per the requirements of Section 3.7 of EPRI TR-112657. The analysis estimates the net change in risk due to the positive and negative influence of adding and removing locations from the inspection program. A risk quantification was performed using the "Simplified Risk Quantification Method" described in Section 3.7 of EPRI TR-112657. The conditional core damage probability (CCDP) and conditional large early release probability (CLERP) used for high consequence category segments was based on the highest evaluated CCDP (3.33E-02) and CLERP (7.33E-04), whereas, for medium consequence category segments, bounding estimates of CCDP (1E-04) and CLERP (1E-05) were used. The likelihood of pressure boundary failure (PBF) is determined by the presence of different degradation mechanisms and the rank is based on the relative failure probability. The basic likelihood of PBF for a piping location with no degradation mechanism present is given as x_o and is expected to have a value less than 1E-08. Piping locations identified as medium failure potential have a likelihood of $20x_o$. These PBF likelihoods are consistent with References 9 and 14 of EPRI TR-112657. In addition, the analysis was performed both with and without taking credit for enhanced inspection effectiveness due to an increased POD from application of the RI-ISI approach.

Table 3.6 presents a summary of the RI-ISI program versus ASME Section XI Code Edition program requirements and identifies on a per system basis each applicable risk category. The potential presence of FAC or PWSCC was adjusted for in the performance of the quantitative analysis by excluding its impact on the risk ranking. FAC and PWSCC are excluded in the risk ranking and therefore in the determination of the change in risk, because FAC and PWSCC are damage mechanisms managed by separate, independent plant

augmented inspection programs. The RI-ISI Program credits and relies upon these augmented plant inspection programs to manage these damage mechanisms. The plant FAC and PWSCC Programs will continue to determine where and when examinations shall be performed. Hence, since the number of FAC and PWSCC examination locations remains the same "before" and "after" and no delta exists, there is no need to include the impact of FAC or PWSCC in the performance of the risk impact analysis. However, in an effort to be as informative as possible, for those systems where FAC or PWSCC may be present, the information in Table 3.6 is presented in such a manner as to depict what the resultant risk categorization is both with and without consideration of FAC and PWSCC. This is accomplished by enclosing the FAC or PWSCC damage mechanism, as well as all other resultant corresponding changes (failure potential rank, risk category and risk rank), in parenthesis. Again, this has only been done for information purposes, and has no impact on the assessment itself. The use of this approach to depict the impact of degradation mechanisms managed by augmented inspection programs on the risk categorization is consistent with that used in the risk impact analysis performed for plants throughout the United States. An example is provided below.

System	Risk		Consequence Rank	Failure Potential	
	Category	Rank[1]		DMs	Rank
FWS	5 (3)	Medium (High)	Medium	TASCS, TT, (FAC)	Medium (High)

In this example if FAC is not considered, the failure potential rank is "medium" instead of "high" based on the TASCS and TT damage mechanisms. When a "medium" failure potential rank is combined with a "medium" consequence rank, it results in risk category 5 ("medium" risk) being assigned instead of risk category 3 ("high" risk).

In this example if FAC were considered, the failure potential rank would be "high" instead of "medium". If a "high" failure potential rank were combined with a "medium" consequence rank, it would result in risk category 3 ("high" risk) being assigned instead of risk category 5 ("medium" risk).

Note

1. The risk rank is not included in Table 3.6 but it is included in Table 5-2.

As indicated in the table shown on the next page, the risk impact analysis has demonstrated that the change in risk associated with the St. Lucie 2 RI-ISI application meets the acceptance criteria of Regulatory Guide 1.174 and EPRI TR-112657 that the change in CDF must be less than 1E-07 and the change in LERF must be less than 1E-08 per year per system. In fact, the St. Lucie 2 RI-ISI application resulted in a reduction in risk in all instances.

St. Lucie 2 Risk Impact Results

System[1]	$\Delta Risk_{CDF}$		$\Delta Risk_{LERF}$	
	w/ POD	w/o POD	w/ POD	w/o POD
BF	-1.00E-08	1.00E-08	-2.23E-10	2.21E-10
CC	0.00E+00	0.00E+00	0.00E+00	0.00E+00
CH	-4.20E-08	-2.33E-08	-9.24E-10	-5.13E-10
CS	-3.00E-12	-3.00E-12	-3.00E-13	-3.00E-13
MS	5.50E-12	5.50E-12	5.50E-13	5.50E-13
RC	-6.00E-08	-1.33E-08	-1.32E-09	-2.93E-10
SI	-2.13E-08	-1.33E-08	-4.69E-10	-2.93E-10
Grand Total	**-1.33E-07**	**-4.00E-08**	**-2.93E-09**	**-8.77E-10**

Note

1. Systems are described in Table 3.1.
2. A positive number indicates and increase in risk while a negative number indicates a decrease in risk.

3.6.2 Defense-in-Depth

The intent of the inspections mandated by ASME Section XI for piping welds is to identify conditions such as flaws or indications that may be precursors to leaks or ruptures in a system's pressure boundary. Currently, the process for picking inspection locations is based upon structural discontinuity and stress analysis results. As depicted in ASME White Paper 92-01-01 Rev. 1, "Evaluation of Inservice Inspection Requirements for Class 1, Category B-J Pressure Retaining Welds," this method has been ineffective in identifying leaks or failures. EPRI TR-112657 and Code Case N-578-1 provide a more robust selection process founded on actual service experience with nuclear plant piping failure data.

This process has two key independent ingredients, that is, a determination of each location's susceptibility to degradation and secondly, an independent assessment of the consequence of the piping failure. These two ingredients assure defense-in-depth is maintained. First, by evaluating a location's susceptibility to degradation, the likelihood of finding flaws or indications that may be precursors to leaks or ruptures is increased. Secondly, the consequence assessment effort has a single failure criterion. As such, no matter how unlikely a failure scenario is, it is ranked High in the consequence assessment, and at worst Medium in the risk assessment (i.e., Risk Category 4), if as a result of the failure there is no mitigative equipment available to respond to the event. In addition, the consequence assessment takes into account equipment reliability, and less credit is given to less reliable equipment.

All locations within the Class 1 and 2 pressure boundaries will continue to receive a system pressure test and visual VT-2 examination as currently required by the Code regardless of its risk classification.

4. IMPLEMENTATION AND MONITORING PROGRAM

Upon approval of the RI-ISI program, procedures that comply with the guidelines described in EPRI TR-112657 will be prepared to implement and monitor the program. The new program will be implemented for the fourth inservice inspection interval. No changes to the Updated Safety Analysis Report are necessary for program implementation.

The applicable aspects of the ASME Code not affected by this change would be retained, such as inspection methods, acceptance guidelines, pressure testing, corrective measures, documentation requirements, and quality control requirements. Existing ASME Section XI program implementing procedures will be retained and modified to address the RI-ISI process, as appropriate.

The monitoring and corrective action program will contain the following elements:

A. Identify
B. Characterize
C. (1) Evaluate, determine the cause and extent of the condition identified
   (2) Evaluate, develop a corrective action plan or plans
D. Decide
E. Implement
F. Monitor
G. Trend

The RI-ISI program is a living program requiring feedback of new relevant information to ensure the appropriate identification of high safety significant piping locations. As a minimum, risk ranking of piping segments will be reviewed and adjusted on an ASME period basis. In addition, significant changes may require more frequent adjustment as directed by NRC Bulletin or Generic Letter requirements, or by industry and plant specific feedback.

5. PROPOSED ISI PROGRAM PLAN CHANGE

A comparison between the RI-ISI program and ASME Section XI Code program requirements for in-scope piping is provided in Tables 5-1 and 5-2. Table 5-1 provides a summary comparison by risk region. Table 5-2 provides the same comparison information, but in a more detailed manner by risk category, similar to the format used in Table 3.6.

St. Lucie 2 is implementing the RI-ISI program before completion of the first period of its fourth inspection interval. As such, 100% of the required RI-ISI program inspections will be completed in the fourth interval. Examinations shall be performed during the interval such that the period examination percentage requirements of ASME Section XI, paragraphs IWB-2411 and IWC-2411 are met.

6. REFERENCES/DOCUMENTATION

EPRI TR-112657, "Revised Risk-Informed Inservice Inspection Evaluation Procedure, Rev. B-A"

ASME Boiler and Pressure Vessel Code Section XI, 2007 Edition with 2008 Addenda

ASME Code Case N-578-1, "Risk-Informed Requirements for Class 1, 2, and 3 Piping, Method B, Section XI, Division 1"

Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis"

Regulatory Guide 1.178, "An Approach for Plant-Specific Risk-Informed Decisionmaking Inservice Inspection of Piping"

ASME Section XI, Nonmandatory Appendix R, "Risk-Informed Inspection Requirements for Piping"

Supporting Onsite Documentation

Calculation/File No. PSL-BFJR-14-012, " PRA Evaluation for Risk-Informed ISI Application for St. Lucie Units 1 & 2", Revision 0

Calculation/File No.FPL-009-C02, "Risk-Informed Inservice Inspection Degradation Mechanism Evaluation for St. Lucie Nuclear Power Plant Unit 2 ", Revision 0

Calculation/File No. FPL-009-C03, "St. Lucie 2 Service History and Susceptibility Review", Revision 0

Calculation/File No. FPL-009-C04, "Risk Ranking Summary, Matrix and Report for St. Lucie Nuclear Power Plant, Unit 2", Revision 0

Calculation/File No. FPL-009-C05, "Risk Impact Analysis for St. Lucie Nuclear Power Plant, Unit 2", Revision 0

Record of Conversation No. ROC-003, "Minutes of the Element Selection Meeting for the Risk-Informed ISI Project for the St. Lucie Nuclear Power Plant, Unit 2", Revision 0

Table 3.1
St. Lucie 2 - System Selection and Consequence Segment / Element Definition

System Description	ASME Code Class	Number of Consequence Segments	Number of Elements
RC – Reactor Coolant System	Class 1	44	290
CH – Chemical and Volume Control System	Class 1	9	121
SI – Safety Injection System	Class 1 and 2	40	1337
BF – Main Feedwater and Auxiliary Feedwater Systems	Class 2	9	96
CC – Component Cooling Water System	Class 2	4	261
MS – Main Steam System	Class 2	6	107
CS – Charging System	Class 2	7	124
Totals		**119**	**2336**

Table 3.3
St. Lucie 2 - Failure Potential Assessment Summary

System[1]	Thermal Fatigue		Stress Corrosion Cracking				Localized Corrosion			Flow Sensitive	
	TASCS	TT	IGSCC	TGSCC	ECSCC	PWSCC	MIC	PIT	CC	E-C	FAC
RC	X	X				X					
CH		X									
SI		X	X		X						
BF	X	X									X
CC											
MS											X
CS											

Note
1. Systems are described in Table 3.1.

Table 3.4
St. Lucie 2 - Number of Elements by Risk Category With and Without Impact of FAC

System[1]	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without	With	Without
RC	0	0	52	52	0	0	238	238	0	0	0	0	0	0
CH	0	0	25	25	0	0	3	3	0	0	66	66	27	27
SI	0	0	10	10	0	0	309	309	0	0	306	306	712	712
BF	34	0	0	14	56	0	0	20	6	26	0	30	0	6
CC	0	0	0	0	0	0	0	0	0	0	261	261	0	0
MS	91	0	0	0	0	0	0	91	16	0	0	0	0	16
CS	0	0	0	0	0	0	53	53	0	0	59	59	12	12
Total	**125**	**0**	**87**	**101**	**56**	**0**	**603**	**714**	**22**	**26**	**692**	**722**	**751**	**773**

Notes

1. Systems are described in Table 3.1.

Table 3.5
St. Lucie 2 - Number of Elements Selected for Inspection by Risk Category Excluding Impact of FAC

System[1]	High Risk Region						Medium Risk Region				Low Risk Region			
	Category 1		Category 2		Category 3		Category 4		Category 5		Category 6		Category 7	
	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected	Total	Selected
RC	0	0	52	13	0	0	238	24	0	0	0	0	0	0
CH	0	0	25	7	0	0	3	1	0	0	66	0	27	0
SI	0	0	10	4	0	0	309	31	0	0	306	0	712	0
BF	0	0	14	4	0	0	20	2	26	3	30	0	6	0
CC	0	0	0	0	0	0	0	0	0	0	261	0	0	0
MS	0	0	0	0	0	0	91	10	0	0	0	0	16	0
CS	0	0	0	0	0	0	53	6	0	0	59	0	12	0
Total	**0**	**0**	**101**	**28**	**0**	**0**	**714**	**74**	**26**	**3**	**722**	**0**	**773**	**0**

Notes

1. Systems are described in Table 3.1.

Table 3.6
St. Lucie 2 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspection Locations			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	Section XI[2][3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
BF	2 (1)	High (High)	TASCS (FAC)	Medium (High)	7	4	-3	-1.00E-08	1.00E-08	-2.20E-10	2.20E-10
BF	2 (1)	High (High)	TT (FAC)	Medium (High)	0	0	0	no change	no change	no change	no change
BF	4 (1)	Medium (High)	None (FAC)	Low (High)	4	2	-2	1.00E-12	1.00E-12	1.00E-13	1.00E-13
BF	5a (3)	Medium (High)	TASCS, TT (FAC)	Medium (High)	3	2	-1	-1.80E-11	1.00E-11	-1.80E-12	1.00E-12
BF	5a (3)	Medium (High)	TT (FAC)	Medium (High)	1	1	0	-1.20E-11	0.00E+00	-1.20E-12	0.00E+00
BF	6a (3)	Low (High)	None (FAC)	Low (High)	1	0	-1	negligible	negligible	negligible	negligible
BF	7a (5b)	Low (Medium)	None (FAC)	Low (High)	2	0	-2	negligible	negligible	negligible	negligible
BF Total								**-1.00E-08**	**1.00E-08**	**-2.23E-10**	**2.21E-10**
CC	6a	Low	None	Low	0	0	0	no change	no change	no change	no change
CC Total								**0.00E+00**	**0.00E+00**	**0.00E+00**	**0.00E+00**
CH	2	High	TT	Medium	0	7	7	-4.20E-08	-2.33E-08	-9.24E-10	-5.13E-10
CH	4	Medium	None	Low	0	1	1	-5.00E-13	-5.00E-13	-5.00E-14	-5.00E-14
CH	6b	Low	TT	Medium	0	0	0	no change	no change	no change	no change
CH	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
CH Total								**-4.20E-08**	**-2.33E-08**	**-9.24E-10**	**-5.13E-10**
CS	4	Medium	None	Low	0	6	6	-3.00E-12	-3.00E-12	-3.00E-13	-3.00E-13
CS	6a	Low	None	Low	0	0	0	no change	no change	no change	no change
CS	7a	Low	None	Low	0	0	0	no change	no change	no change	no change
CS Total								**-3.00E-12**	**-3.00E-12**	**-3.00E-13**	**-3.00E-13**
MS	4 (1)	Medium (High)	None (FAC)	Low (High)	21	10	-11	5.50E-12	5.50E-12	5.50E-13	5.50E-13
MS	7a (5b)	Low (Medium)	None (FAC)	Low (High)	3	0	-3	negligible	negligible	negligible	negligible

Table 3.6
St. Lucie 2 - Risk Impact Analysis Results

System[1]	Category	Consequence Rank	Failure Potential		Inspection Locations			CDF Impact[4]		LERF Impact[4]	
			DMs	Rank	Section XI[2][3]	RI-ISI	Delta	w/ POD	w/o POD	w/ POD	w/o POD
MS Total								**5.50E-12**	**5.50E-12**	**5.50E-13**	**5.50E-13**
RC	2	High	TASCS	Medium	0	1	1	-6.00E-09	-3.33E-09	-1.32E-10	-7.33E-11
RC	2	High	TASCS, TT	Medium	9	9	0	-3.60E-08	0.00E+00	-7.92E-10	0.00E+00
RC	2 (2)	High	TASCS, TT (PWSCC)	Medium	0	0	0	no change	no change	no change	no change
RC	2	High	TT	Medium	0	3	3	-1.80E-08	-1.00E-08	-3.96E-10	-2.20E-10
RC	2 (2)	High	TT (PWSCC)	Medium	0	0	0	no change	no change	no change	no change
RC	4	Medium	None	Low	38	22	-16	7.00E-12	7.00E-12	7.00E-13	7.00E-13
RC	4 (2)	Medium	None (PWSCC)	Low	1	2	1	5.00E-13	5.00E-13	5.00E-14	5.00E-14
RC Total								**-6.00E-08**	**-1.33E-08**	**-1.32E-09**	**-2.93E-10**
SI	2	High	IGSCC	Medium	0	1	1	-3.33E-09	-3.33E-09	-7.33E-11	-7.33E-11
SI	2	High	TT	Medium	0	3	3	-1.80E-08	-1.00E-08	-3.96E-10	-2.20E-10
SI	2	High	TT, IGSCC	Medium	0	0	0	no change	no change	no change	no change
SI	4	Medium	None	Low	44	31	-13	6.50E-12	6.50E-12	6.50E-13	6.50E-13
SI	6a	Low	None	Low	20	0	-20	negligible	negligible	negligible	negligible
SI	6b	Low	ECSCC	Medium	1	0	-1	negligible	negligible	negligible	negligible
SI	6b	Low	IGSCC	Medium	0	0	0	no change	no change	no change	no change
SI	6b	Low	TT, IGSCC	Medium	0	0	0	no change	no change	no change	no change
SI	7a	Low	None	Low	44	0	-44	negligible	negligible	negligible	negligible
SI Total								**-2.13E-08**	**-1.33E-08**	**-4.69E-10**	**-2.93E-10**
Grand Total								**-1.33E-07**	**-4.00E-08**	**-2.93E-09**	**-8.77E-10**

Notes

1. Systems are described in Table 3.1-1.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1998 Edition through 2000 Addenda of ASME Section XI. Only previous Section XI examinations can be utilized in the risk impact analysis. Therefore, examinations performed in the third interval per the previous RI-ISI Program cannot be considered.

Since the RI-ISI Program for Class 1 welds was implemented during the entire third Interval, piping weld examinations performed during the second interval per the 1989 Edition of ASME Code Section XI were used for comparison purposes for Class 1 welds.

3. Only those ASME Section XI Code inspection locations that received a volumetric examination in addition to a surface examination are included in the count. Inspection locations previously subjected to a surface examination only were not considered in accordance with Section 3.7.1 of EPRI TR-112657.
4. Per Section 3.7.1 of EPRI TR-112657, the contribution of low risk categories 6 and 7 need not be considered in assessing the change in risk. Hence, the word "negligible" is given in these cases in lieu of values for CDF and LERF Impact. For those cases in high, medium or low risk region piping where the change in risk calculation produces a value of zero for CDF or LERF Impact, "no change" is listed.

Table 5-1
St. Lucie 2 - Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Region

System[1]	Code Category	High Risk Region					Medium Risk Region					Low Risk Region				
		Weld Count	Section XI[2]		EPRI TR-112657		Weld Count	Section XI[2]		EPRI TR-112657		Weld Count	Section XI[2]		EPRI TR-112657	
			Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[3]		Vol/Sur	Sur Only	RI-ISI	Other[43
RC	B-F	1	0	0	0	0	4	0	3	0	0	0	0	0	0	0
RC	B-J	51	9	22	13	0	234	39	26	24	0	0	0	0	0	0
CH	B-J	25	0	20	7	0	3	0	1	1	0	93	0	38	0	0
SI	B-J	10	0	1	4	0	2	0	0	0	0	139	13	1	0	0
SI	C-F-1	0	0	0	0	0	307	44	0	31	0	359	52	2	0	0
SI	C-F-3 [4]	0	0	0	0	0	0	0	0	0	0	520	0	0	0	0
BF	C-F-2	14	7	0	4	0	46	8	0	5	0	36	3	0	0	0
CC	C-F-2	0	0	0	0	0	0	0	0	0	0	261	0	0	0	0
MS	C-F-2	0	0	0	0	0	91	21	4	10	0	16	3	0	0	0
CS	C-F-3 [4]	0	0	0	0	0	53	0	0	6	0	71	0	0	0	0
Total	B-F	1	0	0	0	0	4	0	3	0	0	0	0	0	0	0
	B-J	86	9	43	24	0	239	39	27	25	0	232	13	39	0	0
	C-F-1	0	0	0	0	0	307	44	0	31	0	359	52	2	0	0
	C-F-2	14	7	0	4	0	137	29	4	15	0	313	6	0	0	0
	C-F-3 [4]	0	0	0	0	0	53	0	0	6	0	71	0	0	0	0

Notes

1. Systems are described in Table 3.1.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1998 Edition through 2000 Addenda of ASME Section XI. Only previous Section XI examinations can be utilized in the risk impact analysis. Therefore, examinations performed in the third interval per the previous RI-ISI Program cannot be considered. Since the RI-ISI Program for Class 1 welds was implemented during the entire third Interval, piping weld examinations performed during the second interval per the 1989 Edition of ASME Code Section XI were used for comparison purposes for Class 1 welds.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. St. Lucie 2 RI-ISI application did not rely on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. Code Category C-F-3 consists of Code Examination Category C-F-1 welds that were previously excluded from examination per Table IWC-2500-1 due to being welds in "thin wall piping". For the RI-ISI application, this exclusion does not exist.

Table 5-2
St. Lucie 2 - Inspection Location Selection Comparison
Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Conseq Rank	Failure Potential		Code Category	Weld Count	Section XI Inspection Locations[2]		TR-112657 Selections	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
BF	2 (1)	High (High)	High	TASCS (FAC)	Medium (High)	C-F-2	10	7	0	4	
BF	2 (1)	High (High)	High	TT (FAC)	Medium (High)	C-F-2	4	0	0	0	
BF	4 (1)	Medium (High)	High	None (FAC)	Low (High)	C-F-2	20	4	0	2	
BF	5a(3)	Medium (High)	Medium	TASCS, TT (FAC)	Medium (High)	C-F-2	16	3	0	2	
BF	5a(3)	Medium (High)	Medium	TT (FAC)	Medium (High)	C-F-2	10	1	0	1	
BF	6a(3)	Low (High)	Medium	None (FAC)	Low (High)	C-F-2	30	1	0	0	
BF	7a (5b)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	6	2	0	0	
CC	6a	Low	Medium	None	Low	C-F-2	261	0	0	0	
CH	2	High	High	TT	Medium	B-J	25	0	20	7	
CH	4	Medium	High	None	Low	B-J	3	0	1	1	
CH	6b	Low	Low	TT	Medium	B-J	66	0	14	0	
CH	7a	Low	Low	None	Low	B-J	27	0	24	0	
CS	4	Medium	High	None	Low	C-F-3 [4]	53	0	0	6	
CS	6a	Low	Medium	None	Low	C-F-3 [4]	59	0	0	0	
CS	7a	Low	Low	None	Low	C-F-3 [4]	12	0	0	0	
MS	4 (1)	Medium (High)	High	None (FAC)	Low (High)	C-F-2	91	21	4	10	
MS	7a (5b)	Low (Medium)	Low	None (FAC)	Low (High)	C-F-2	16	3	0	0	
RC	2	High	High	TASCS	Medium	B-J	2	0	0	1	
RC	2	High	High	TASCS, TT	Medium	B-J	13	9	0	9	
RC	2 (2)	High (High)	High	TASCS, TT (PWSCC)	Medium (Medium)	B-F	1	0	0	0	
						B-J	1	0	0	0	
RC	2	High	High	TT	Medium	B-J	33	0	22	3	
RC	2 (2)	High (High)	High	TT (PWSCC)	Medium (Medium)	B-J	2	0	0	0	
RC	4	Medium	High	None	Low	B-F	3	0	3	0	
						B-J	221	38	26	24	
RC	4 (2)	Medium (High)	High	None (PWSCC)	Low (Medium)	B-F	1	0	0	0	

Table 5-2
St. Lucie 2 - Inspection Location Selection Comparison Between ASME Section XI Code and EPRI TR-112657 by Risk Category

System[1]	Risk		Conseq Rank	Failure Potential		Code Category	Weld Count	Section XI Inspection Locations[2]		TR-112657 Selections	
	Category	Rank		DMs	Rank			Vol/Sur	Sur Only	RI-ISI	Other[3]
						B-J	13	1	0	0	
SI	2	High	High	IGSCC	Medium	B-J	1	0	1	1	
SI	2	High	High	TT	Medium	B-J	8	0	0	3	
SI	2	High	High	TT, IGSCC	Medium	B-J	1	0	0	0	
SI	4	Medium	High	None	Low	B-J	2	0	0	0	
						C-F-1	307	44	0	31	
SI	6a	Low	Medium	None	Low	C-F-1	125	20	2	0	
						C-F-3 [4]	156	0	0	0	
SI	6b	Low	Low	ECSCC	Medium	C-F-1	4	1	0	0	
SI	6b	Low	Low	IGSCC	Medium	B-J	16	0	0	0	
SI	6b	Low	Low	TT, IGSCC	Medium	B-J	5	0	1	0	
SI	7a	Low	Low	None	Low	B-J	118	13	0	0	
						C-F-1	230	31	0	0	
						C-F-3 [4]	364	0	0	0	

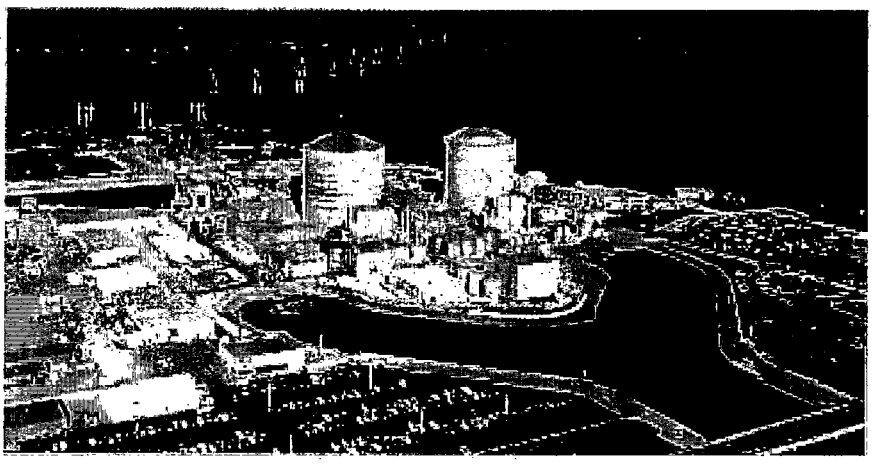
Notes

1. Systems are described in Table 3.1-1.
2. Since no examination selections had been made for the fourth interval ISI Program prior to the development of the RI-ISI Program, the third interval selections were used for comparison purposes. The Code of record for the third interval was the 1998 Edition through 2000 Addenda of ASME Section XI. Only previous Section XI examinations can be utilized in the risk impact analysis. Therefore, examinations performed in the third interval per the previous RI-ISI Program cannot be considered. Since the RI-ISI Program for Class 1 welds was implemented during the entire third Interval, piping weld examinations performed during the second interval per the 1989 Edition of ASME Code Section XI were used for comparison purposes for Class 1 welds.
3. The column labeled "Other" is generally used to identify augmented inspection program locations credited per Section 3.6.5 of EPRI TR-112657. The EPRI methodology allows augmented inspection program locations to be credited if the inspection locations selected strictly for RI-ISI purposes produce less than a 10% sampling of the overall Class 1 weld population. St. Lucie 2 RI-ISI application did not rely on augmented inspection program locations beyond those selected by the RI-ISI process. The "Other" column has been retained in this table solely for uniformity purposes with the other RI-ISI application template submittals.
4. Code Category C-F-3 consists of Code Examination Category C-F-1 welds that were previously excluded from examination per Table IWC-2500-1 due to being welds in "thin wall piping". For the RI-ISI application, this exclusion does not exist.

ATTACHMENT A

PRA TECHNICAL ADEQUACY IN SUPPORT OF RISK-INFORMED IN-SERVICE INSPECTION FOR ST. LUCIE 1 & 2

PRA Technical Adequacy in Support of Risk-Informed In-Service Inspection for St. Lucie Units 1 & 2



FPL

Probabilistic Risk Assessment Group
PSL-BFJR-14-028
Revision 0
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1.0 INTRODUCTION

This evaluation summarizes the assessment of the St. Lucie (PSL) Probabilistic Risk Assessment (PRA) capability as measured against the current ASME/ANS PRA Standard (Reference 1) as endorsed by the U.S. NRC Regulatory Guide (RG) 1.200 Revision 2 (Reference 2). This assessment addresses the technical adequacy of the St. Lucie PRA for use in risk-informed applications. Specifically, this document assesses the technical adequacy of the St. Lucie PRA with respect to the Risk-Informed In-Service Inspection (RI-ISI) program.

The assessment is based on a series of formal peer reviews, other external reviews, and internal self-assessments documented and summarized in this report. This assessment uses the latest PSL PRA baseline model update (Reference 3) and internal flooding analysis (Reference 4) which is evaluated using EPRI guidance for PRA technical adequacy for RI-ISI programs (Reference 5).

2.0 BACKGROUND

2.1 RG 1.200 and PRA Standard

The ASME/ANS PRA Standard (ASME/ANS RA-Sa-2009) has eight "parts" with technical elements, high level requirements (HLRs), and detailed supporting requirements (SRs). These parts represent the major classes of hazards included in a PRA:

- Part 2, internal events (addressed in Section 3.1),
- Part 3, internal flood (addressed in Section 3.1),
- Part 4, internal fire (addressed in Section 3.2),
- Part 5, seismic events (addressed in Section 3.2),
- Parts 6 to 9, other external hazard events (addressed in Section 3.2).

Note - Part 1 of the PRA Standard is introductory information and does not contain any requirements except configuration control (addressed in Section 3.3).

NRC Reg Guide 1.200, Revision 2 endorses this Standard with minor "clarifications."

The Standard supporting requirements allow the assessment of the portions of the PRA as Capability Category CC-I, CC-II, or CC-III, with increasing scope and level of detail, plant-specificity, and realism. Thus, the overall assessment of PRA capability is the collection of the assessments of the hundreds of supporting requirements. NRC requires PRA quality to meet Capability Category II, at minimum, for use in risk-informed applications.

2.2 St. Lucie PRA History and Quality

In response to NRC Generic Letter 88-20 (Reference 6), the St. Lucie PRA Level 1 and Level 2 models (collectively known as internal events analysis) were originally developed and submitted in 1993. The document called St. Lucie Individual Plant Examination (IPE) (Reference 7). The PRA models addressed risk assessment of at-power operation. Later, in 1996, the IPE was supplemented with the Individual Plant Examination for External Events (IPEEE) document submittal (Reference 8) to address external events such as internal fire and flood, among other events. This PRA was then subjected to a number of reviews, internal and external, during its preparation as well as extensive reviews by NRC through national labs following its publication.

In July 2002, the Combustion Engineering Owners Group (CEOG) performed a peer review of the St. Lucie Units 1 and 2 Level 1 and Level 2 2002-PSA models update. The review followed a process that was adopted by industry reference NEI 00-02, Rev. A3

(Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, Nuclear Energy Institute, March 2000, Reference 9). The resulting Findings and Observations (F&Os) of the CEOG peer review were published in the February 2003 Westinghouse publication WCAP-16034, Rev 0, St. Lucie Unit 1 and 2: Probabilistic Risk Assessment Peer Review Report, CEOG Task 1037 (Reference 10). The assessment covered all aspects of the PRA model and documentation. The result of the assessment ranked the findings on a scale of "A" to "D", with "A" being the most significant. Each of the findings was presented with observations and comments.

In December 2005, MARACOR Software & Engineering, Inc. performed an independent review of St. Lucie Unit 1 & 2 PSA models update. The review process was based on conformance to Category II of the ASME PRA Standard (ASME-RA-S-2002), Addenda A (RA-Sa-2003). The resulting recommendations were published in the MARACOR report, "An Independent Review of the Port St. Lucie PRA against the Requirements of the ASME PRA Standard" dated December 13, 2005 (Reference 11).

Following the issuance of the ASME PRA Standard and Regulatory Guide 1.200 (RG 1.200), in October 2007, St. Lucie PRA Group performed a self-assessment of the 2002 peer review and 2005 assessment to identify gap and actions needed to conform to the requirements delineated by RG-1.200 Rev 1. The current PSL gap analysis uses the RA-Sa-2009 version of the standard as endorsed by RG 1.200, Revision 2.

To supplement the original peer review and internal gap analysis, and to further improve the quality of the updated internal events models used in the Fire PRA, subsequent focused scope peer reviews for St. Lucie were conducted. A LERF Focused Peer Review (Reference 12) was conducted in July 2009, which identified no findings.

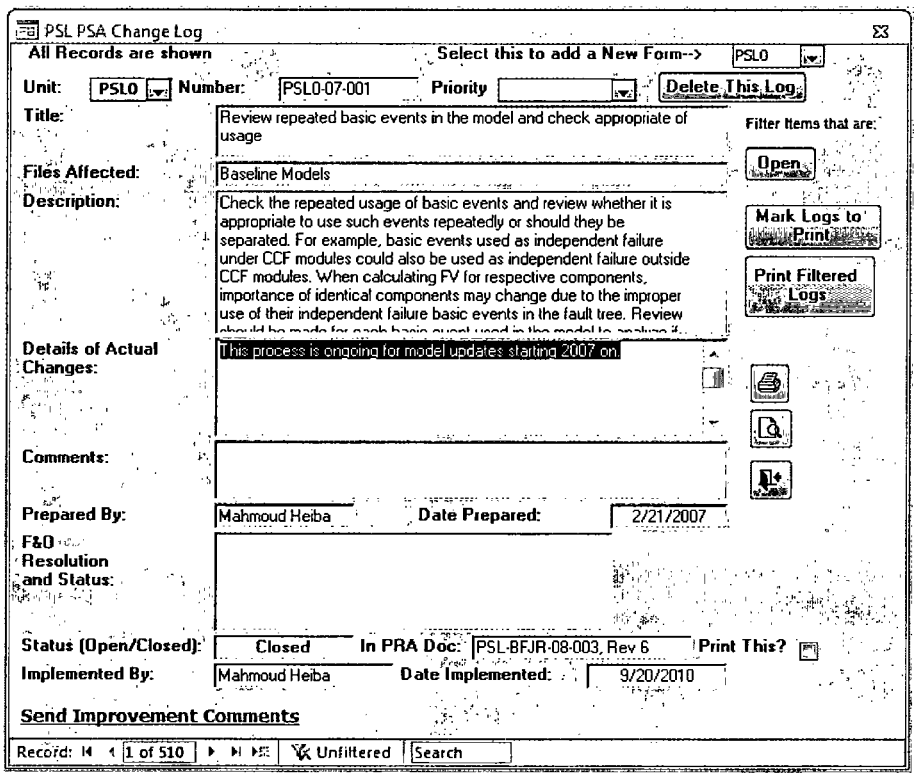
A focused peer review of Common-Cause Failure (CCF) methodology and data (Reference 13) was completed in August 2009. In April 2011, a focused peer review (Reference 14) was performed by the PWROG that included Human Reliability Analysis (HRA), Internal Flooding Analysis (IF), and Data Analysis (DA) for compliance against the most current combined PRA standard, ASME/ANS RA-Sa–2009, as endorsed by Regulatory Guide 1.200, Rev. 2. The internal flooding analysis focused peer review was performed because the latest internal flooding analysis was much more comprehensive than the original screening analysis that was performed for the IPEEE. Although the basic methods used for the HRA had not changed substantially, the HRA focused peer review was performed because of the enhanced HRA dependency analysis and the use of the HRA Calculator software in the latest model, and the fact that HRA plays a significant role in the determination of the dominant sequences and overall risk profile.

In October 2013, a focused peer review was conducted for Interfacing System LOCA (ISLOCA) initiating events (Reference 15) in accordance with RG 1.200 Revision 2. The ISLOCA analysis utilized a completely different approach from the previous analysis.

Significant findings from the peer reviews are listed in Attachment A, along with their resolutions.

2.3 Model Change Database

The living PRA is maintained through use of the plant's Model Change Database (MCDB) in accordance with the PRA Configuration Control and Model Maintenance Procedure (Reference 16). A sample screen shot of the input form is shown below.



This database is used to store the details of all modifications, proposed and actual, and open or closed, for the St. Lucie PRA model. This includes findings and observations from peer reviews, self-assessments, and issues identified during use and update of the PRA model. There are currently 519 records in the database, of which 66 are open. The open items are all model enhancements or documentation issues, and have been judged not to significantly impact PRA model applications. Open items will be addressed in future PRA updates, based on the significance of the open item and the scope of the update.

2.4 St. Lucie PRA Capability Target

The target capability level for the St. Lucie PRA is Capability Category II (CC-II). That is, the goal is to meet all supporting requirements (SRs) at least at the CC-II level. This is the minimum capability level needed by any foreseeable application.

Note that in many supporting requirements, the requirement spans all three capability categories. Thus, if the SR is met, it meets CC III. While CC II is the target, CC III is met in many SRs.

2.5 Assessment Process

In the risk-informed in-service inspection (RI-ISI) program at PSL, the EPRI RI-ISI methodology (Reference 17) is used to define alternative in-service inspection requirements. Plant-specific PRA-derived risk significance information is used during the RI-ISI plan development to support the consequence assessment, risk ranking and delta risk evaluation steps.

The importance of PRA consequence results, and therefore the necessary scope of PRA technical capability, is tempered by two processes in the EPRI methodology. First, PRA consequence results are binned into one of three conditional core damage probability (CCDP) and conditional large early release probability (CLERP) ranges before any welds are chosen for RI-ISI inspection. The risk importance of a weld is therefore not tied directly to a specific PRA result. Instead, it depends only on the range in which the PRA result falls. The wide binning provided in the methodology generally reduces the significance of specific PRA results.

Secondly, the influence of specific PRA consequence results is further reduced by the joint consideration of the weld failure potential via a non-PRA-dependent damage mechanism assessment. The results of the consequence assessment and the damage mechanism assessment are combined to determine the risk ranking of each pipe segment (and ultimately each element) according to the EPRI Risk Matrix.

Based on EPRI TR-1021467 a specific list of capability category requirements (Appendix A) has been developed for the RI-ISI program defining which of the ASME PRA Standard supporting requirements should fall under categories I, II, or III.

The assessment of PRA capability judges the St. Lucie PRA against each supporting requirement in the PRA Standard as "Meets" CC-I, CC-II, or CC-III. If the PRA does not meet the requirements of any category for a specific SR, it is assessed as "Not Met." This assessment is captured in a Microsoft Access database. There is a table in this database with the SR-by-SR assessments from industry peer reviews and internal self-assessments, interlinked with tables that included Facts and Observations (F&Os) from the PWROG peer review and the focused peer reviews, along with their status and resolutions.

3.0 CONFORMANCE WITH PRA ASME STANDARD

The following sections describe the conformance and capability of the St. Lucie PRA against the major parts of PRA ASME Standard.

3.1 ASME Parts 2 - Internal Events

The internal events portion of the St. Lucie PRA has been updated a number of times since the original IPE submittal.

As described in Section 2.2, there have been one global peer review (full scope) and several focused peer reviews to include various ASME elements such as HR, DA, and LE as well as other PRA areas with cross-connection among ASME elements (e.g., CCF, and ISLOCA). The following peer reviews have been conducted against internal events supporting requirements:

- In 2002, a review of all technical elements was performed by CEOG using the industry PSA Certification process, the precursor to the PRA Standard. All of the findings and observations have been addressed in the model updates following this peer review.

- In 2005, a self-assessment was performed by MARACOR against ASME-RA-Sa-2003.

- In 2009, a focused peer review on Large Early Release Frequency (LERF) was performed (HLR-LE). This review was conducted by PWROG using ASME RA-Sb-2005 as endorsed by RG 1.200 Revision 1 and resulted in zero findings and observations.

- In 2011, a focused peer review was performed by PWROG for the elements DA, and HR. This assessment replaced the 2002 peer review for those elements that were in scope. This review was done using the current PRA Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2. All of the findings and suggestions have been resolved, and, where changes were necessary, addressed in a model update

In addition to these peer reviews; there have been 2 subsequent focused peer reviews for specific PRA areas associated with the St. Lucie PRA models; mainly common-cause failure (CCF) methodology and Interfacing Systems LOCA (ISLOCA) modeling. Each of these PRA areas included review of applicable cross-cut of multiple ASME elements.

- In 2009, a focused peer review of CCF methodology and respective data was performed. This review covered all SRs in the ASME Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2 which have a relationship to CCF. All of the findings and observations have been addressed in the model updates following this peer review.

- In 2013, a focused peer review of ISLOCA methodology and respective data was performed. This review covered all SRs in the ASME Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2 which have a relationship to ISLOCA. The respective findings and observations are currently being reviewed and responses are being provided. Each finding will be addressed in a model update (if applicable) and documented accordingly.

Conclusion

The current open items do not represent significant deficiency in the analyses necessary to support the RI-ISI application. The current St. Lucie PRA meets all Part 2 (internal event) CC II requirements of the PRA Standard.

3.2 ASME Part 3 - Internal Flooding

In 2011, a focused peer review was performed by PWROG for Internal Flooding element (IF). This review was done using the current PRA Standard (ASME/ANS RA-Sa-2009) as endorsed by RG 1.200 Revision 2. All of the findings and suggestions have been resolved, and, where changes were necessary, addressed in a model update

Conclusion

There are no current open items that represent significant deficiency in the analyses necessary to support the RI-ISI application. The current St. Lucie PRA meets all Part 2 (internal event) CC II requirements of the PRA Standard.

3.3 ASME Parts 4 through 9 – External Events

The NEI 04-10 methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. For those cases where the STI cannot be modeled in the plant PRA (or where a particular PRA model does not exist for a given hazard group), a qualitative or bounding analysis is performed to provide justification for the acceptability of the proposed test interval change.

3.3.1 ASME Part 4 - Internal Fire

A fire PRA was performed for St. Lucie as part of the 1994 IPEEE submittal. Since it was done for the IPEEE, it was more of a screening analysis to discover any fire vulnerabilities than an attempt to determine a realistic estimate of core damage risk due to fire. It has not been updated since the original submittal.

St. Lucie is an NFPA-805 plant, and therefore has a fire PRA to support the NFPA-805 effort. The fire PRA uses the latest internal events PRA model as a basis. The St. Lucie

NFPA-805 fire PRA uses NUREG/CR-6850 guidance as required by NFPA-805, and thus produces a conservative estimate of core damage risk due to fire.

A peer review of the St. Lucie (PSL) Fire PRA was performed at PSL using the NEI 07-12 Fire PRA peer review process, and the PRA standard (ASME/ANS RA-Sa–2009) as endorsed by RG 1.200, Revision 2. The purpose of this review was to provide a method for establishing the technical quality and adequacy of the Fire PRA for the spectrum of potential risk-informed plant licensing applications for which the Fire PRA may be used. The PSL Fire PRA Peer Review was a full-scope review of all the technical elements of Part 4 of the ASME/ANS standard.

The Fire PRA update addressed the Supporting-Requirement-assessed deficiencies (i.e., Not Met or CCI). Completion of recommendations related to Supporting Requirement assessments and 'Finding' F&Os results in a Capability Category II assessment for the majority of the Supporting Requirements.

Conclusion

Based on the completion of peer review recommendations and the assessment of deferred items, the St. Lucie Fire PRA is adequate to support this application, with the caveat that the PRA is a conservative representation of the fire risk from operation of St. Lucie Nuclear Plant. The Fire PRA model will be exercised to obtain quantitative fire risk insights, but refinements may need to be made on a case-by-case basis.

3.3.2 ASME Part 5 - Seismic Events

St. Lucie is sited in an area of very low seismicity. There is no seismic PRA or seismic margin analyses for St. Lucie. For the seismic portion of the St. Lucie IPEEE, FPL used the FPL site-specific seismic program associated with Unresolved Safety Issue (USI) A-46 (Generic Letter 87-02, "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors"). This primarily consisted of extensive walkdowns of the St. Lucie site looking for seismic vulnerabilities.

Staff at St. Lucie recently performed additional seismic walkdowns in response to Near-Term Task Force Recommendation 2.3, the NRC issued a 10CFR50.54 letter in March 2012 requesting that all licensees perform seismic walkdowns to identify and address plant degraded, non-conforming, or unanalyzed conditions, with respect to the current seismic licensing basis. As per the EPRI guidance, two Seismic Walkdown Equipment Lists (SWELs) were generated. The first consisted of a variety of Seismic Category I components that support the safety functions of reactivity control, reactor coolant pressure and inventory control, decay heat removal, and containment function. The second consisted of a variety of Seismic Category I components whose failure could lead to a rapid drain-down of the spent fuel pool. Walkdowns were performed in order to verify proper anchorage of the applicable SWEL components and to identify any adverse seismic conditions. No operability concerns were identified. Only minor issues were found that required documentation updates, improved housekeeping, or small repairs due to corrosion or minor concrete cracking. These issues were, or are scheduled to be, addressed through the site's Corrective Action Program.

Conclusion

The fact that St. Lucie is in a region of very low seismicity; multiple seismic walkdowns have been performed to verify the seismic design of equipment important to safety. Seismic CDF is evaluated using the most updated site mean hazard curve developed by EPRI and published in 2014. LERF is conservatively considered as 10% of CDF value. The seismic risk is evaluated as 3.49E-06/year for CDF and 3.49E-07/year for LERF for both units. This supports the conclusion that seismic risk at St. Lucie is minimal and will not be a significant factor in the RI-ISI application.

3.3.3 ASME Parts 6 to 9 - Other External Hazards

The risk analyses of the other external hazards were performed and published in the St. Lucie IPE and the St. Lucie IPEEE by 1996 and have not been updated since. These analyses were typically bounding and screening in nature, and therefore are not well-suited for configuration-specific risk applications. Therefore, in performing the assessments for the other hazard groups, a qualitative or a bounding approach will be utilized in most cases.

3.3.4 Conclusion – External Events

As stated earlier, the NEI 04-10 methodology allows for STI change evaluations to be performed in the absence of quantifiable PRA models for all external hazards. Therefore, in performing the assessments for the other hazard groups, a qualitative or a bounding approach will be utilized in most cases. The Fire PRA model will be exercised to obtain quantitative fire risk insights but refinements may need to be made on a case-by-case basis. This approach is consistent with the accepted NEI 04-10 methodology.

3.4 PRA Model Maintenance and Control

PRA model maintenance and control requirements are described in the PRA Standard, Section 1-5. These requirements are addressed in the current set of Nextera/FPL fleet procedures that address model maintenance and control:

EN-AA-105, Probabilistic Risk Assessment Program

EN-AA-105-1000, PRA Configuration Control and Model Maintenance

EN-AA-105-10000, Control of PRA Documentation and Evaluations

4.0 CONCLUSION

The St. Lucie PRA model of record fully meets all the requirements of Part 2 (internal events) and Part 3 (internal flood) of the ASME/ANS PRA Standard as endorsed by RG 1.200 Revision 2. All findings (level "A" or "B", or specified) from peer reviews or other technical reviews have been (or are currently being) addressed or closed with no impact to the RI-ISI program.

Based on the completion of peer review recommendations and the assessment of deferred items, the St. Lucie Fire PRA is adequate to support applications. The Fire PRA model can be exercised to obtain quantitative fire risk insights, but refinements may need to be made on a case-by-case basis.

Seismic risk at St. Lucie is minimal and will not be a significant factor in the RI-ISI application.

Current F&Os that are considered "open" are listed in Attachment B. These findings are the result of the most recent focused peer review of the ISLOCA analysis. For ISLOCAs, the ISLOCA itself is a single-order cutset, i.e., the ISLOCA by itself is sufficient for core damage, so mitigating equipment is not a factor. As for the introduction of an initiating event, the quality of the ISLOCA modeling will not be a concern in the assessment of the weld failure's impact through the introduction of an initiating event.

Furthermore, a sensitivity study was completed for F&Os IE-C5-01, IE0C9-01, and SY-A2-01. The results of this study can be found in Attachment C to this document. The overall conclusion was that modeling changes to address these F&Os would lead to a decrease in overall risk. F&O IE-C6-01 is a documentation issue only. Therefore, F&Os related to the ISLOCA modeling will not have an impact on the RI-ISI analysis.

5.0 REFERENCES

1. ASME/ANS RA-Sa-2009, "Addenda to ASME/ANS RA-S-2008 Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications", American Society of Mechanical Engineers and American Nuclear Society, February 2009.

2. NRC Regulatory Guide 1.200, Revision 2, An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities, Office of Nuclear Regulatory Research, March 2009.

3. PSL-BFJR-12-001, Revision 0, St. Lucie EPU PSA Model Update for Units 1&2, Florida Power and Light Co., May 2012.

4. PSL-BFJR-11-005, Revision 0, Internal Flood Analysis for St. Lucie Units 1 & 2, Florida Power and Light Co., June 2013.

5. EPRI 1021467-A, Nondestructive Evaluation: Probabilistic Risk Assessment Technical Adequacy Guidance for Risk-Informed In-Service Inspection Programs, Electric Power Research Institute, June 2012.

6. Generic Letter No. 88-20, Individual Plant Examination for Severe Accident Vulnerabilities - 10 CFR 50.54(f), Nuclear Regulatory Commission, November 1988.

7. PSL-IPE, Revision 0, St. Lucie Units 1 & 2 Individual Plant Examination, Florida Power and Light Co., December 1993.

8. PSL-IPEEE, Revision 0, Individual Plant Examination of External Events for St. Lucie Units 1 and 2, Florida Power and Light Co., December 1994.

9. NEI 00-02, Revision A3, Probabilistic Risk Assessment (PRA) Peer Review Process Guidance, Nuclear Energy Institute, March 2000.

10. WCAP-16034, Revision 00, St. Lucie Units 1 and 2: Probabilistic Risk Assessment Peer Review Report, CEOG Task 1037, Westinghouse, February 2003.

11. An Independent Review of the Port St. Lucie PRA Against the Requirements of the ASME PRA Standard, Revision 0, Maracor Software & Engineering, Inc., December 2005.

12. LTR-RAM-II-09-038, Revision 0, Focused Scope Peer Review of the St. Lucie Units 1 and 2 Large Early Release Frequency PRA Against the ASME PRA Standard Requirements, Westinghouse, July 2009.

13. Focused Peer Review of St. Lucie PSA Common Cause Failure (CCF) Methodology, Revision 0, Michael Lloyd, August 2009.

14. LTR-RAM-II-11-054, Revision 0, RG 1.200 PRA Focused Peer Review Against the ASME PRA Standard Requirements for Plant St. Lucie (PSL) Probabilistic Risk Assessment, Westinghouse, July 2011.

15. RSC 13-49, Revision 0, St. Lucie Nuclear Power Plants Focused Scope PRA Peer Review: Interfacing Systems LOCA, Reliability and Safety Consulting Engineers, Inc., December 2013.

16. EN-AA-105-1000, Revision 2, PRA Configuration Control and Model Maintenance, Florida Power and Light Co., June 2014.

17. EPRI TR-112657, Revision B-A, Revised Risk-Informed Inservice Inspection Evaluation Procedure (PWRMRP-05), Electric Power Research Institute, December 1999.

18. PSL-BFJR-13-020, Revision 0, NRC RG-1.200 Rev. 2 Self-Assessment for St. Lucie Units 1 & 2, Florida Power and Light Co., March 2014

Attachment A - Peer Review Findings

Table A-1 summarizes facts and observations with significance ranking "A", "B", or "Finding" from the previously referenced peer reviews for Internal Events and Internal Flooding.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
AS-01	AS-A5 AS-A7 SY-A1	Cutset %ZZSU1*CMM1AVCCCF appears overly conservative. Each CCW header provides approximately 8000 gpm. The largest accident loads are the shutdown cooling heat exchangers (4500 gpm) and the fan coolers (1200 gpm each). The N-loads are the SPF HXs (2900 gpm), let down heat exchanger (less than 1400 gpm), the RCP cooling (250 gpm each), and the boric acid concentrators (775 gpm). During a small LOCA, the heat load on the containment fan coolers is significantly lower than a design basis accident. The load on the SDC HXs does not exist until re-circulation. Eventually, the LOCA will lead to the failure of the RCPs even if the operators do not trip the pumps. When the RCPs are not running the heat load is further reduced. The SPF will act to moderate temperature changes due to the large volume of water. Not only will the peak containment temperature and pressures will be much lower during the small break LOCA, but also the decay heat removal will not solely be provided through the break. Secondary side heat removal is quite effective during the small LOCA break sizes. This issues combine to form a reasonable basis for not requiring N-header isolation during a small break LOCA. Considering the initial flow rates through the TCCW HXs and the Open Blowdown HXs, it would be a more difficult argument to make to not require the closure of ICW MOVs 21-2 and 21-3. Considering that most of the heat removal can still be provided by the S/Gs, it is probably reasonable to removal this closure as well. This being said the failure both ICW isolation and N-header isolation should probably be considered failure unless a more detailed calculation is available.	A	Add basis for excluding N-header isolation following a small break LOCA.	Success Criteria for N-Header and ICW-to-TCW HXs isolation valves have been revised as well as CCF data analysis during previous model maintenance and update.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
AS-02	AS-A5 AS-A7 ASA10 AS-B1 SC-A6 QU-A1	Considering the significance of aligning OTCC and the high human action failure probability. It would be prudent to credit to develop multiple human action failure probabilities depending on the type of trip. Breaking out the trips based on S/G water level would be a good start (low, normally, and high).	A	Add three flavors of OTCC mitigation.	Multiple operator actions were added to the model to account for different available times to initiate OTC (based on SG level and trip and whether AFW operated for some time after trip). The current model included Human Failure Events (HFEs) to initiate OTC following normal or low level trips with short term loss of FW, loss of FW after operating for at least 4 hrs, and loss of FW after CST depletion.
AS-03	AS-A2 AS-A4 AS-A3 AS-A5 AS-A7 AS-A10 AS-B1 HR-E1 HR-F1 HR-F2 SC-A3 SC-A6 SY-A2 SY-A22 QU-A1	MFW is not credited post trip. This leads to quite a few high level cutsets that are overly conservative. If post S/G level control is automatic, then only the control system hardware need be modeled. If not, then the human action to control S/G water level need be modeled. The availability of the TBVs post quick open prevents the need for hot well make-up. Crediting the ADVs for use with MFW would require the modeling of hotwell make-up.	A	Credit MFW post trip following IEs where MFW is available.	MFW modeling was enhanced and credited post trip where applicable.
AS-04	AS-A5 AS-A7 AS-A10 AS-B1 SC-A6 QU-A1	RWT rupture is assumed to fail shutdown cooling. This seems overly conservative. Without make-up the level in the RCS would drop, but there is more than enough fluid in the Boric Acid Tanks and the VCT to restore this level. The level does not need to be fully restored to allow shutdown cooling. The level need only be above the hot leg. Estimated Level Drop 2250 psia at 600 F (0.0217 ft^3/lbm) to 100 psia at 300F (0.01766 ft^3/lbm). Given RCS liquid volume of 10,400 ft^3, this means approximately 18,500 gallons are required to restore the PRZ level. Each Boric Acid Tank contains 9700 gallons the VCT contains 4000 gallons. Fully PRZ level is not required full shutdown cooling when core damage is the alternative.	B	Require either (RWT) or (2 BASTs and a VCT) for shutdown cooling	This is a highly hypothetical scenario that would require Ops to violate EOPs when operating the plant toward SDC. Use of RWT and BAMTs is required when RCS needs makeup due to shrinkage following Rx trip. The reviewer assumes that Ops will continue to SDC, even in case if RWT rupture were to occur during makeup to the point where RCS cooling continues at a rate of 100F/hr, and RCS level drops down to about midloop level. EOP-02, step 4.5 prevent against this behavior by requiring that OP ensure RCS inventory control is maintained, and PZR level is restored between 30% to 35%. SDC will not continue until PZR level is restored. The concerned scenario is perceived as not credible.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
AS-06	AS-A3 AS-A4 AS-A5 AS-A7 AS-A10 AS-B1 HR-E1 HR-F1 HR-F2 SC-A3 SC-A6 SY-A2 SY-A22 QU-A1.	Consider adding low pressure feed (using Condensate pumps) to the model for accident sequences involving loss of all MFW/AFW. Using condensate pumps to feed the SG's is in both EOP 6 'Total Loss of Feed' and EOP-15 'Functional Recovery Procedure'. Operations is directed to use low pressure feed in 1-EOP-06 (Step 8.B.3.1). Crediting low-pressure feed will eliminate those core damage sequences where the MFW pumps are lost, but the condensate pumps are available. If the TBVs are not available, then the hot well make-up control system (or an operation action) must be modeled to incorporate this alternative. Adding LPF could reduce dependency on Once Through Cooling for a number of accident sequences. (See F&O AS-03 also)	B	Consider including Low Pressure Feed from the condensate pumps in accident sequences that include TLOF. (Current model remains conservative).	LOMFW IE events were combined in data update into a single IE and recovery events were developed based on the type of events that have occurred. Low pressure feed would be included if applicable based on IE data review.
AS-08	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6 SY-A1	Check Valves 09294 and 09252 are common for both AFW, MFW, and Low Pressure Feed. These CKVs currently appear only in the AFW system. The may be some events (e.g. LOL) where the turbine trips and steam generator pressure rises enough to cause the closure of these check valves. Under these scenarios, the failure of both of these checks would fail all secondary side heat removal. Currently, these CKVs are modeled under FMM1SGCVLV. This event has a failure probability far lower than several three element CKV groups in the AFW system. There does not appear to be a basis for this difference. The failure likelihoods (independent and common cause) of the check valves in the AFW system should be consistent or the basis for the difference is documented. Further, as the random failure of these CKVs could cause a LOFW trip and eliminate all secondary side feed to a single S/G, this is worthy of consideration as an initiating event.	A	Document basis for CKV failure rates and ensure CKVs appear in the MFW and low pressure feed portions of the tree.	The models were revised to ensure failure of the referenced check valves have the expected impact on loss of feedwater to applicable SG. Data Analysis was revised to use the latest industry as well as plant-specific data.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
AS-11	N/A	Documentation used to provide the basis of event tree structure is not adequately traceable to the underlying analysis.	B	Provide references in accident sequence analysis documents to the supporting thermal-hydraulic analyses. Describe how analyses are made applicable to the PSA, e.g., justify why licensing and design basis analyses, which make various non-PSA related assumptions and are often very conservative, can be used for defining accident sequences and timings.	The current revision of the stand-alone Accident Sequence Analysis has revised and corrected many editorial issues existed in its predecessor analysis document.
AS-12	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6	Currently, shutdown cooling is credited as a long-term cooling method to eliminate the re-circulation requirement on certain ranges of LOCA breaks. A certain amount of water must be above the bottom of the hot leg to avoid drawing vapor into the shutdown cooling system. Some calculation must be done to ensure that the RCS will be above this critical point. This calculation could be quite simple: determine the RCS water level at the point of shutdown cooling entry conditions, determine the leakage rate at the point, verify the RCS level will be adequate for the remaining part of the 24 hr mission without re-circulation or RCS make-up. If this is not true, then addition make-up must be modeled through the emergency sump or CVCS.	B	Do simple calculation. Take appropriate action.	To enter SDC during the course of mitigating a LOCA, an RCS level of 30% in the Pressurizer is required by EOP-3. Further, once SDC has been entered ONP-01-03 requires OTC to be re-established if RCS level falls below 29 feet 9.5 inches (Top of Hot Leg). Operation of SDC in the above referenced condition is procedurally not allowed and physically not possible. Therefore the question is highly hypothetical and not applicable at PSL.
AS-13	AS-A2 AS-A3 AS-A5 AS-A7 AS-A10 AS-B1 HR-C1 HR-C2 HR-C3 QU-A1 QU-B10 SC-A3 SC-A6 SY-A1 SY-A7 SY-A8 SY-A9 SY-A11 SY-A12 SY-C1 SY-C2	The PORVs are only assumed to lift given total loss of secondary side heat removal or a loss of load with no anticipatory trip. This appears non-conservative. The only loss of load trips considered are TT and loss of off-site power trips. This is based on an informal calculation that shows the RCS pressure exceeds 2300 psia, but stays below the PORV open set point of 2400 psia. This does not consider variations in the time delay between the turbine trip and the reactor trip nor does it consider variations in the PRZ pressure set point. Consideration of these variables may lead the analyst to conclude that the likelihood of a PORV lift during this condition is much larger than analyzed. Further, the portion of the tree (under Gate U1QT99) that models the circuitry associated with the anticipatory trip only contains a single basic event. No other support system dependencies appear. For example, does the status of pressurizer spray affect this calculation? Are there support system failures that could cause a loss of load and disable or degrade the anticipatory trip function?	B	Model support systems for anticipatory trip. Consider a likelihood that the PORVs open on LOL with anticipatory trip. Consider a higher PORV challenge likelihood during other types of trip where the delay between the turbine and reactor trip is larger (e.g. spurious MSIV closure)	Input on trips likely to challenge PORVs were received from fuels group and incorporated in models update - see logic under gate U1QT03.

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ID	SRs	Description	Level	Comment	Resolution
AS-14	AS-A5 AS-A7 AS-A10 AS-B1 QU-A1 SC-A6	The top Unit 1 CD cutset is %ZZSU1*GMM1MRMOV. This cutset appear to be overly conservative. The base failure rate of a hand valve to transfer closed without a demand is in the 2E-7 range. The likelihood of a HV transferring closed should be 5 to 10 times lower. Further, the mission time for these MOVs is based on a 3-month test interval. As these MOVs are common to CS, LPSI, and HPSI (6 total pumps), it is highly doubtful that the MOVs will go more than a few weeks without passing flow. Considering this, the likelihood of this event is between a factor of 20 to 50 lower than currently estimated.	A	Use a more realistic failure likelihood for hand valves (or electrically isolated MOVs) spurious transferring closed. Adjust the mission time considering that 6 pumps with quarterly surveillance testing use the same flow path.	Changed exposure time for recirc valves transferring closed during standby to 2.5 months (see discussion below) (a) Changed TC rate for manual valve TC to 1.2E-07/hr based on latest generic data calc. (b) Per discussion with the pump and valve test engineer, the ECCS pumps (HPSI, LPSI, CS), and thus recirc flow paths, are tested within a week or so of each other. The 3-month exposure could be reduced at most by a couple of weeks. #-month test assumption is valid. No further change required.
DA-A1-01	DA-A1 DA-E1 DA-E2	Identifiers (i.e., type codes) are provided for the various types of components included in the PRA models for PSL Units 1 and 2. However, no evidence was provided to illustrate how the type codes are linked to the basic events in the PRA models or how to verify that the type codes are properly implemented. The system notebooks identified the basic events for which probabilities are required. This included basic events for independent and common cause failure of equpment to start and run, unavailability due to test/maintenance, and recovery of a function. Common cause failure basic events are listed in Tables 2-5 of PSL-BFJR-06-008, Rev. 2, and basic events due to unavailability are listed in Tables 13 - 16 of PSL-BFJR-11-08, Rev. 0. The identifiers for types of components are included in Tables 21 and 22 of PSL-BFJR-11-08, Rev. 0. However, a link between the type codes and basic events could not be identified.	Finding	The documentation should be enhanced to demonstrate the connection between the various types codes in Tables 21 and 22 of PSL-BFJR-11-08 and the basic events that are included in the PRA model. The enhancement can include a description of the naming scheme used for the basic events.	Naming scheme, Type Codes, and how Type Codes are linked to basic events in CAFTA models are all described in the IPE document which has not changed. PRA maintenance and update documents only describe departure or addition to specific modeling logic since IPE. However, the latest revision of PSL Data Analysis document considered all elements of the peer review recommendations were applicable.

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ID	SRs	Description	Level	Comment	Resolution
DA-C13-01	DA-C13 DA-E2	It was not clear how the assumptions obtained from knowledge plant personnel that was used to establish out-of-service hours. Table 13 identifies OOS hours and only modes 1-4 (on-line) were considered, as reported in Section 5.3. It is noted in the self assessment that the comment column in Table 13 contains "assumptions obtained from interviews with knowledgeable plant personnel" however this is not thoroughly documented. Look at examples in Section 5.3.3. - Check service water unavailability. (installed spare was OOS for a year).	Finding	Information obtained from knowledge plant personnel should be documented and include how the information is being used as part of the data analysis.	Intake Cooling Water (ICW) system has 3 pumps that are rotated for service on equal basis to allow only two pumps operating and deliver flow to 2 trains during at power operation. The identified comment was not found in the Data Analysis Document. The SR states "INTERVIEW knowledgeable plant personnel (e.g., engineering, plant operations, etc.) to estimate ranges in the unavailable time per maintenance act for components, trains, or systems for which the unavailabilities are significant basic events." There is no specific requirement to document such interview listed in this SR. All inputs to PSL Data Analysis are provided either by system engineers or ex-shift manager used to hold SRO license at PSL and currently working as PRA group member in support of PSL PRA development whose review and signature meets, if not exceeds, the requirements of this SR. The Revised PSL Data Analysis document was further enhanced to add a paragraph to clarify how input is received or communicated from knowledgeable plant personnel during completion of PSL Data Analysis. This F&O is considered resolved.

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ID	SRs	Description	Level	Comment	Resolution
DA-C14-01	DA-C14 DA-E1 DA-E2	It appears that the treatment of coincident unavailability for inter-systems was considered. However, there was no clear documentation to demonstrate such treatment. Coincident unavailability due to maintenance for differnet trains of the same system (intra-system) is not allowed by established plant procedures. Therefore, the calculation of coincidenct unvailabilities for intra-systems as a result of planned and repetitive activities was not calculated. Ther was no clear documentation on the treatment of coincident unavailability for inter-systems. Discussion with the utility PRA staff indicated that review of the plant operating history was performed to identify potential coincident unavailablity for inter-systems. No such unavailabilities were identified. The PRA staff also demonstrated that the PRA model accounts for coincident unavailability for inter-sytems by the use of appropriate mutually exclusive logic.	Finding	The documentation for data analysis should be enhanced to clearly address the treatment of coincident unavailability for inter-systems. This can include a documented review of plant operating history to demonstrate whether planned repetitive activities are performed that allowed the unvailability due to maintenance between systems.	This SR requires "EXAMINING" coincident unavailability due to maintenance for redundant equipment (both intrasystem and intersystem) that is a result of a planned, repetitive activity based on actual plant experience. The key words here are "PLANNED" and "REPETITIVE". At PSL, there is no coincident unavailability of "PLANNED" and "REPETITIVE" maintenance for redundant equipment (both intrasystem and intersystem) to be allowed, per plant T/S, procedure, guidelines, and instructions.
DA-D4-01	DA-D4 DA-E1 DA-E2	No clear evidence was provided to demonstrate that the posterior distributions resulting from Bayesian updates were checked for resonableness based on the plant-specific evidence that was used. These items have been considered and checked however no further discussion or examples are given (see Section 5.4.6). It appears this has been done, however more information should be provided to verify this (or that all the numbers should have been Bayesian updated).	Finding	The documentation should be updated and enhanced to included criteria to be used in establishing the appropriateness of posterior distributions resulting from Bayesian updates. NUREG/CR-6823 describes an approach for performing a consistency check of prior distributions used in Bayesian updates.	The Data Analysis calculation document discussed the criteria that were used to meet the intent of this SR. Revision 1 of the same document is enhanced to specifically address this F&O and added discussion for how consistency of data and Prior was performed by examining the difference between Bayesian Updated Mean value and Prior Mean value for each Bayesian updated analysis and showed that such difference is significantly less than 9E-03. NUREG/CR-6823 considered a difference of 0.05 as small. This concludes that use of Prior data in each Bayesian Update analysis was reasonable as Priors and Posteriors are comparable and close to each other with small difference.
DE-01	DA-C1 DA-D5 DA-D6 SY-B1 SY-B3 SY-B4	The common cause analysis has very few electrical components (AC and DC) considered for common cause grouping. The EDG's, batteries, and reactor trip breakers appear to be the only electrical components in the CC analysis. An evaluation or analysis to justify the exclusion of other electrical components (breakers, relays, inverters, MCC's, etc...) could not be found in the references.	B	Expand common cause analysis to justify limited electrical equipment inclusion. Or Provide clear justification for exclusion of certain AC & DC components.	CCF modeling and data was expanded to include data and components consistent with component types provided in INEL 94-0064 and latest related industry documents.

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ID	SRs	Description	Level	Comment	Resolution
HR-D1-01	HR-D1	Some inconsistencies have been identified between the documentation, the HRA calculator file and the CAFTA model; one example is AHFL1CSTIV, which is indicated as 2.7E-5 in the summary table 3.0 while appears to have the floor value from ASEP in the HRA calculator file (i.e., 1E-5) (See F&O HR-D1-01). A more conservative value has ben entered in the model, with respect of what the HRA calculator provides.	Finding	Ensure consistency between the HRA calculator results and the sumamry table 3.0 (also explicitly include the conversion from median to mean for better treacability).	The bases for pre-initiators HFEs were documented in Calculation "St. Lucie Pre-Initiator Human Interaction Analysis" that was developed by ERIN Engineering and Research, Inc. in July 2009, and was later revised by a subsequent repetitive model updates. The referenced event was confusing for peer reviewer due to use of earlier revision of HRA calculator that maintained Median values which were converted to Mean values outside the calculator and before use in CAFTA. All values used in CAFTA are Mean values.

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ID	SRs	Description	Level	Comment	Resolution
HR-G6-01	HR-G6	This SR requires a check of the post-initiator HEP quantifications, and a review of the final HEPs to check their reasonableness given the scenario context, plant history, procedures, etc. Although a cutset review was performed as part of the quantification to verify the correct application of HEPs, no detailed discussion of the final HEPs to check their reasonableness with respect to each other was found. In particular, several single HEPs were identified that had higher probabilities of success than combination events that credited the same HEPs. (e.g. event JHFPSDCR vs Combination138), but discussion of how/why that was reasonable was provided. The combinations identified do not appear to be complete. For example, the combination of FHFP1RECMFW-N and GHFPOTCTGT42 shows up in the cutsets, but is not addressed in the combination analysis. Additionally, the dependency analysis that was performed did not have a reasonableness check of the total combined human failure provided. Several total combined failures were significantly lower than 1E-10, with several lower that 1E-16. The HRA notebook does not have a lower bound for total combined human failures. A failure probability of 1E-16 is equivalent to the operation failing for all actions in the given sequence to be 1 out of 10,000,000,000,000,000 times. Based on feedback from other reviews and regulator opinion, this is a very optimistic view of the potential for operations to recover a sequence.Several total combined human failure probabilities applied in the model maybe over optimistic and exceed the best practice for the lower limit.	Finding	Provide a discussion of the reasonableness of the final HEP values taking into consideration the scenario context, plant history, procedures, etc. In particular discuss the single events that have higher probabilities of success versus their associated combination events since this is not typically seen. Apply a lower bound for total combined human failures. A typical best practice for a lower limit is 1E-06. If no lower bound is deemed warranted, discuss the validity of the combination events that have extremely low HEPs.	(a) The recovery rules in the recovery rule file are applied in order of ascending probability. In some cases, the probability of a single HFE can be less than a combination event in which the HFE is a constituent. This can occur when the HFE is not the first chronological HFE in the combination. As an example, let's say the probability of HFE A is lower than the combination of HFE A and HFE B. This can occur when the probability of HFE A is appreciably higher than HFE B, and HFE A is dependent on HFE B in the combination event AB. It can be argued that combination event probability should not be higher than any of the constituent HFE probabilities, and that is how the PSL recovery rules are applied. No matter what the interpretation, the effect on the risk calculation is minimal. (b) This is due to the fact that the list of combination events was generated for the HRA well before the PSL models had been finalized. The list of combination events was generated from the latest draft model updates that were available at the time. The models have changed considerably since that time, so it is not surprising that some unanalyzed combination events are showing up. Those cutsets that have a combination of HFEs that do not appear in the HRA dependency analysis will at least have credit for one of the HFEs applied in the cutsets. At worst, the cutsets frequency will be somewhat conservative. (c) This finding is REJECTED. Some of the combination events consist of as many as 6-7 HFEs and can have probabilities in the E-15 range. Some of the combination events in the PSL PRA model contain 3-4 HFEs which are totally independent. If the independent probabilities for these events are 1E-3, that means the combination probability will be less than 1E-9 - 1E-12, and that's before considering the other HFEs in the combination event.

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ID	SRs	Description	Level	Comment	Resolution
					Stuart Lewis of EPRI was contacted about this and his reply was ""It is not correct to say that the NRC wouldn't accept lower values for combinations of HFEs; there is nothing like a consensus position on the part of the NRC on this issue. The Good Practices NUREG suggests a value of 1E-5, but the guy who wrote that (John Forester of Sandia, with whom Jeff, Kaydee and I happened to be attending a meeting yesterday at NRC to prepare for two ACRS meetings today) claims that what he meant was that if you produce a probability for a set of potentially dependent HFEs below that value you should look at them to confirm that you have adequately addressed dependence."
HR-I2-01	HR-I2	This SR is associated with the documentation of the process used to identify, characterize, and quantify the pre-initiator, post-initiator, and recovery actions considered in the PRA including the inputs, methods and results. Although the overall HRA analysis looks very good, the current documentation for the dependency analysis and treatment of post-initiator HRAs is incomplete. The current documentation only states that all post-initiator HEPs are set to 1.0 and then fed into the HRA calculator to determine the dependency between the HEP events. There is no no discussion of how the rest of the process is performed, including how the HRA Recovery File is used to "reset" combination events to the appropriate values based on the dependency analysis, no discussion on why the HEP values in the BE file are set to 0.5, no discussion of how the HRA calculators dependency analysis was validated, etc. Additionally, there is no assurrance that all HEP combinations have been identified and evaluated - see F&O HR-G6-01 for more detail.	Finding	Provide a complete discussion of the dependency analysis process followed, including the format/structure of the recovery rules file and the basis/use of nominal values for HEPs in the BE file, the check/validation of the HRA Calculator output file for dependencies, etc.	The latest revision of HRA analysis included use of revised dependency analysis methodology that ensured generation of HEP combinations consistent with dependencies between HFEs that are considered in HRA Calculator. The HRA Analysis document included detailed steps taken to generate the revised dependency analysis.

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ID	SRs	Description	Level	Comment	Resolution
HR-I3-01	HR-13	There is no discussion on model related uncertainties for pre-initiator HRA calculations. For post-initiator HFE, the EF indicated in Table 9 are then not propagated in the CAFTA file. It is therefore not clear how the uncertainty parameters are treated in the model. A complete uncertainty assessment involves both stocastic uncertainties (included in the HRA calculator) and epistemic (model) uncertinties. A discussion on the assumptions made in the analysis and their potential for imapct on the HEP calculations is required to meet the SR. The inconsistency between the EF discussed in the post-initiator HRA notebook (table 9 in Section 3.3) and the actual CAFTA file does not allow a correct uncertainty analysis. Table 9 states that generic Error Factors are used, but there are no error factors in the BE file, so it is unclear how the error factors are propagated. Also, the Combination events, and the renamed post-initiator single events are not included in the BE file so it is unclear how their error factors are included in the analysis - or if they are even considered.	Finding	Provide a discussion on the model uncertainties associated with pre-initiator and post-initiator HFEs. Ensure that the the Efs discussed in the documentation are used in the CAFTA model. Provide a discussion of how the Efs are modified/impacted by the dependency assessment.	Pre-initiator and post-initiator HFE EFs were added to CAFTA RR-file so UNCERT can use them in the uncertainty calculations. Discussion of HRA EF uncertainties is provided in the PSL HRA analysis document.
IE-01	AS-B1 IE-A1 IE-A2 IE-A5 IE-A6 IE-A9	LOSP Initiating Event was extracted from generic industry data going back 20 years or more (Calculation No.PSL-BF-JR-01-005). No data trending was applied to establish a downward trend in LOSP annual frequency. The latest biannual EPRI report on LOSP frequency concluded that: LOSP frequency has trended downward and has stabilized over the last few years. The non-trended derived PSL LOSP frequency (Total value approximately 5.3E-02 Table 6.1) is very conservative; and the probability of non-recovery of offsite power (shown in the Log-normal cumulative figures) is also too high. This high degree of conservatism in PSL LOSP frequency and associated non-recovery probabilities may lead a PRA practitioner to determine unnecessarily high risk for some applications that would otherwise be acceptable.	B	Perform PSL LOSP data trending or use the trended and analyzed data provided in the latest biannual EPRI report on Loss of Offsite Power at Nuclear Power Plants through 1999 (contact Frank Rahn at EPRI). The up-to-date generic data would be applicable to PSL and would be sufficiently conservative since PSL1 and PSL2 have not experienced any LOSP during the last dozen years. Once the EPRI data is used, the existing detailed report on LOSP frequency (PSL-BFJR-01-005) may be archived.	The latest Off-Site Power Non-Recovery probability calc document included consideration of LOSP industrial events occurred during 1997-2008 as published by EPRI documents. Further events from 2008 on will be considered during the cyclical maintenance and update of this calculation document. The downward trend of LOSP annual frequency should have improved effects on the model and thus the current levels are considered conservative.

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ID	SRs	Description	Level	Comment	Resolution
IE-04	AS-B1 IE-A2 IE-A5 IE-A6	St. Lucie includes loss of individual 120VAC instrument buses as initiators, but does not address multiple bus failures as initiators. The 120 VAC buses power the RPS/ESFAS. Failure of 2 buses could result in spurious actuation of multiple safety systems given the 2 of 4 actuation logic. This type of initiator has not been addressed for the St. Lucie PRA. Multiple actuations could have unanticipated effects such as actuation of the Feed Only Good logic for both steam generators at the same time that AFAS was actuated. This would result in no auxiliary feedwater being supplied to the steam generators. Note: panels are co-located in pairs. Construction activity noted in area. Construction materials/ debri could block cooling intakes and cause failure. This is one example of potential common cause mechanism.	B	FP&L needs to specifically evaluate the potential impact of failure of multiple 120VAC buses in section 2 of calculation PSL-BFJR-02-001, REVISION 0. If these potential initiators are found to have a significant impact on system response, they should be incorporated in the model.	Multiple instrument bus failures are judged to be a low probability. Impact of loss of two instrument busses is judged to be covered by the LODC IE which impacts two instrument channels.
IE-05	AS-B1 IE-B1 IE-B2 IE-B3 IE-B3 IE-B4 IE-C13	PSL PRA-2.P presents the ISLOCA calculation. It has not been updated since 1992. This calculation does not address the RCP seal cooler heat exchanger tube leak ISL path nor does it discuss treatment of common cause failure of valves for the other ISL paths.	B	The ISLOCA analysis needs to be updated to include the RCP seal cooler heat exchanger tube leak ISL path. The issue of common cause failure of valves also needs to be discussed. in many cases common cause failure of valves in various pathways can be discounted based on different operating conditions, but this issue needs to be discussed explicitly.	Latest update of ISLOCA analysis was completely revised and issued on 4/18/2011. The revised analysis considered all previously identified issues related to ISL.
IE-07	DA-C16 IE-A8 IE-A9 IE-C1 IE-C11 IE-C2 IE-C3 IE-C4	The IE data documentation is scattered in different reports and in different revisions of the same report. Sometimes, inconsistent values are provided (see F&O IE-05).	B	Revise IE data reports in a consistent manner	Updated IE data are revised and documented in the latest stand-alone PSL Data Analysis document.
IE-08	IE-D1 IE-D2	All of the St. Lucie PRA documentation are calculations covered by the FP&L engineering calculation procedure, ENG-QI-1.5. This procedure requires independent review and signoff of all calculations performed per this procedure. The latest St. Lucie PRA update was not fully completed at the time of the peer review so most documents had not been independently reviewed at the time of the peer review.	A	Do Review	The peer review team reviewed draft calculation document dated 2002. All calculation documents generated in support of current PRA model update were independently reviewed/signed-off and approved consistent with the current Quality Instructions and PRA standards requirements.

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ID	SRs	Description	Level	Comment	Resolution
IE-C5-01	IE-C5	The approach for generating the initiating event frequency utilizes an adjustment factor that ratios the exposure time. The exposure time is not the mission time for reliability and the approach does not produce an appropriate value for frequency of occurrence. Utilize the idea of initiating event as the first valve failure based on having to remain isolated for the period of one year. Then consider the unavailability of the other valves in the line based on exposure time. Systematically address each valve as if it is the holding valve.	Finding	The adjustment factor is not appropriate for unavailability.	Resolution in-progress.
IE-C6-01	IE-C6	A screening approach is utilized for some lines based on low frequency but this is not quantified. The SR indicates a frequency expectation for screening. Define the estimate for the lines screened on low frequency and show that the calculated frequency supports screening.	Finding	There is not quantification of excluded ISLOCA scenarios.	Resolution in-progress.
IE-C9-01	IE-C9 IE-C10 SC-A5	The fault tree model used for the ISLOCA paths assumes that the status of all valves is known when the plant is brought online and the corresponding exposure time is the refueling interval. However, based on discussions with knowledgeable staff, there is no positive means to know that more than one isolation valve is actually holding. Use of status lights is not definitive since there is a +/-5% margin between light changing and valve seating. The exposure time should be based on a positive flow test which may not occur on a refueling basis but based on other studies could be as much as the life of the plant.	Finding	Model should use one year exposure time for the first failure and then do the CCDP following the failure. F&O finding level produced.	Resolution in-progress.
IFEV-A1-01	IFEV-A1 IFSN-A5	The identification of scenario-induced failures of components was not complete. A check of the flood scenario discussion and supporting information provided in Excel spreadsheet, PSL_FrequencyCalculations_AndTag_Database_3-24-2011.xlsx, indicated that corresponding plant initiating event group for each flood scenario was identified. However, identification of scenario induced failures of components was not complete, as discussed in F&O IFSN-A5-01.	Finding	Refer to F&O IFSN-A5-01.	Equipment locations and vulnerabilities are discussed in the Internal Flooding Calculation document which is currently being further revised and in the process of final approval.

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ID	SRs	Description	Level	Comment	Resolution
IFEV-A7-01	IFEV-A7	The considereation of human-induced floods was not included in the internal flooding evaluation. The considereation of human-induced floods was not included in the internal flooding evaluation. EPRI report 1013141 that provided generic data for flood initiating event frequencies stated that "Human induced causes of flooding that do not involve piping system pressure boundary failure such as overfilling tanks and inappropriate valve operations that release fluid from the system are not included."	Finding	Human-induced floods should be evaluated and included in the internal flooding analysis.	As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operating practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review that identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.
IFEV-B3-01	IFEV-B3	Sources of model uncertainty and related assumptions were not documented. A review of the notebook sections that discussed the flood scenarios did not provide any evidnce that uncertainties associated with the internal flooding initiating event frequencies was addressed and documented. For example, a source of uncertainty may include the range of the initiating event frequencies for each flood scenario.	Finding	The sources of model uncertainty and related assumptions should be documented to meet this requirement.	The main internal flood analysis document as well as Internal Flood Quantification document were further revised to addresses the analysis uncertainty.

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ID	SRs	Description	Level	Comment	Resolution
IFPP-B1-01	IFPP-A5 IFPP-B1 IFPP-B2 IFSN-A12 IFSN-A13 IFSN-A15	The documentation associated with the plant partitioning is scattered between the initial portion of the document and the walkdown report in Attachment B, which in reality is a discussion of the screening of main structures such as major structures. The walkdown report does not include any explanatory picture and mixes the definition of the area and their screening, without spelling out the generic criteria used for the screening of specific structures. This organization of the information is prone to confusion; moreover, since the area identification and the screening are mixed, some overlook have been noticed. For example, the walkdown notes explicitly mention which bldg has been walked down and the DG BLDG is not listed among those, still, the screening of the DG BLDG is only discussed in the walkdown report and they are all screened out on the basis that there is no service water (DG are air cooled); there is nevertheless no mention of the potential spray effects of Fire Protection system on a single DG (FP lines have been noticed during the walkdown that may have the potential to spray on the DG cabinet). While the screening of the DG BLD may still be possible (FP lines may be dry since there are large FP valves immediately outside of the DG building that may be diluge valve, or the DG AOT may be sufficient to recover from a spray event on the DG cabinet), the presence of a flood source that has the potential for impacting PRA equipment needs to be addressed. The screening out of the Turbine Building is another example of screening process inconsistent with the screening criteria provided in the standard. While it is true that the TB BDLG is open, a rupture in the condenser expansion joints will induce an initiating event and for this reason the area cannot be screened out for flood considerations. The flood scenario generated by a rupture of the condenser expansion joint may be screened for other reasons (e.g., it may be folded into an already existing IE category with identical plant effect but higher IEF), still a discussion of the reasoning and of the screening criteria needs to be provided. Finally, section 4.1.1 points to the walkdown notes but incorrectly indicating Attachment C rather than Attachment B.	Finding	Clearly spell out in the text the screening criteria used for not including in the analysis some of the major structure and assure that the walkdowns notes support the screening. Specify in the walkdown report if/how the information collected through existing plant database/documentation has been checked for accuracy to ensure that the plant partitioning reflects the as built, as operated plant. For screening of areas, it is suggested that a summary table is provided with the areas that are screened and the asscoiated rationale.	No flood zones or flood sources located within the reactor auxiliary building were screened out. The flooding analysis document was further revised to address flooding originating in the other buildings or areas even if this is not truly "internal" flooding. These do not result in additional scenarios that need be quantified as no previously unaddressed reactor scram need ensue after such an event. This SR explicitly requires discussing other than spray/submergence failure modes.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
IFPP-B3-01	IFPP-B3	There is no discussion of sources of uncertainties associated with the plant partitioning phase. The SR specifically requires a discussion of the uncertainties associated with the plant partitioning. The self-assessment table (Table 3.5.1.1) points to the quantification notebook, which is not addressing uncertainties associated with the plant partitioning but only stocastic uncertainties propoagation in the final results.	Finding	Explicitly discuss sources of uncertainties associated with the plant partitioning phase. Plant partitioning is for example highly dependent on location and normal position of doors (i.e., a normally open rather than a normally closed door can change the definition of a flood area).	The internal flood analysis documents were revised to addresses the analysis uncertainty. The principles governing the plant partitioning into flood zones are discussed in the analysis document. In general, flood zones are individual plant areas that could reasonably contain or delay propagation of water or in which water levels might be significantly different to those in adjoining areas. Walls, curbs and doors were used to identify flood boundaries. Individual adjoining rooms were combined into single flood zones if there is no impediment to the propagation of flood water between them. Conservative assumptions governing the quantification of the model – all equipment vulnerable to spray damage assumed to fail at the onset of flooding in a flood zone, no credit for recovery of flooded components, no credit for drains that would limit the height of the flood - would more than compensate for any uncertainties in the partitioning scheme. The analysis document was further revised to state that the possibility of a normally closed door being open was considered in calculating flood heights but was found to mitigate the consequences in scenarios of concern.

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ID	SRs	Description	Level	Comment	Resolution
IFQU-A10-01	IFQU-A10	Some inconsistencis in the mapping between the flood events and the basic events associated with impacted equipment has been identified. According to table 20, the top cutsets have to do with ATWS induced by a spray event on the reactor trip switchgear. Spray on the reactor trip switchgear would result in loss of power, which would result in the trip itself. Therefore, even though the reactor trip switchgears are actually impacted by the spray event, the flood initiator needs not to be mapped with the basic event associated with the switchgear because their failure is in the direction of the success. Another example of suspect inconsistency in the mapping is observed from the review of the main CDF contributors. A spray from room 1RAB43-59/58 is not expected to impact both trains since the originating room only hosts 1 train of batteries.	Finding	Review mapping between impacted components and associated basic events to ensure that the flood induce failure is consistent with the failure mode modeled in the actual BE.	Mapping between impacted components and associated basic events was reviewed to ensure that the flood induce failure is consistent with the failure mode modeled in the PRA model. Changes to the mapping tables were implemented to address the concerns of this F&O. The one inconsistency related to the spray event affecting the trip switchgear has been corrected. The scenario referenced in the second part of the review comment does not involve a spray in rooms 1RAB43-58 and -59 but rather a flood emanating from the battery rooms to the neighboring switchgear rooms through the connecting doors and submerging various electrical components inside. Since the analysis does not credit isolation of the break, the flooding will persist over the number of hours. The mapping reflects the equipment disabled by the accumulating water not only in the battery rooms but also in the neighboring electrical rooms, affecting both electrical trains.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
IFQU-A1-02	IFQU-A1	Some inconsistencies in the mapping between the flood events and the basic events associated with impacted equipment has been identified. According to table 20, the top cutsets have to do with ATWS induced by a spray event on the reactor trip switchgear. Spray on the reactor trip switchgear would result in loss of power, which would result in the trip itself. Therefore, even though the reactor trip switchgears are actually impacted by the spray event, the flood initiator needs not to be mapped with the basic event associated with the switchgear because their failure is in the direction of the success. Another example of suspect inconsistency in the mapping is observed from the review of the main CDF contributors. A spray from room 1RAB43-59/58 is not expected to impact both trains since the originating room only hosts 1 train of batteries.	Finding	Review mapping between impacted components and associated basic events to ensure that the flood induce failure is consistent with the failure mode modeled in the actual BE.	"Mapping between impacted components and associated basic events was reviewed to ensure that the flood induce failure is consistent with the failure mode modeled in the PRA model. Changes to the mapping tables were implemented to address the concerns of this F&O. The one inconsistency related to the spray event affecting the trip switchgear has been corrected. The scenario referenced in the second part of the review comment does not involve a spray in rooms 1RAB43-58 and -59 but rather a flood emanating from the battery rooms to the neighboring switchgear rooms through the connecting doors and submerging various electrical components inside. Since the analysis does not credit isolation of the break, the flooding will persist over the number of hours. The mapping reflects the equipment disabled by the accumulating water not only in the battery rooms but also in the neighboring electrical rooms, affecting both electrical trains.
IFQU-A5-01	IFQU-A5	Flood-specific actions are credited as successful without a supporting HRA. There are examples of flood specific HRA (e.g., isolation of CC header) that are credited as successful without an HRA being performed.	Finding	Perform HRA on the action that are credited.	The latest revision of Internal Flooding Quantification included revised Flooding-HRA analysis consistent with the EPRI guideline.
IFQU-A6-01	IFQU-A6	Flood impact on HRA is not documented. Section 4.5 on HRA only lists the changes made to the HEP (or not made) without any explanation of the reason why. There is no discussion on how the flood specific PSF are addressed and of which flood scenario requires modification of existing HEPs or dependency values.	Finding	Document the HRA for flood-induced PSF.	The latest revision of Internal Flooding Quantification included revised Flooding-HRA analysis consistent with the EPRI guideline.
IFQU-B2-01	IFQU-B2	The flooding qauntification notebook only provides a list of FRANX files and a summary of the results. The documentation of the PRA modeling and quantification of the internal flooding can only be inferred by the FRANX tables.	Finding	Explicitly provide a discussion on how the information from the Internal Flooding analysis notebook and associated attachments is translated in the PRA modeling through the FRANX software.	The latest revision of Internal Flooding Quantification document included discussions and listing of mapped tables and data used in FRANX.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
IFQU-B3-01	IFQU-B3	There is no discussion on how the EF are calculated for the stocastic uncertainties in the Internal Flooding analysis notebook (section 4.2 or 4.3 provide the IEF calculations for the various cases but not the EF, which do not also appear in the "calculation" tab of the Excel file PSL Frequency Calculations and TAG Database (3-24-2011).xlsx. An unceratinty analysis is presented in the Flooding quantification notebook but the EF associated with the Flooding initiators are not discussed.	Finding	Discuss the EF associated with the flooding initiators.	The latest revision of the internal flooding quantification document provided definition of Error Factors and discussion of how the Error Factors were used in the quantification.
IFSN-A5-01	IFEV-A1 IFSN-A1 IFSN-A2 IFSN-A5 IFSN-A6 IFSN-A10	While the equipment located in each area is indicated, its vulnerability is not always specified. For example: Appendix A for room 1RAB43-56 lists (under the "equipment located within area") equipment without any consideration on its vulnerability (i.e., elevation and spray vulnerability columns are empty). This appears in numerous other flood areas. This challenges the reliability of the selection process for equipment impacted for each scenario since there is no clear way to defend how the equipment impacted from each scenario is selected.	Finding	Ensure the information associated with the "equipment located within area" is entered. Specify if the equipment listed in that section is PRA equipment or other equipment that may be able to induce an IE.	Equipment locations and vulnerabilities are presented and discussed in the Internal Flooding Analysis document.
IFSN-A6-01	IFSN-A6	There is no discussion on other than spray/submergence failure modes SR explicitly requires to discuss other than spray-submergence failure modes.	Finding	If other-than spray and submergence failure modes are not addressed, explicitly say so.	Failure modes associated with pipe whip, jet impingement and the other consequences of High Energy Line Breaks were explicitly considered in the St. Lucie flooding analysis. However, as noted in the analysis document, the main steam and feedwater lines at PSL run outside the reactor auxiliary buildings into containment; their rupture is therefore not an internal flooding event. The concern is therefore limited to the rupture of the steam generator blowdown system in the piping penetration room and the steam generator blowdown room and the chemical and volume control system in the piping penetration room, the letdown heat exchanger room and the valve gallery. These flooding scenarios are fully addressed in multiple sections in the PSL Internal Flooding Analysis document.

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ID	SRs	Description	Level	Comment	Resolution
IFSN-A7-01	IFSN-A7	There is no discussion on the basis for the assessing the vulnerability of equipment located in the area. There is no clear treacability of the actual equipment impacted for each scenarios. For example, the equipment listed in room 2RAB-10-16B includes equipment such as HCV-3625 and HCV-3627. For the first valve there is an indication of the elevation from the ground, but not for the second one; for both valves there is no indication of spray vulnerabilities. Both these valves are not indciated as impacted in the scenario 4.3.1.71 (according to the Excel file "PSL Frequency Calculations and TAG Database (3-24-2011).xlsx" under the calculation tab or under the "Unit_2_Tagged_Sections(QA)" tab. It is not possible to asses why they are not included and if this is appropriate.	Finding	Identify for each equipment the basis for their vulnerability such that the impacted equipment selection can be assessed. Examples of information that can be provided to facilitate the analysis are: Are they protected/sealed? Do they fail as they are and in the required position to respond to an accident? Are they manual or check valves not vulnerable to spray or submergence? Are they PRA equipment or not (see also F&O IFSN-A5-01).	As listed in selected scenario descriptions in the internal flood analysis document, damage to the individual components can be subsumed into the damage to pumps and the loss of equipment trains. Accordingly, there is no need to address the failure of these individual components. The document was further revised to highlight this position.
IFSO-A3-01	IFSO-A3	The screening of potential flood sources for each of the flood areas could not be determined. It does not appear the screening of any potential flood sources was performed. The flood areas for St. Lucie units 1 and 2 are listed in Table 4.1.1.1 and shown graphically in Figures 4.1.1.2-4.1.1.7 of PSL-BFJR-11-005, Rev. 0 for St. Lucie Unit 1. Likewise, the flood areas for St. Lucie Unit 2 are shown in Figures 4.1.1.8-4.1.1.13 of PSL-BFJR-11-005, Rev. 0. There is no listing of flood areas that were screened out from futrher evaluation and it could not be determine if screening of flood areas was performed. For example, no potential flood source is listed in Attachment A for the Control Room for Unit 2 (i.e., flood area 2RAB62-42i). This is a flood area that may be screened out becaue of the lack of a potential flood source. Attachment A noted that the emergency diesel generator flood areas were screened out, see page B-2 of PSL-BFJR-11-005, Rev. 0, from further evaluation. The screening of the emergency diesel generator flood areas is not allowed becaused it contains fire protection piping as a potential flood source and PRA-related component.	Finding	The screening of flood areas should be performed and documented. This should included the documentation of criteria used and their application in performing the screening and an assessment of fire protection piping that requires pre-actuation. [More refinement later!]	No flood zones or flood sources located within the reactor auxiliary building were screened out. The flooding analysis document was augmented to address flooding originating in the other buildings or areas even if this is not truly "internal" flooding. These do not result in additional scenarios that need be quantified as no previously unaddressed reactor scram need ensue after such an event. This SR explicitly requires discussing other than spray/submergence failure modes.

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ID	SRs	Description	Level	Comment	Resolution
IFSO-A4-01	IFSO-A4	No evidence was provided to indicate that human-induced mechanisms were considered to determine their impact as potential sources of flooding. The flooding notebook indicated that the EPRI guideline, as documented in report 1019194, was used in performing the flooding analysis. The EPRI Guidance identified the flooding mechanism that would result in a release. No evidence could be found on the treatment of human-induced flooding. It appears that only pipe failures were considered as flooding mechanism.	Finding	All flood-induced mechanisms should be identified, including human-induced mechanisms that occur during maintenance activities and the failure modes of piping and other components.	As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operation practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review who identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.
IFSO-A4-02	IFSO-A4	Flooding mechanisms including the failure modes of pipes, tanks, etc. are required to be identified for each potential flooding source. No evidence can be found in the documentation to indicate that human-induced flooding mechanisms were addressed. The flooding notebook indicated that the EPRI guideline, as documented in report 1019194, was used in performing the flooding analysis. The EPRI Guidance identified the flooding mechanism that would result in a release. No evidence could be found on the treatment of human-induced flooding. It appears that only pipe failures were considered as flooding mechanism.	Finding	All flooding mechanisms identified in the supporting requirement should be addressed in the identification of plant-specific flood sources.	As noted in the main Flooding Analysis document, no condition reports that would reflect such problems, associated with the possibility that plant design and operation practices might affect the likelihood of flooding, were to be found. This possibility was reviewed with experienced plant staff after the peer review who identified one issue "the periodic transfer of waste water from Unit 2 to Unit 1." The document was further revised to address the issue.
IFSO-A5-01	IFSN-A1 IFSN-A3 IFSN-A16 IFSO-A5	Capacity and temperature/pressure of flood sources are not clearly defined. While the flow rate is explicitly discussed for each source that is indicated in Appendix A and then discussed in the various scenario definition sections, there is no explicit discussion of the overall capacity of each source. In the SR explicitly requires identifying the capacity. For example, for scenario 4.2.1.1, a human action is credited to isolate the CC header, which has impact on the overall source capacity, which in turn impacts how far along the potential propagation path the scenario may have potential impact (i.e., if the capacity is higher because the human action is not successful, some water can propagate in the switchgear room at the lower elevation).	Finding	Clearly define for each scenario what is the overall flood capacity is associated to each source. If the capacity is modified by an operator action, clearly identify the operator action and discuss the operator action in the analysis, addressing whether different set of equipment (due for example to a different or more extensive propagation) is impacted in case of a successful or unsuccessful operator action.	Temperature and pressure data have been added to the release scenario spreadsheet created in response to the finding related to SR IFSN-B1.

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ID	SRs	Description	Level	Comment	Resolution
IFSO-A5-02	IFSO-A5	The capacity of the flood sources and operating conditions (i.e., pressure and temperature) were not included in the characterization of the release for each flood. The characteristics of release for each flood source were identified in terms of the type of breach, range of flow rates, and capacity of the source. Based on information provided in Attachment D, a breach that results in a spray event is characterized by a flow rate of approximately 100 gpm. It appears that any breach that results in a flow rate greater than 100 gpm is characterized as a "flood" rupture. A check of the information related to Flooding Scenarios provided in Sections 4.2 and 4.3 revealed that in general the flood sources for each of the flood areas are characterized the a range of flow rates, and in certain cases the capacity of the flood source is also included. However, no clear evidence was provided to indicate the operating characteristics (i.e., pressure and temperature) of the flood source).	Finding	The operating conditions and capacity of each flood source should be included in the characterization of the release.	Temperature and pressure data have been added to the release scenario spreadsheet created in response to the finding related to SR IFSN-B1.
IFSO-A6-01	IFQU-A11 IFSN-A17 IFSO-A6	Confirmatory walkdown to assess the accuracy of the information associated with the source identification and scenario definition were not performed. One walkdown was performed before the identification of the flood source began but flood sources have not been confirmed during a dedicated confirmatory walkdown. Some potential inconsistencies between the isometric drawings used for the identification of the flood sources and actual configuration has been observed during the peer review walkdown. For example Appendix A indicates more than 138' of CC piping in the U2 Battery room A (2RAB43-35) but no CC piping has been observed in the room during the walkdown. On the other hand, demin water lines to the emergency eyewash have been observed during the peer review walkdown in the battery room, which are not listed in the Appendix A datasheet. 2RAB43-36 also does not show DW lines although it is expected that eyewash station are also present and they are indeed shown in the architectural drawing). In 2RAB43-36, the batteries are mentioned to be potentially vulnerable to spray from fire protection but no fire protection is listed as potential source in the room. Appendix A shows multiple examples of datasheet being incomplete even for critical rooms such as the ECCS rooms (see for example 2RAB-10-16B) that would challenge the selection of impacted equipment.	Finding	Perform confirmatory walkdown for flood source identification, and vulnerable equipment and confirm the information collected from other sources and listed in Appendix A.	Confirmatory partial-walkdowns were performed after development of this F&O and pipe isometric drawings were re-reviewed for accuracy. The walkdown revealed that the CCW piping segments inside the vital battery rooms at el. 43' are hidden within a pipe chase near the ceiling and therefore not visible. As a result, the analysis has been corrected by deleting the CCW piping from the list of potential flood sources in the battery rooms. Any water from postulated breaks inside the chase was assumed to divert from the battery rooms to the adjoining rooms. However, the water supply pipe to the shower station was added as a potential flood source in each battery room. The spreadsheet calculating rupture frequencies was updated with the above corrections which also corrected the input to FRANX. Finally, the datasheets were reviewed and updated as well.

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ID	SRs	Description	Level	Comment	Resolution
IFSO-B2-01	IFSO-B2	A list of flood sources that require further evaluation was not developed. The criteria that were used for screening flood sources were not clearly developed and documented. Flood sources were identified and included in the documenation. The screening criteria used to eliminate flood areas and flood sources from further evaluation was not documented. It appears that some level of screening was performed for flood areas and flood sources. However, a listing of the flood sources that require further examination was not provided.	Finding	Screening criteria used to eliminate flood sources from further evaluation should be developed and documented. The results obtained from applying the screening criteris should also be included in the documentation.	As stated in Section 4.1.3 of the PSL Internal Flooding Analysis, no screening of flood sources (within the reactor auxiliary building) was performed. The flood sources in all flood zones within the auxiliary and fuel handling buildings are listed in analysis document. All listed flood sources are explicitly considered in the analysis. Flood sources in other buildings were not considered unless their rupture might precipitate or ensuing damage causes a reactor scram. The screening of these other buildings is addressed in the response to the finding pertaining to IFEV-A6.
IFSO-B3-01	IFSO-B3	No evidence was provided that discusses modeling uncertainties and related assumptions associated with flood sources. A review of the notebook sections that discussed the potential plant floods and flood scenarios did not provide any evidnce that modeling uncertainties and related assumptions associated with flood sources was documented.	Finding	A discussion on modeling uncertainties and related assumptions associated with flood sources should be included as part of the documentation.	The main internal flood analysis document as well as Internal Flood Quantification document were further revised to addresses the analysis uncertainty.
MU-02	N/A	PSL developed no criteria upon which to base the need for a model update. Impacts written against the model may remain pending for a long time. Incorporating into the model a pending impact is based only on judgment call. In addition, a fixed periodic PRA model update schedule should be established. The update periodicity should be consistent with the principle of a living PRA.	A	Like most other plants, PSL should establish a time cutoff (say within 30 days) for implementing model impacts that has about 15% or more change to CDF / LERF values. The 15% change could be up or down from the existing CDF / LERF values. Prior living PRA-based decisions should then be re-examined for continued validity. PRA customers should be advised of the change in CDF / LERF as soon as the implementation of the change is completed. A fixed periodic PRA model update schedule should be established. The update periodicity should be consistent with the principle of a living PRA.	The PRA models maintenance and update are developed in accordance with NextEra fleet procedure/standard "PRA CONFIGURATION CONTROL AND MODEL MAINTENANCE". The frequency of model update is based on priority setting of the proposed changes to be developed by pertinent model custodian/staff engineer.
QU-02	AS-C3 DA-E3 HR-I3 IE-D3 QU-E4 SC-C3 SY-C3	A lot of results sections in the quantification report are blank with a "later" in place of the table or results.	B	Finish quantification and fill appropriate tables.	The current PRA Update documents included final results and completed analysis.

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ID	SRs	Description	Level	Comment	Resolution
QU-04	AS-B5 AS-B6 AS-C3 DA-E3 HR-I3 IE-D3 QU-B6 QU-E1 QU-E2 QU-E4 QU-F2 QU-F4 SC-C3 SY-C3	No uncertainty analysis has been performed on the results from Unit 1 or Unit 2 quantification results.	B	Complete uncertainty analysis on results.	Completed uncertainty/sensitivity analysis and related evaluations are included in the current flood Quantification document.
SL-CCF-02	IE-A6	CCF of Turbine Building supply fans, i.e., IMFFCCFTBSWGRF$ (CCF FACTOR - (2/2) TURB SWGR RM FANS FAIL TO RUN), is modeled to result in the loss of 6.9KV buses 1A1 and 1B1 and the loss 4KV buses 1A2 and 1B2, but this event is not modeled as a contributor to loss of these bus initiators. A similar comment also applies to the Unit 2 model. It should be noted that these HVAC-related CCF contributors could not be incorporated into the current St. Lucie PRA model, because this model does not include the loss of HVAC as an initiating event, per a statement in Section 3.2 of the St. Lucie Initiating Events Notebook (Reference 22). However, other than to say that only two plants model a Loss of HVAC initiating event, this notebook provides little basis for its exclusion. Due to its potential risk significance, such basis is appropriate. Basis for Significance: This F&O was assigned a Significance of B, because the absence of this CCF initiator in the initiating event fault trees is judged to not meet SR IE-A6 for any CC level on CCF and this contributor is judged to be risk significant. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Add the subject CCFs to their appropriate initiating event fault trees or document basis for their exclusion.	This CCF is credited under the system part of the fault tree and if it were to be credited under IE fault tree, it would be double counted. Nevertheless, adding the CCF under the IE fault tree will have no impact on the results and conclusion of overall risk insights as the CCF probability is of the order of 2E-5 while the IE fault tree is dominated by annual breaker failure of the order of 1E-3. Thus, the finding is considered of no significance.

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ID	SRs	Description	Level	Comment	Resolution
SL-CCF-06	SY-B8	No documented evidence was available that a process was applied to identify new common cause failures due to spatial and environmental hazards. This is required by SR SY-B8 for CCF. Basis for Significance: This F&O was assigned a significance of B (rather than A), because it is judged unlikely that the results of a plant walkdown or alternative investigation intending to identify new common cause failures due to spatial or environmental hazards would discover any CCFs that are not already modeled. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Perform plant walkdowns seeking common cause failures due to spatial and environmental hazards. These may be due to radiation, heat, humidity, vibration, etc. Special attention should be given to like components, with similar functions, in the same system, and which are in close proximity of each other. Document locations inspected. Note that an operator or system engineer may be good company on such walkdowns, because they will be familiar with plant equipment. Document findings, walkdown dates, locations visited, personnel participating, etc. Note that these walkdowns could be incorporated into walkdowns conducted for fire risk model development or flooding model updates. However, the CCF inspection is not necessarily compatible with the objectives and focus of these other inspections.	The System Notebooks were revised to include walkdowns worksheets where applicable.
SL-CCF-10	DA-D6	Plant-specific data was not reviewed for CCF events as required by SR DA-D6. Basis for Significance: This F&O was assigned a Significance of B, because the plant-specific review is required for CC II, but the "discovery" of plant-specific CCFs that are not included in the generic CCF database is unlikely given that the generic databases were developed via extensive and exhaustive research efforts. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	B	Perform a plant-specific data review for CCF events. Compare findings with data in the NRC generic database. If any new CCF events are identified, request INEL to consider them for possible inclusion into the NRC database.	Due to limited resources, limited scope review of plant-specific data found no event to be added to those being considered in NRC CCF database.

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ID	SRs	Description	Level	Comment	Resolution
SL-CCF-12	IE-A6	The CCF of ICW traveling screen plugging and the CCF of ICW strainers plugging as contributors to the loss of ICW initiator fault tree are missing from the model and no explanation for their absence is provided. Common cause contributors to the loss of ICW are judged to be both credible and potentially risk-significant. This judgment is based on the failure of a intake screen reported in LER 84-09, 1011/84 (Unit 1), the fact that these issues are addressed in the plant Off-Nominal Operating Procedure 064030, and that data is available for both of these failures in the NRC CCF database. Given that common cause is likely to be a dominant contributor to the loss of ICW and that the nominal loss of ICW frequency is judged to be very low (~1E-5/rx-yr), the modeling of the loss of ICW initiating event is judged to not meet SR IE-A6 for any CC level on CCF. Basis for Significance: This F&O was assigned a Significance of A due to its potential risk significance. Additional Discussion: This finding applies to both St. Lucie Unit 1 and 2 PRA models.	A	Evaluate the appropriateness of including common cause contributors to loss of ICW and, if appropriate, include them in the model, otherwise, document basis for their exclusion. Note that that it may be more appropriate to use a plant-specific estimate of the loss of ICW due to environmental effects in lieu of using generic CCF component CCF failure data (such as traveling screen plugging or service water pump strainer plugging) to estimate the loss of ICW frequency.	The PSL models were revised to include CCF of ICW traveling screen plugging and the CCF of ICW strainers plugging as contributors to the loss of ICW system. They were not considered under IE fault tree to eliminate double-counting.
ST-01	SC-A6	FP&L does not directly address reactor vessel capability. In section 2.3.5 of PSL-BFJR-02-001, FP&L dismisses reactor vessel rupture as being of low risk significance because of a low generic failure probability and also dismissed PTS as being of low risk significance based on generic analyses. Therefore, reactor vessel failure is not included in the model at all.	B	Reactor vessel rupture should be included in the model and quantified using the generic failure frequency. This will not have any significant impact on CDF but will provide a placeholder to address any future issues. This is consistent with industry practices and NEI subtier evaluation criteria.	Vessel rupture Initiating Event is considered in the PSL models. The considered frequency was adopted from CEOG position paper "Evaluation of the Initiating Event Frequency for Reactor Vessel Rupture".
ST-02	LE-D1 LE-D1 LE-D2 LE-D2 SC-A6	The containment capability analysis included in the IPE submittal is a simplified analysis based on the generic approach in NUREG/CR-2442 and NUREG/CR-3653 using St Lucie specific information in the simplified equation. This analysis provided an estimate of containment ultimate pressure value that was used to generate a containment fragility curve based on containment fragility curves for other similar containment designs shifted so that the median was at the St. Lucie ultimate containment pressure. The analysis did not address temperature effects and only included a single failure mode, liner tear at the spring line. The analysis did not address other containment failure points such as liner tear at the containment hatches or penetrations. The level 2 analyses did consider release pathways including containment bypass, containment isolation and containment failure. Only the single failure mode of unspecified location was used for containment.	B	If the level 2 analyses are to be updated, a more detailed containment capability analysis should be performed to include temperature impacts on material properties and to evaluate other potential failure locations and sizes. If the level 2 analyses are not updated, FP&L may want to switch to the NRC simplified LERF model where details of the containment capability evaluation are not as important.	A simplified approach for Level 2 analysis was developed and issued by Westinghouse in 2009. The analysis has considered all aspects of containment capability features. The analysis was peer reviewed and there with no findings.

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ID	SRs	Description	Level	Comment	Resolution
SY-01	N/A	There are no references to engineering calculations or analyses to support the system analysis success criteria, either in the system analysis documents or the accident sequence analysis. The basis for success criteria should be included in the system analysis documentation in order to facilitate review, update, and application of the model. For example, for AFW, the success criteria section of the AFW system analysis document does not give a basis for the success criterion that is described (flow to 1 SG). The required flow rate to remove decay heat should be compared to the capacity of a single pump, including the effects of potential flow diversion through the recirc line (since failure of the recirc line is assumed to be subsumed in the injection failure) and blowdown (since isolation of blowdown is assumed not to be needed). This could be done using engineering analysis or thermal-hydraulic analysis, but the basis should be described in the AFW system analysis document. Another example is the basis for the AFW success criterion for ATWS (flow to both SGs).	B	Include in success criteria section of system analysis documents references to engineering calculations and thermal-hydraulic analyses that provide the basis for the success criteria.	Stand-alone Success Criteria documents were developed and issued in 2009 for pre-EPU and post-EPU, respectively.
SY-08	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	It appears that in general key control systems in the St. Lucie Plant are not modeled. AFW flow control is not modeled. The AFAS system appear to control the based on the NR SG water level (opens at 19% decreasing) closes at 29% NR. There is no calculation available to determine the number of cycles required for automatic flow control. Additionally the consequence of steam generator overfill is not modeled. It should be noted that increased cycles affect a wide array of components: the relays in the control circuitry, the check valves cycled as flow is interrupted to the SG, etc. This affects not only the independent failure rates, but the common cause failure likelihood as well.	A	Model the components within the control systems down to the relay level. Provide a basis for number of cycles. Model all of the consequences control system failures (e.g. overfill/underfill) or provide a defendable discussion on why the consequences are not modeled.	Per discussions with operations personnel, AFAS would start pumps and open flow valves to provide AFW flow to SGs. Small adjustments to valve position over time would be performed by the operator to maintain desired SG level. There would not be a series of valve open and close cycles. It is judged that the assumed 3 valve cycles would be adequate to capture or bound the total valve failure prob.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
SY-12	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	It appears that in general key control systems in the St. Lucie Plant are not modeled. In the fault tree the AFW flow control system is demanded 3 times, but the basis for using 3 demands is unclear. No analysis has been done to determine the number of cycle the AFW system will undergo. Further, the common cause MOV demand failure rate does only considers a single demand. The model does not differentiate between an overfill and underfill. Overfills in general could lead to the failure of the turbine driven AFW pumps. Note: If the MOVs are demanded twice, it is doubtful that the failure likelihood would double. But it is also clear the failure likelihood will increase. Given the importance of the AFW MOVs, any increase to the failure rates can be quite significant.	A	Model the components within the control systems down to the relay level. Provide a basis for number of cycles. Model all of the consequences control system failures (e.g. overfill/underfill) or provide a defendable discussion on why the consequences are not modeled.	Per discussions with operations personnel, AFAS would start pumps and open flow valves to provide AFW flow to SGs. Small adjustments to valve position over time would be performed by the operator to maintain desired SG level. There would not be a series of valve open and close cycles. It is judged that the assumed 3 valve cycles would be adequate to capture or bound the total valve failure prob.
SY-14	AS-A10 AS-A5 AS-A7 AS-B1 QU-A1 SC-A6 SY-A1	There is not a single event model that represents debris clogging the sump (i.e. both headers blocked).	B	Add a single sump clogging basic event to both sump headers.	The current models included CCF event for sump plugging.
SY-15	DA-A4 DA-C1 DA-C2 DA-C3 DA-C6 DA-C7 DA-C8 DA-C9 SY-A11 SY-A12 SY-A19 SY-A6 SY-A7	The implementation of the Alpha Parameter methodology for common cause analysis has resulted in conditions that appear to be an over estimation of the contribution from common cause and results that do not make obvious sense (i.e. cutsets in which the common cause failure of three check valves [three AFW pump discharge check valves] is more likely than the common cause failure of two check valves [two MFW check valves to the steam generators I-V09294 and I-V09252]). The implementation of the methodology includes an assumption in the development of the parameters of staggered testing. This assumption may be non-conservative. The common cause failure of the check valves in the pump recirculation lines was not considered and justification provided for not including them was not included. Some of the issues may be the result of the use of component specific and generic alpha parameter data.	B	A clearer description of the implementation of the alpha parameter methodology would clarify part of the problem. Common cause events that are important contributors to the results should be developed consistently with component specific alpha parameter development.	The latest revision of CCF analysis document clearly describes the application of Alpha-Factor method and applicable data using 2009 INL/NRC database.

Table A-1: Peer Review Findings for Internal Events and Internal Flooding

ID	SRs	Description	Level	Comment	Resolution
SY-A2-01	SY-A2 QU-D2	The current ISLOCA model includes a failure of the SDC isolation valve to fail to close as one manner by which a loss of isolation may occur combined under an "OR" gate with a pre-initiator error involving the operator failing to correctly close the valve. If the valve mechanically failed during startup the operators would not enter into power operation so the failure mode is not valid. It could be postulated that if it failed the operators could fail to take appropriate actions which would be a pre-initiator action, but this would require the two events to be "ANDed" which would substantially decrease the likelihood of occurrence. Closing the valve at power is not plausible due to the high RCS pressure so the closure would not be valid with regard to isolation.	Finding	The current assessment for a failure to return to power with the valves in the correct position does not appear to take credit for self annunciated faults. The assessment should include failures of the operators to ensure restoration similar to post-maintenance operations.	Resolution in-progress.
TH-02	AS-A9 SC-A2 SC-A6 SC-B1 SC-B2 SC-B3 SC-B4	FP&L uses a combination of FSAR and best estimate analyses to support success criteria. There is a calculation, PSL-1FJF-93-063, which documents MAAP runs supporting success criteria evaluation. However, the accident sequence analysis report, PSL-BFJR-02-001, does not have any direct references to the cases within the MAAP analyses report linking specific success criteria assumptions to specific MAAP runs.	B	The accident sequence analysis report should directly reference any MAAP of other transient analyses used to establish specific success criteria or timing.	Stand-alone Success Criteria documents were developed and issued in 2009 for pre-EPU and post-EPU, respectively.

Attachment B - Currently Open F&Os and Impact on RI-ISI Application

Table B-1: Currently Open F&Os and Impact on RI-ISI Application

SR	Cat II Description	F&Os	Comment	Impact on Application
IE-C5	Calculate initiating event frequencies on a reactor-year basis [Note (1)]. Include in the initiating event analysis the plant availability, such that the frequencies are weighted by the fraction of time the plant is at-power.	IE-C5-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.
IE-C6	USE as screening criteria no higher than the following characteristics (or more stringent characteristics as devised by the analyst) to eliminate initiating events or groups from further evaluation:(a) the frequency of the event is less than 1E-7 per reactor-year (/ry) and the event does not involve either an ISLOCA, containment bypass, or reactor pressure vessel rupture (b) the frequency of the event is less than 1E-6/ry and core damage could not occur unless at least two trains of mitigating systems are failed independent of the initiator, or (c) the resulting reactor shutdown is not an immediate occurrence. That is, the event does not require the plant to go to shutdown conditions until sufficient time has expired during which the initiating event conditions, with a high degree of certainty (based on supporting calculations), are detected and corrected before normal plant operation is curtailed (either administratively or automatically). If either criterion (a) or (b) above is used, then CONFIRM that the value specified in the criterion meets the applicable requirements in Data Analysis (2-2.6) and Level 1 Quantification (2-2.7).	IE-C6-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.
IE-C9	If fault tree modeling is used for initiating events, QUANTIFY the initiating event frequency [as opposed to the probability of an initiating event over a specific time frame, which is the usual fault tree quantification model described in Systems Analysis, (2-2.4)]. MODIFY as necessary the fault tree computational methods that are used so that the top event quantification produces a failure frequency rather than a top event probability as normally computed. USE the applicable requirements in Data Analysis (2-2.6) for the data used in the fault-tree quantification.	IE-C9-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.
IE-C10	If fault-tree modeling is used for initiating events, CAPTURE within the initiating event fault tree models all relevant combinations of events involving the annual frequency of one component failure combined with the unavailability (or failure during the repair time of the first component) of other components.	IE-C9-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.

Table B-1: Currently Open F&Os and Impact on RI-ISI Application				
SR	**Cat II Description**	**F&Os**	**Comment**	**Impact on Application**
QU-D2	REVIEW the results of the PRA for modeling consistency (e.g., event sequence model's consistency with systems models and success criteria) and operational consistency (e.g., plant configuration, procedures, and plant-specific and industry experience).	SY-A2-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.
SC-A5	SPECIFY an appropriate mission time for the modeled accident sequences. For sequences in which stable plant conditions have been achieved, USE a minimum mission time of 24 hr. Mission times for individual SSCs that function during the accident sequence may be less than 24 hr, as long as an appropriate set of SSCs and operator actions are modeled to support the full sequence mission time. For example, if following a Loss of Coolant Accident (LOCA), low pressure injection is available for 1 hr, after which recirculation is required, the mission time for Low Pressure Safety Injection (LPSI) may be 1 hr and the mission time for recirculation may be 23 hr. For sequences in which stable plant conditions would not be achieved by 24 hr using the modeled plant equipment and human actions, PERFORM additional evaluation or modeling by using an appropriate technique. Examples of appropriate techniques include: (a) Assigning an appropriate plant damage state for the sequence; (b) Extending the mission time, and adjusting the affected analyses, to the point at which conditions can be shown to reach acceptable values; or (c) Modeling additional system recovery or operator actions for the sequence, in accordance with requirements stated in the Systems Analysis (2-2.4) and Human Reliability (2-2.5) to demonstrate that a successful outcome is achieved.	IE-C9-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.
SY-A2	COLLECT pertinent information to ensure that the systems analysis appropriately reflects the as-built and as-operated systems. Examples of such information include system P&IDs, one-line diagrams, instrumentation and control drawings, spatial layout drawings, system operating procedures, abnormal operating procedures, emergency procedures, success criteria calculations, the final or updated SAR, technical specifications, training information, system descriptions and related design documents, actual system operating experience, and interviews with system engineers and operators.	AS-03 AS-06 SY-A2-01	Resolutions to ISLOCA F&Os are currently in-progress and will be included in future updates to the PSL PSA models.	No Impact on ISI. F&Os will be tracked to resolve.

Attachment C - ISLOCA F&O Sensitivity Analysis

As part of resolving these findings and assessing their impacts, a sensitivity analysis was developed whereby the ISLOCA Internal Events fault tree model logic was completely revised and updated. F&Os IE-C5-01, IE-C9-01, and SY-A2-01 were all addressed in this update. F&O IE-C6-01 is a documentation issue and does not result in a change in modeling. The cumulative impact of the F&Os resulted in a decrease in Internal Events CDF and LERF for St. Lucie Unit 1 and Unit 2. The following are baseline Internal Events sensitivity results:

PSL1 CDF (Baseline Model) = 5.34E-06 /yr.
PSL1 CDF (w/ ISLOCA update) = 4.92E-06 /yr.
~8% decrease

PSL1 LERF (Baseline Model) = 7.79E-07 /yr.
PSL1 LERF (w/ ISLOCA update) = 3.57E-07 /yr.
~54% decrease

PSL2 CDF (Baseline Model) = 6.77E-06 /yr.
PSL2 CDF (w/ ISLOCA update) = 6.74E-06 /yr.
<1% decrease

PSL2 LERF (Baseline Model) = 2.32E-07 /yr.
PSL2 LERF (w/ ISLOCA update) = 2.03E-07 /yr.
~12% decrease

Based on the nature of the ISLOCA update which was limited to logic and fault tree structure changes and elimination of conservatism, as well as the resulting reduction in internal events CDF and LERF for the PSL Internal Events model, the changes, and associated F&O resolution, will not adversely impact PRA applications.