Florida Power & Light Company, 6501 S. Ocean Drive, Jensen Beach, FL 34957

FPL

September 17, 2004

L-2004-210
10 CFR 50.55a

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

RE: St. Lucie Unit 2
Docket No. 50-389
Third 10-Year Inservice Inspection Interval
Relief Request No. 2 Request for Additional Information Response

By letter L-2003-189 dated August 6, 2003, Florida Power & Light Company (FPL) submitted the Third Interval Inservice (ISI) Inspection Program. Included in the submittal was ISI Relief Request No. 2 regarding the risk informed Inservice Inspection program for ASME Class 1 Piping. On August 6, 2004 the NRC issued a request for additional information (RAI).

The attachment to this letter provides the FPL response to the RAI.

Please contact George Madden at 772-467-7155 if there are any additional questions about this response.

Very truly yours,

William Jefferson, Jr.
Vice President
St. Lucie Plant

WJ/GRM

A047

St. Lucie Plant Unit 2
Response to NRC Request For Additional Information
ISI Third Inspection Interval
Relief Request No. 2

NRC Request 1:

In Section 1.2 (page 3) core damage frequency (CDF) and large early release frequency (LERF) values for St. Lucie Unit 2 are summarized in the following statement: "The core damage frequency (CDF) based on 5.02-11 truncation and the large early release frequency (LERF) based on 0.01 conditional containment failure probability are 1.58E-05/Yr and 1.58E-07/Yr respectively." It is noted that the current submittal is based on a Probabilistic Safety Assessment (PSA) model updated in September 2001 (as compared to the 1999 PSA model update used for the original St. Lucie Unit 2 Risk-informed Inservice Inspection (RI-ISI) submittal of July 23, 2002). In the July 23, 2002 submittal, CDF and LERF information was provided as follows: CDF - 1.25E-05/yr and LERF - 6.00E-06/yr. This suggested a conditional containment failure probability (CLERP) of about 0.48, using the 1999 PSA models. Citing another datapoint, the St. Lucie Unit 1 RI-ISI submittal of July 30, 2003, also using a CDF model from 1999, but using a LERF model updated to 2001, provided the following data: CDF - 1.45E-05/yr and LERF - 3.43E-06/yr, suggesting a CLERP of about 0.24.

The current submittal is different in that it appears that LERF is stated as a function of CLERP, rather than CLERP having been calculated from PSA model generated results of CDF and LERF. Please explain how you derived the CLERP value of 0.01, particularly in light of the suggestion that it is now much more optimistic than in the previous calculations.

Also, please confirm that both the CDF and LERF PSA models were updated in September 2001. This is not clear from the text.

FPL Response 1:

The baseline CDF and LERF at a truncation of 5E-11 for the PSA model used for the relief request risk assessment are 1.58E-05/year and 7.44E-06/year, respectively. The LERF is calculated as follows:

$$LERF = (FRAC_{LERF} * CDF_{NON\text{-}SGTR}) + CDF_{SGTR} + ISLOCA$$

Where: $FRAC_{LERF}$ = fraction of the baseline non-direct bypass CDF leading to a large early release

$CDF_{NON\text{-}SGTR}$ = non-SGTR contribution to the total Level 1 CDF = $CDF_{TOTAL} - CDF_{SGTR}$

CDF_{TOTAL} = total Level 1 CDF

CDF_{SGTR} = Steam Generator Tube Rupture contribution to the total Level 1 CDF
ISLOCA = Interfacing System LOCA frequency

$FRAC_{LERF}$ is taken from the St. Lucie Level 2 analysis. The "0.01" addressed in Request 1 above is not the CLERF. The $FRAC_{LERF}$ documented in the 2001 Level 2 update is "0.00066". "0.01" is the original value for $FRAC_{LERF}$ calculated for the IPE. "0.01" was conservatively used for the risk calculations in support of this relief request.

Since the relief request was submitted, an updated (post-Peer Review) Level 1 model update has been completed. The updated CDF is 2.17E-05/year at a truncation of 5E-11. A Level 2 update has also been performed based on the updated Level 1 model. $FRAC_{LERF}$ is 0.00067 based on the updated Level 2 calculation. The updated input for this relief request conservatively assumes a $FRAC_{LERF}$ of "0.01" instead of the updated Level 2 calculated value of "0.00067".

The PSA input (CDF and LERF) in support of this relief request has been re-evaluated based on the updated (post-Peer Review) PSA model. The results are provided in the responses to Requests 2, 3, and 4 below.

NRC Request 2:

The discussion in support of showing that Identified IPE Weakness #2 (page 4) has no significant impact on this RI-ISI application, begins similarly to Florida Power and Light Company's (FPL's) response 11b in D.E. Jernigan to USNRC letter "Relief Request 29 Request for Additional Information Response", dated January 16, 2003. However, the current submittal omits a number of supporting statements that were included in that RAI response. Please restore the supporting statements, or otherwise explain why the use of screening values for pre-initiator human actions in STL2's current PSA continue to have an insignificant impact on your RI-ISI application.

FPL Response 2:

As discussed in the response to Request 1, the PSA input in support of this relief request has been re-evaluated based on the updated PSA model. Consistent with the previous submittal, the new cutset files generated in support of this submittal were reviewed to determine the Fussel-Vesely (FV) values for the dominant pre-initiator actions. The largest FV value noted was approximately 3.6E-02 (only one event for one case evaluated). Two events for each case had FV values between 2% and 3% and two had FV values between 1% and 2%. The risk reduction worth for the dominant pre-initiator would thus only be approximately 3%. Consistent with the original submittal, the potential impact on CDF due to more detailed analyses of the pre-initiators would thus be approximately 3% or less for the highest pre-initiator and much less than 1% for most.

It is judged that the use of unrefined pre-initiator screening values does not have a significant impact on the results and conclusions of the RI-ISI application.

NRC Request 3:

It was stated on page 7 that a Combustion Engineering Owner's Group (CEOG) peer review of a draft 2002 PSA update model was conducted in May 2002. Yet an older (September 2001) model was used in support of this RI-ISI program submittal. Please determine if any differences between the September 2001 model and the draft 2002 models might impact on the RI-ISI risk ranking and risk evaluation result, and provide the results of this determination.

FPL Response 3:

A PSA update was completed in July 2004. The PSA input for this relief request, including applicable sensitivity analyses, was re-calculated using the new model. The risk assessment results and applicable updated relief request tables are provided below.

UPDATED
TABLE 3.5-1
NUMBER OF SEGMENTS AND PIPING RISK CONTRIBUTION BY SYSTEM (WITHOUT ISI)

System	# of Segments	CDF without Operator Action (/yr.)	CDF with Operator Action (/yr.)	LERF Without Operator Action (/yr.) (1)	LERF With Operator Action (/yr.) (1)
CH	20	2.260E-11	2.210E-11	4.446E-13	4.319E-13
RC	126	6.473E-05	6.473E-05	5.836E-07	5.836E-07
SI	59	6.076E-16	6.076E-16	2.800E-17	2.800E-17
TOTAL	205	6.473E-05	6.473E-05	5.836E-07	5.836E-07

(1) LERF was based on quantification of the Level 2 model. Contributions to LERF include ($FRAC_{LERF}$ * $CDF_{NON-SGTR}$), CDF_{SGTR}, and ISLOCA, as discussed in the response to Request 1 [0.01 was conservatively assumed for $FRAC_{LERF}$]

UPDATED
TABLE 3.10-1
COMPARISON OF CDF/LERF FOR CURRENT SECTION XI AND RISK-INFORMED ISI PROGRAMS

Case	Current Section XI (/yr)	Risk-Informed (/yr)
CDF No Operator Action	5.549E-05	5.549E-05
- CH	1.379E-11	2.260E-11
- RC	5.549E-05	5.549E-05
- SI	5.048E-17	6.076E-16
CDF with Operator Action	5.549E-05	5.549E-05
- CH	1.332E-11	2.210E-11
- RC	5.549E-05	5.549E-05
- SI	5.048E-17	6.076E-16
LERF No Operator Action (1)	5.004E-07	5.004E-07
- CH	3.146E-13	4.446E-13
- RC	5.004E-07	5.004E-07
- SI	2.598E-18	2.800E-17
LERF with Operator Action (1)	5.004E-07	5.004E-07
- CH	3.023E-13	4.319E-13
- RC	5.004E-07	5.004E-07
- SI	2.598E-18	2.800E-17

(1) LERF was based on quantification of the Level 2 model. Contributions to LERF include ($FRAC_{LERF}$ * $CDF_{NON-SGTR}$), CDF_{SGTR}, and ISLOCA, as discussed in the response to Request 1 [0.01 was conservatively assumed for $FRAC_{LERF}$]

Segment Dominance Analysis:

As was the case for the original submittal, one segment (the pressurizer surge line) dominated the analysis with risk reduction worth (RRW) of 1.64 and 39% of the total CDF/LERF. An investigation showed that this was due to a high failure rate due to relatively high imposed stresses. A sensitivity study was performed to determine if the dominance of this segment masked other segments from becoming significant (i.e., RRW >1.005). For this study, the failure rate for the segment was reduced by an order of magnitude, and all calculations were repeated. Consistent with the original evaluation, the RRW for this segment was reduced to 1.064 with 6% of the total CDF/LERF while two additional segments increased their RRW from 1.003 to 1.005. The results of this sensitivity analysis performed for the original submittal contributed to the Expert Panel's decision that these two additional segments should be considered high safety significant. These two segments will remain in the high safety significant category.

Sensitivity analyses were performed in support of the previous submittal to address NRC Staff identified weaknesses in the St. Lucie IPE. The following addresses the sensitivity analysis results for updated PSA input:

1. One sensitivity analysis was related to the point estimates used for loss of component cooling water (CCW), loss of intake cooling water (ICW), loss of turbine cooling water (TCW), loss of DC, and loss of instrument air (IA) initiating event frequencies. This sensitivity analysis is not applicable for the updated risk assessment since fault trees are used in the updated PSA model to calculate the initiating event frequencies for these events instead of point estimates. It is also judged that any potential impact of these IEs on the RI-ISI would be small as RI-ISI focuses on Class 1 piping and the LOCA frequencies are estimated using probabilistic fracture mechanics.

2. Another sensitivity analysis was related to the use of human error probability (HEP) calculated using a time-independent technique. The St. Lucie PSA update included a revision to the human reliability analysis (HRA). The HEPs of concern that were originally quantified using a time-independent technique are now quantified using a time-dependent technique. The HEPs used for the original sensitivity analysis in support of the previous submittal used the same time-dependent technique as that used for the PSA update.

3. The sensitivity analysis related to use of screening values for pre-initiator HEPs is discussed in the response to Request 2.

Uncertainty analysis: See the response to Request 4 for a discussion of the uncertainty analysis results.

NRC Request 4:

WCAP 14572, Rev 1-NP-A, "Westinghouse Owners Group Application of Risk-informed Methods to Piping Inservice Inspection Topical Report," (p. 125) calls for an uncertainty analysis to ensure that no low safety significant segments could move into high safety significance when reasonable variations in the pipe failure and conditional CDF/LERF probabilities are considered. Yet, Section 3.5 (page 9) of your submittal indicates that based on insights gained from previous uncertainty analysis and based on sensitivity studies performed to address CEOG Peer Review Team comments, there was no 5th and 95th percentile uncertainty analysis performed.

FPL Response 4:

An uncertainty analysis was performed using the 5th and 95th percentile bounds for the pipe failure probabilities and CCDF/CLERF using the updated PSA results. The RRW was re-calculated for each segment to determine if any segments would move from low safety significance to high safety significance.

No low safety significant segments moved into high safety significance.

Two segments moved from the medium to the high safety significance category. These are the same two segments that moved to high safety significant based on the segment dominance sensitivity analysis discussed in the response to Request 3. These segments were classified as high safety significant by the original expert panel RI-ISI program review based on:

- Sensitivity studies resulted in increased RRW (above 1.005)
- The segments are a part of the RCS loop piping
- Failure of the segments results in a large LOCA
- Failure of the segments impacts the ability to provide SDC

Three segments that fell within the medium safety significance range for the previous evaluation remained in the medium range. An expert panel review determined these segments should be classified as low safety significant based on:

- The probability of pipe failure and associated CDF/LERF are relatively low
- The RRW for these segments is in the lower portion of the medium risk significance range
- Failure of these segments results in a small LOCA
- Failure of these segments has little impact on the ability to provide SDC

NRC Request 5:

Section 3.8 (page 11) indicates that there are 202 piping segments in the scope. Yet it breaks down the allocation into the different regions as follows: Region 1B - 9 segments, Region 2 - 2 segments, and Region 4 - 194 segments. This adds up to 205 segments total, different from the 202 segments noted. However, Tables 3.1-1 and 3.7-1 support the notion of a total of 202 segments in the scope.

a. Please confirm that there are only 191 segments in Region 4, rather than the 194 segments noted in Section 3.8.

b. Please explain why 3 segments were removed from the scope between the previous St. Lucie Unit 2 RI-ISI submittals and this one.

c. Please confirm that the segments identified as High Safety Significant (HSS) in the previous St. Lucie Unit 2 RI-ISI submittal are the same as those identified as HSS in the current submittal, or otherwise explain what has changed and why.

d. In connection with item c above, please explain why, despite the fewer number of reactor coolant (RC) system segments, there are, per Table 3.7-1, now 3 segments with RRW between 1.005 and 1.001 where there used to be only 2 such segments. Also, how was it decided that this additional "gray area" segment was not to be placed in the HSS category, particularly if the Expert Panel was not convened, as

noted in Section 3.6? WCAP-14572, Rev. 1-NP-A, denotes that the Expert Panel is to make the categorization decisions.

FPL Response 5:

Response to 5a:

There was a typographical error in the relief request submittal. The number of segments for Region 4 should be 194. The total number of segments is 205.

UPDATED
Table 3.1-1
System Selection and Segment Definition for Class 1 Piping

System Description	PRA	Section XI	Number of Segments
CH - Chemical & Volume Control	Yes	Yes	20
RC - Reactor Coolant[1]	Yes	Yes	126
SI - Safety Injection [1,2]	Yes	Yes	59
Total			205

Notes:

1. Includes shutdown cooling flowpaths.
2. Includes flow paths for high pressure safety injection, low pressure safety injection, and the passive accumulator in portions of SI.

Response to 5b:

There was a typographical error in the relief request submittal. Three segments were not removed from scope. The total number of segments is 205.

Response to 5c:

There has been no change in the segments classified as high safety significant. Consistent with the original submittal, the same 11 segments are in the high safety significant category.

Response to 5d:

Table 3.7-1 should have shown that the expert panel previously reviewed five segments that fall in the medium (1.005>RRW>=1.001) safety significant category. Two of these five segments were determined to be High Safety Significant based on:

- Sensitivity studies resulted in increased RRW (above 1.005)

- The segments are a part of the RCS loop piping
- Failure of the segments results in a large LOCA
- Failure of the segments impacts the ability to provide SDC

The other three segments that fell in the medium category were classified as low safety significant by the expert panel based on:

- The probability of pipe failure and associated CDF/LERF are relatively low
- The RRW for these segments are generally in the lower to mid portion of the medium risk significance range
- Failure of these segments results in a small LOCA
- Failure of these segments has little impact on the ability to provide SDC

An updated Table 3.7-1 is provided below.

UPDATED
Table 3.7-1
Summary of Risk Evaluation and Expert Panel Categorization Results

System	Number of segments with any RRW > 1.005	Number of segments with any RRW between 1.005 and 1.001	Number of segments with all RRW < 1.001	Number of segments with any RRW between 1.005 and 1.001 placed in HSS	Number of segments with all RRW < 1.001 selected for inspection	Total number of segments selected for inspection (High Safety Significant Segments)
CH	0	0	20	0	0	0
RC	9	5	112	2	0	11
SI	0	0	59	0	0	0
Total	9	5	191	2	0	11

NRC Request 6:

Table 3.10-1 (page 17) shows that the RI-ISI alternative inspection program will be slightly beneficial overall from a risk perspective, relative to the traditional American Society of Mechanical Engineers (ASME) Code, Section XI program. Within this overall perspective, however, the table shows a noticeable fractional increase in risk from the charging and from the safety injection systems, with a slight fractional decrease in risk from the reactor coolant system. Contained in D.E. Jernigan to USNRC letter "Relief Request 29 Supplemental Request for Additional Information," dated March 26, 2003, is Attachment 2, which is identified in this letter as Revision 3 to Risk-Informed Inservice Inspection Plan for St. Lucie Unit 2, the most recent revision on the docket that pertains to the second inspection interval at St. Lucie Unit 2. Table 3.10-1 therein indicates absolutely no change in risk in any system between the traditional Section XI and the Risk-Informed alternative programs. Please explain these apparently contrasting results, particularly in light of the changes in the population of HSS piping segments noted in the above question. Also, if relevant, please indicate how any changes in

selected weld locations between the second and third interval risk-informed programs are influencing these "new" differences in risk.

FPL Response 6:

As discussed in the response to Request 5, there has not been a change in the number of high safety significant segments.

The updated Table 3.10-1 results are included in the response to Request 3 above. The results indicate that there is no change in risk for the dominant contributor (RC) between the current section XI and proposed risk-informed programs. This is true for both CDF and LERF with and without operator action. As pointed out by the NRC Staff review, there is a fractional change between the current and risk-informed programs for the CH and SI cases. The difference in risk for the CH and SI CDF operator and no operator cases is less than 1E-11 and less than 1E-15 respectively. The change in risk for the CH and SI LERF operator and no operator cases is less than 1E-12/year and less than 1E-16/year respectively. A change in CDF of 1E-11/year or less and a change in LERF of 1E-12/year or less falls within Region III of RG 1.174 Figures 3 and 4 and are thus very small.

Consistent with RG 1.178, it is concluded that the very small increase in risk is acceptable in accordance with RG 1.174 guidelines and is consistent with the intent of the Commission's Safety Goal Policy Statement.

NRC Request 7:

In D.E. Jernigan to USNRC letter "Relief Request 29 Supplemental Request for Additional Information", dated March 26,2003, FPL Response 9 and FPL Clarification to Response 9 provided FPL's commitment to RI-ISI program situations requiring NRC notification and approval. Please restate FPL's commitment to these situations requiring NRC notification and approval as part of this submittal.

FPL Response 7:

FPL intends to comply with the standards developed by the industry and accepted by the NRC.

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 Section XI inspection period basis. Significant changes may require more expedited adjustment as directed by NRC Bulletin or Generic Letter requirements, or by plant specific feedback. The RI-ISI program would be resubmitted to the NRC for approval for:

- Changing from one methodology to another
- Changing scope of application
- Plant-specific impact of revised methodology or safety evaluations
- Industry experience determines that there is a need for significant revision to the program as described in the original submittal for that interval
- Changes that impact the basis for NRC approval in the FPL St. Lucie Unit 2 specific Safety Evaluation
- ASME Section XI Ten-Year update.

NRC Request 8:

The industry experience based on cracking of dissimilar metal welds, such as safe-end welds at V. C. Summer, Three Mile Island, and Ringhals 3 and 4, attributes the degradation mechanism to be primary water stress corrosion cracking involving Alloys 82 and 182. This degradation mechanism has not been addressed in the Topical Report WCAP-14572, Rev 1-NP-A. Please indicate if this recent industry experience was taken into account when selecting dissimilar metal welds in B-F and B-J categories for volumetric examination.

FPL Response 8:

The recent industry experience has been taken into account as it has occurred. When the St. Lucie Unit 2 ISI program was submitted (L-2003-189 on August 6, 2003), which included a relief request to implement an alternative risk informed program, only the V.C. Summer and Ringhals PWSCC events had been identified. The industry guidance at the time (INPO SEN-216) was to perform visual inspections of the RV safe-end welds. The St. Lucie Unit 2 RCS construction utilizes clad lined carbon steel pipe and does not have the alloy 82/182 weld safe ends on the reactor vessel inlet and outlet nozzles. However, this cracking event was entered into the St. Lucie corrective action program and all alloy 82/182 safe end welds were identified and visually inspected as recommended by INPO. The recent industry experience with other alloy 82/182 safe end welds has resulted in the selection of several dissimilar metal welds from high safety significant segments, in the highest temperature locations at St. Lucie Unit 2 for inclusion into the RI-ISI program. The industry experience has shown that increased temperature plays an important roll in determining susceptibility to PWSCC for alloy 82/182 welds. Therefore, FPL selected the bi-metallic weld on the pressurizer surge line nozzle to safe end weld closest to the hot leg for volumetric inspection.

In addition, FPL is following the work of the EPRI Material Reliability Project (MRP) and has committed to performing the bare metal visual inspections of all RCS alloy 82/182 welds and alloy 600 pressure boundary components as identified in MRP Letter 2003-039 (ADAMS Accession No. ML040360483). That commitment was identified in St. Lucie Units 1 and 2 response to NRC Bulletin 2004-01 (FPL Letter L-2004-160 dated July 27, 2004). The MRP recommendations were also prioritized by temperature by indicating that "priority should be given to the hottest locations (such as the pressurizer and hot leg weld locations) during the next refueling outage."

Furthermore, the EPRI MRP Alloy 600 Butt Weld Working Group is preparing its safety assessment for these bimetallic welds. Inspection recommendations (type and frequency) are expected to follow when the work is complete. When those recommendations are issued, FPL will incorporate them as they apply to the applicable locations in the St. Lucie Unit 2 RCS and include them in either the risk informed ISI program or an augmented ISI program.

NRC Request 9:

In Table 5-1, "Structural Element Selection," none of the Category B-F welds have been selected for volumetric examination. Please discuss the basis for how these welds were excluded from volumetric examination, including selection criteria and a description of each weld (size, materials (including weld metal), location and operating temperature). Also, provide the same information for B-J category dissimilar metal welds that will not receive a volumetric examination per the St. Lucie Unit 2 RI-ISI program.

FPL Response 9:

FPL has not changed the sample selected for volumetric examinations. The table below identifies the dissimilar metal welds. The weld material is alloy 82/182 (alloy 600 weld metal) for the welds in the table. The configuration varies with each joint. In general, most welds are alloy 82/182 buttering welded on the carbon steel pipe/nozzle/component followed by a post weld heat treatment (PWHT). The buttered component is then welded, with alloy 82/182, to a stainless steel safe end piece transition piece using alloy 82/182 weld metal without PWHT. The exceptions are the 3 associated with a tee on the pressurizer spray line. These three shop welds associated with the PZR spray tee are stainless steel to alloy 600 dissimilar metal butt welds without PWHT.

The original submittal identified the 6 Category B-F, item #s B5.40 & B5.50, welds associated with the pressurizer. None of these welds were initially selected due to their location in low safety significant segments (spray, safety, and relief nozzle to safe-ends) or, in the case of the pressurizer nozzle to surge line, because the dissimilar metal weld closer to the hot leg on the surge line was selected as part of the 25% sample.

The remaining 27 B-J welds contain nine welds in high safety significant segments. One weld is located on the pressurizer surge line and 8 welds are located on the reactor coolant cold legs at the reactor coolant pump connections (4 inlet and 4 outlet). Of these 9, 3 have been selected for volumetric examination. The selected welds consist of 1 surge line nozzle to safe end (identified in the B-F discussion) and 2 reactor coolant pump outlet welds. Within the surge line the only B-J dissimilar metal weld has been selected. This represents 100% of the available B-J dissimilar metal welds in the surge line or 50% of the total dissimilar metal welds associated with the surge line (combining the B-F and B-J dissimilar metal welds). For the remaining 8 welds within the high

safety significant segments, 2 of 8 welds, 25%, have been selected for volumetric examination. These 8 welds are all located on the cold leg side of the steam generator. The industry experience has shown that increased temperature plays an important roll in determining susceptibility to PWSCC for alloy 82/182 welds. The remaining 18 welds are located on low safety significant segments and are not subject to volumetric examination.

In addition, FPL is following the work of the EPRI Material Reliability Project (MRP) and has committed to performing the bare metal visual inspections of all RCS alloy 82/182 welds and alloy 600 pressure boundary components as identified in MRP Letter 2003-039 (ADAMS Accession No. ML040360483). That commitment was identified in St. Lucie Units 1 and 2 response to NRC Bulletin 2004-01 (FPL Letter L-2004-160 dated July 27, 2004). The MRP recommendations were also prioritized by temperature by indicating that "priority should be given to the hottest locations (such as the pressurizer and hot leg weld locations) during the next refueling outage."

Furthermore, the EPRI MRP Alloy 600 Butt Weld Working Group is preparing its safety assessment for these bi-metallic welds. Inspection recommendations (type and frequency) are expected to follow when the work is complete. When those recommendations are issued, FPL will incorporate them as they apply to the applicable locations in the St. Lucie Unit 2 RCS and include them in either the risk informed ISI program or an augmented ISI program.

Weld ID #	Component	Location	Material	Pipe Size	Temp (°F)	1998/A00 Code Category
503-671-A	PZR Safety Nozzle	PZR Top Head	Butt Weld	3	653	B-F
503-671-B	PZR Safety Nozzle	Top Head	Butt Weld	3	653	B-F
503-671-C	PZR Safety Nozzle	Top Head	Butt Weld	3	653	B-F
RC-506-671	PZR Relief Valve	Top Head	Butt Weld	4	653	B-F
RC-514-671	PZR Surge Nozzle	PZR bottom head	Butt Weld	12	653	B-F
RC-504-671	PZR Spray Nozzle	Top Head	Butt Weld	4	549	B-F
RC-301-771	HL to Surge Line	Hot Leg B	Butt Weld	12	600	B-J
RC-302-771	Shut Down Cooling	Hot Leg B	Butt Weld	12	600	B-J
RC-501-771	Shut Down Cooling	Hot Leg A	Butt Weld	12	600	B-J
RC-114-502-771	HL Drain	Hot Leg A	Butt Weld	2	600	B-J
RC-103C-SW-1	PZR Spray Tee	Top Head	Butt Weld	4	549	B-J
RC-103C-SW-2	PZR Spray Pipe	Tee @ Spray Nozzle	Butt Weld	4	549	B-J
RC-103C-SW-3	PZR Spray Tee @ Reducer	Top Head	Butt Weld	4	549	B-J
RC-115-1501-771-A	Elbow to A2 RCP	Cold Leg	Butt Weld	30	549	B-J
RC-115-701-771	A2 RCP to Pipe	Cold Leg	Butt Weld	30	549	B-J
RC-121-1501-771-B	Elbow to B1 RCP	Cold Leg	Butt Weld	30	549	B-J
RC-121-901-771	B1 RCP to Pipe	Cold Leg	Butt Weld	30	549	B-J
RC-112-1501-771-C	Elbow to A1 RCP	Cold Leg	Butt Weld	30	549	B-J
RC-112-1066-771	A1 RCP to Pipe	Cold Leg	Butt Weld	30	549	B-J
RC-124-1501-771-D	Elbow to B2 RCP	Cold Leg	Butt Weld	30	549	B-J
RC-124-1301-771	B2 RCP to Pipe	Cold Leg	Butt Weld	30	549	B-J
RC-1104-771	A1 SI Nozzle	Cold Leg	Butt Weld	12	549	B-J
RC-703-771	A2 SI Nozzle	Cold Leg	Butt Weld	12	549	B-J
RC-903-771	B1 SI Nozzle	Cold Leg	Butt Weld	12	549	B-J
RC-1304-771	B2 SI Nozzle	Cold Leg	Butt Weld	12	549	B-J
RC-906-771	PZRbypass spray nozl	Cold Leg B1	Butt Weld	3	549	B-J
RC-1305-771	PZRbypass spray nozl	Cold Leg B2	Butt Weld	3	549	B-J
RC-1503-771-C	Letdown/A1 drain nozl	Cold Leg	Butt Weld	2	549	B-J
RC-1503-771-A	Letdown/A2 drain nozl	Cold Leg	Butt Weld	2	549	B-J
RC-1503-771-B	Letdown/B1 drain nozl	Cold Leg	Butt Weld	2	549	B-J
RC-1503-771-D	Letdown/B2 drain nozl	Cold Leg	Butt Weld	2	549	B-J
CH-904-771	Charging Inlet Nozzle	Cold Leg B1	Butt Weld	2	549	B-J
CH-704-771	Charging Inlet Nozzle	Cold Leg A2	Butt Weld	2	549	B-J