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

FPL

January 15, 2010

L-2010-001
10 CFR 50.4
10 CFR 50.55a

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

Re: St. Lucie Unit 1
Docket No. 50-335
Inservice Inspection Plan
Fourth Ten-Year Interval Unit 1 Relief Request No. 6

Pursuant to 10 CFR 50.55a(a)(3)(i), Florida Power & Light (FPL) requests relief from Code requirements for existing and future "half nozzle" repair technique that leaves the cracks (flaws) in place which is in conflict with ASME B&PV Code Section XI. The Fourth Inspection Interval Relief Request No. 6 is essentially the same as the previously approved Third Inspection Interval Relief Request No. 26, and is being submitted to extend approval of the third inspection interval relief request into the fourth inspection interval.

The proposed relief request is based on the NRC staff's safety evaluation of the Westinghouse Topical Report WCAP-15973-P-A for half nozzle repair techniques. The attachment to this letter provides the details for the relief request.

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

Sincerely,

ES Katzman

Eric S. Katzman
Licensing Manager
St. Lucie Plant

Attachment

ESK/KWF

A047
NRR

St. Lucie Unit 1
Fourth Inspection Interval
Relief Request No. 6

Proposed Alternative in Accordance with 10CFR 50.55a(a)(3)(ii)

Hardship or Unusual Difficulty without Compensating
Increase in Level of Quality or Safety

**"REPAIR OF ALLOY 600 SMALL BORE NOZZLES WITHOUT FLAW REMOVAL"**

## 1. ASME Code Component(s) Affected

Small bore alloy 600 nozzles welded to the reactor coolant piping hot legs
St. Lucie Unit 1
Reactor Coolant Piping Nozzle Details
FPL Drawing Numbers: 8770-366, 8770-1496, 8770-3344

**TABLE 1**
**PSL1 Replacement History Alloy 600 Small Bore Nozzles on Hot Leg Piping**

| Tag ID | Hot Leg A or B | Replacement Date | Replacement Method | Reason for Replacement | Flaw Left |
|---|---|---|---|---|---|
| PDT-1121D | B | 2001 | 1/2 Nozzle Technique | Leakage | Yes |
| TE-1112HA | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1112HB | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1112HC | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1112HD | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1111X | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1122HA | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1122HB | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1122HC | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1122HD | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| TE-1121X | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1111A | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |

| | | | | | |
|---|---|---|---|---|---|
| PDT-1111B | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1111C | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1111D | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1121A | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1121B | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| PDT-1121C | B | 2005 | 1/2 Nozzle Technique | Preventive | No* |
| RC-143 | A | 2005 | 1/2 Nozzle Technique | Preventive | No* |

*No leakage had been identified during previous inspections.

(Table 1 provided in Reference 6, Request Items 1, 2 and 3.)

The ID tags identify the nozzle function; all nozzles were installed with the same bore diameter in the hot leg piping. The two sets of dimensions shown in Table 2 are dimensions of the original weld preparation. The nozzles prefixed with "PDT" and "RC-143" have the same weld prep dimensions and the dimensions are identified as "flow nozzle." The nozzles prefixed with "TE" are identified as "RTD nozzle." All nozzles are bounded by the TR.

(Preceding paragraph provided in Reference 7, Request Item 2.)

## 2. Applicable Code Edition and Addenda

ASME Sect. XI, "Rules for In-Service-Inspection of Nuclear Power Plant Components" 2001 Edition with 2003 Addenda.

## 3. Applicable Code Requirement

Pursuant to 10 CFR 50.55a(a)(3)(ii) FPL requests an alternative to the requirements of paragraph IWB-3132.2 "Acceptance by Repair/Replacement Activity" that states "A component whose ... examination detects flaws that exceed the acceptance standards ... is unacceptable for continued service ... until the component is corrected by a repair/replacement activity... "

## 4. Reason for Request

This relief request, Fourth Inspection Interval Relief Request No. 6, was previously submitted as the Third Inspection Interval Relief Request No. 26, Reference 5. Two subsequent RAI's were issued and responses were submitted by References 6 and 7. The Relief Request and RAI Responses have been approved by Reference 8.

This relief request, Fourth Inspection Interval Relief Request No. 6, is being submitted to extend approval of the Third Inspection Interval Relief Request No. 26 into the fourth inspection interval.

This relief request, Fourth Inspection Interval Relief Request No. 6, is essentially the same as the Third Inspection Interval Relief Request No. 26. The RAI responses to Relief Request No. 26 have been incorporated and the subject changes have been referenced to the appropriate RAI, References 6 and 7.

Small bore nozzles were welded to the interior of the hot leg of the reactor coolant piping, using partial penetration welds, during fabrication of the piping. Industry experience has shown that cracks may develop in the nozzle base metal or in the weld metal joining the nozzles to the reactor coolant pipe and lead to leakage of the reactor coolant fluid. The cracks are believed to be caused by primary water stress corrosion cracking (PWSCC). The exact leak path, through the weld or through the base metal or through both, cannot be determined. The hardship to remove all possible leak paths requires accessing the internal surface of the reactor coolant piping and grinding out the attachment weld and any remaining nozzle base metal. Such an activity results in high radiation exposure to the personnel involved. Grinding within the pipe also exposes personnel to safety hazards. Additionally, grinding on the internal surface of the reactor coolant piping increases the possibility of introducing foreign material that could damage the fuel cladding. The NRC approved topical report, Reference 3, and the following "Basis for Use" shows that there is "no compensating increase in the level of quality or safety."

## 5. Proposed Alternative and Basis for Use

ALTERNATIVE

The alloy 600 small bore nozzles welded to the interior of the hot leg of the reactor coolant piping, using partial penetration welds, have been replaced using the "half-nozzle" technique. In the "half-nozzle" technique, nozzles are cut outboard of the partial penetration weld, approximately mid-wall of the hot leg piping. The external cut sections of the alloy 600 nozzles are replaced with short sections (half-nozzles) of alloy 690 which are welded to the exterior surface of the pipe. The attachment weld has been relocated from the interior surface of the pipe to the exterior surface of the pipe. The remainder of the alloy 600 nozzles, including the original fabrication partial penetration welds, remain in place.

BASIS FOR USE

A plant-specific evaluation of the small bore nozzles located in the hot leg piping for St. Lucie Unit 1 has been completed. These nozzles are the locations where half nozzles have been utilized, thereby leaving flaws in the original weldments, which could potentially grow into adjacent ferritic material. Postulated flaws were assessed for flaw growth and flaw stability as specified in the ASME Code, Section XI. The results demonstrate compliance with the requirements of the ASME Code, Section XI. The St. Lucie plant specific evaluation, Reference 1, has been submitted to the NRC as Attachments 2 and 3 to Reference 2.

WCAP-15973-P-A Revision 0, Reference 3, evaluates the effect of component corrosion resulting from primary coolant in the crevice region on component integrity and evaluates the effects of propagation of the flaws left in place by fatigue crack growth and stress corrosion cracking mechanisms. In the half-nozzle repair, small gaps of 1/8 inch or less remain between the remnants of the alloy 600 nozzles and the new alloy 690 nozzles. As a result, primary coolant (borated water) will fill the crevice between the nozzle and the wall of the pipe. Low alloy and carbon steels used for reactor coolant systems components are clad with stainless steel to minimize corrosion resulting from exposure to borated primary coolant. Since the crevice regions are not clad, the low alloy and carbon steels are exposed to borated water.

Reference 3 provides bounding analyses for the maximum material degradation estimated to result from corrosion of the carbon or low alloy steel in the crevices between the nozzles and components. Results show that the quantity of material lost does not exceed ASME code limits. The report also provides results of fatigue crack growth evaluations and crack stability analyses for hot leg pipe nozzles. The results indicate that the ASME Code acceptance criteria for crack growth and crack stability are met. Further, available laboratory data and field experience indicate that continued propagation of cracks into the carbon and low alloy steels by a stress corrosion mechanism is unlikely.

The topical report, Reference 3, demonstrates that the carbon and low alloy steel Reactor Coolant System components at St. Lucie Unit 1 will not be unacceptably degraded by general corrosion as a result of the implementation of replacement of small diameter alloy 600 nozzles. Although some minor corrosion may occur in the crevice region of the replaced nozzles, the degradation will not proceed to the point where ASME Code requirements will be exceeded before the end of plant life, including the period of extended operation.

Reference 4 ( NRC letter dated January 12, 2005 Final safety, etc.) stated "The staff has found that WCAP 15972-P, Revision 01, is acceptable for referencing in licensing applications for Combustion Engineering designed pressurized water reactor to the extent specified and under the limitations delineated in the TR (Topical Report) and in the enclosed SE (Safety Evaluation)."

Sections 4.1, 4.2 and 4.3 of the SE present additional conditions to assess the applicability of the topical report. The FPL response for each additional condition is provided below. The FPL response is in *italic font*. The discussion shows that Reference 3 is applicable to St. Lucie Unit 1.

Section 4.1 of the SE states "Licensees seeking to use the methods of the TR will need to perform the following plant-specific calculation in order to confirm that the ferritic portions of the vessels or piping within the scope of the TR will be acceptable for service through the licensed lives of their plants (40 years if the normal licensing basis plant life is used or 60 year is the facility is expected to be approved for extension of the operating license):

1. Calculate the minimum acceptable wall thinning thickness for the ferritic vessel or piping that will adjoin to the MNSA repair or half nozzle repair.

   *FPL Response: limiting diameter of 1.270" identified in Reference 12 of WCAP 15739-P, Rev.01 applicable to St. Lucie Unit 1.*

   *Section 2.4 of WCAP-15973-P applies to St. Lucie Unit 1. The base material of the hot leg, SA-516 Gr. 70, the corrosive environment and operating temperatures of St. Lucie Unit 1 are equivalent to the characteristics described in WCAP-15973-P. The hot leg nozzle design of St. Lucie Unit 1 is equivalent to that described in WCAP-15973-P, as shown in Table 2 herein. Additionally, Section 2.4 of WCAP-15973-P states in several locations that it is applicable to all CE designed plants.*

   *(Preceding paragraph provided in Reference 6, Request Item 4.a)*

2. Calculate the overall general corrosion rate for the ferritic materials based on the calculation methods in the TR, the general corrosion rates listed in the TR for normal operations, startup conditions( including hot standby condition) and cold shutdown conditions and the respective plant-specific times in (in-percentages of total plant life) at each of the operating modes.

   *The overall general corrosion rate was determined using the calculation methods in the TR and St. Lucie Unit 1 generation data from January 1, 1999 to December 31, 2008. The percentage of total plant time spent at each of the temperature conditions follows:*

| | |
|---|---|
| *High temperature conditions* | *93%* |
| *Intermediate temperature conditions* | *2%* |
| *Low temperature conditions* | *5%* |

*The corrosion rate for each temperature condition is taken from the TR and is shown as follows:*

| | |
|---|---|
| *High temperature conditions* | *0.4 mpy* |
| *Intermediate temperature conditions* | *19 mpy* |
| *Low temperature conditions* | *8.0 mpy* |

*The overall corrosion rate was determined using the above time at temperature data and corrosion rate at temperature data and formula 1 of the TR as follows;*

$$CR=0.93 \times 0.4mpy+0.02 \times 19\ mpy+0.05 \times 8mpy$$

*Resulting in an overall corrosion rate of 1.15 mpy*

*(The preceding detailed determination of corrosion rate (methodology) provided in Reference 6, Request Item 4.b. The time at temperature data is current data and therefore it differs slightly from the previous submittal for Third Inspection Interval Relief Request No. 26. )*

3. Track the time at cold shutdown conditions to determine whether this time does not exceed the assumptions made in the analysis. If these assumptions are exceeded, the licensees shall provide a revised analysis to the NRC and provide a discussion on whether volumetric inspection of the area is required.

   *FPL Response: In accordance with section 2.3.4 of the SE, the corrosion rate for CE plants is based on a time split of 88 percent at operating conditions, 2 percent at intermediate temperature startup conditions and 10 percent at low temperature outage conditions. An assessment of operating data for St. Lucie Unit 1 from January 1, 1999 to December 31, 2008 shows 5 percent of plant time at low temperature outage conditions. Therefore, volumetric inspection of the area is not required. In the future, as part of the submittal of this relief request for the fifth inspection interval, the time at cold shutdown conditions will be tracked to determine that the assumptions made in the analysis have not been exceeded.*

   *(Detailed determination of corrosion rate provided in Reference 6, Request Item 4.e and in Reference 7, Request Item 3)*

4. Calculate the amount of general corrosion based thinning for the vessels or piping over the life of the plant, as based on the overall general corrosion rate calculated in Step 2 and the thickness of the ferritic vessel or piping that will adjoin to the MNSA repair or half nozzle repair.

*FPL Response: The first half nozzle repair was made in April 2001. The plant license was renewed and it expires on March 1, 2036. The first half nozzle repair can be expected to see 35 more years of service. Applying the corrosion rate from step 2, 1.15 mils per year, for 35 years results in a material loss of 40.25 mils.*

*(Preceding paragraph provided in Reference 6, Request Item 4.c)*

5. Determine whether the vessel or piping is acceptable over the remaining life of the plant by comparing the worst case remaining wall thickness to the minimum acceptable wall thickness for the vessel or pipe.

   *FPL Response: The calculated material loss in step 4 above was 40.25 mils. Doubling the loss to account for a diametrical change and adding the diameter of 0.997", from Table 2, results in a diameter of 1.08" after 35 years of service. A diameter of 1.08" is less than the limiting diameter of 1.270" identified in Reference 12 of WCAP 15739-P, Rev.01 applicable to St. Lucie Unit 1.*

   *(Preceding paragraph provided in Reference 6, Request Item 4.d)*

Section 4.2 of the SE states "Licensees seeking to reference this TR for future licensing applications need to demonstrate that:

1. The geometry of the leaking penetration is bounded by the corresponding penetration reported in Calculation Report CN-CI-02-71, Revision 01.

   *FPL Response: Calculation Report CN-CI-02-71, Revision 01, Figure 6-1(c) sheets 1 and 2 show the details of the hot leg nozzle that was used for the calculation. A review of drawings of the existing nozzles on the hot leg piping shows that the existing nozzles have essentially the same dimensions as were used in Calculation Report CN-CI-02-71, Rev. 01.*

   *Please see Table 2 which lists the nozzle dimensions used in Calculation Report CN-CI-02-71, Rev. 1 and shows the comparable actual dimensions encountered at St. Lucie Unit 1. Table 2 demonstrates that the alloy 600 small bore nozzles at St. Lucie Unit 1 are bounded by the nozzles used in Calculation Report CN-CI-02-71, Rev. 1.*

| ***TABLE 2 COMPARISON OF NOZZLE DIMENSIONS USED IN CALCULATION AND ACTUAL NOZZLES AT PSL1*** | | | |
|---|---|---|---|
| ***DIMENSION*** | ***VALUE FROM CN-CI-02-71 (in.)*** | ***VALUE AT PSL1 (in.)*** | ***Comparison to VALUE FROM CN-CI-02-71*** |
| *Hot Leg Piping Base Metal Thickness* | *3.75* | *3.75* | *Equivalent* |
| *Hot Leg Piping Inside Radius* | *21* | *21* | *Equivalent* |
| *Cladding Thickness* | *0.25* | *0.25* | *Equivalent* |
| *Nozzle Bore Diameter* | *0.997* | *0.997* | *Equivalent* |
| *Initial Depth of Weld Prep Radial Direction** | *0.938* | *0.938 Flow nozzle*<br>*0.875 RTD nozzle* | *Equivalent*<br>*Less* |
| *Initial Length of Weld Prep Circumferentially** | *0.762* | *0.760 Flow Nozzle*<br>*0.743 RTD Nozzle* | *Less*<br>*Less* |

*Postulated crack dimensions based on initial weld joint dimensions.

*(Table 2 and Evaluation provided in Reference 6, Request Item 5)*

*St. Lucie Unit 1 is bounded by the linear elastic fracture mechanic analysis in Calculation Report CN-CI-02-71, Revision 1, since the estimated Unit 1 hot leg pipe RTNDT is 30 degrees F versus the 60 degree F value used in the TR. The actual RTNDT was not determined for the hot leg piping since this determination was not required at the time of procurement of the piping. However, Charpy V-notch tests were performed on the hot leg piping at +10 degrees F, yielding values of 54, 43, 42 and 55, 44, 52 ft-lb for the two heat numbers involved. Based on Reference 9 herein, Section B.1.1.1(4), the Charpy V-notch test results can be used to justify an estimated RTNDT of the two heat numbers involved as +30 degrees F.*

*(Preceding paragraph provided in Reference 6, Request Item 6.)*

*The response addresses the hot leg piping. The small bore alloy 600 nozzles are welded to straight lengths of hot leg piping. The straight lengths of hot leg piping are made from pieces of plate that are rolled and welded together to form the pipe configuration. The plates were manufactured by Lukens Steel. The material was identified as being from melt number C7293 slab number 65 and melt number C7293 slab number 67.*

*(Preceding paragraph provided in Reference 7, Request Item 1(a))*

*All the straight lengths of hot leg piping at St. Lucie Unit 1 came from the above two heats, melt number C7293 slab number 65 and melt number C7293 slab number 67.*

*(Preceding paragraph provided in Reference 7, Request Item 1(b))*

*Section 3.5 of the TR discusses final crack stability comparisons. The second sentence of the second paragraph states "Also noted in each table was the RTNDT for the bounding cases." The table applicable to the hot leg piping shows RTNDT for the hot leg piping is 60 degrees F. Also Appendix A, page A-5, repeats that the RTNDT for the hot leg piping is 60 degrees F. The bounding value used in the TR is 60 degrees F.*

*(Preceding paragraph provided in Reference 7, Request Item 1(c))*

2. The plant-specific pressure and temperature profiles in the pressurizer water space for the limiting curves (cooldown curves) do not exceed the analyzed profile shown in Figure 6-2 of Calculation Report CN-CI-02-71, Revision 01, as stated in Section 3.2.2 of this SE.

   *FPL Response: As stated in section 6.2.1.1 of Calculation Report CN-CI-02-71, Figure 6-2 of the report applies to the pressurizer. During the 2005 PSL1 refueling outage, the pressurizer was replaced with a new pressurizer, which has new small bore nozzles manufactured from alloy 690. The hot leg piping does not see the transients experienced by the pressurizer. The remainder of the reactor coolant system, including the hot leg, is limited to a 100 °F per hour by Tech Specs. Therefore, the evaluation of the pressurizer limiting curves is considered not applicable.*

3. The plant-specific Charpy USE data shows a USE value of at least 70 ft-lb to bound the USE value used in the analysis. If the plant-specific Charpy USE data does not exist and the licensee plans to use Charpy USE data from other plants' pressurizers and hot leg piping, then justification (e.g., based on statistical or lower bound analysis has to be provided.

   *FPL Response: The Charpy USE data supports an Elastic-Plastic Fracture Mechanics (EPFM) analysis of a pressurizer lower shell axial flaw and not the hot leg piping as described in section 6.3.2.2. of Calculation Report CN-CI-02-71. Therefore, the evaluation of Charpy USE is considered not applicable for nozzle attachments to the hot leg piping.*

Section 4.3 of the SE states "Licensees seeking to implement MNSA repairs or half nozzle replacements may use the WOG's stress corrosion assessment as the bases for concluding that existing flaws in the weld metal will not grow by stress corrosion if they meet the following conditions:

1. Conduct appropriate plant chemistry reviews and demonstrate that a sufficient level of hydrogen overpressure has been implemented for the RCS and that the contaminant concentrations in the reactor coolant have been typically maintained at levels below 10 ppb for dissolved oxygen, 150 ppb for halide ions and 150 ppb for sulfate ions.

*FPL Response: Hydrogen overpressure is typically maintained in the reactor coolant system between 25 and 35 psig. Contaminant concentrations for dissolved oxygen, halide ions and sulfate are maintained at less than 5 ppb. All of these values are steady state values.*

*The reactor coolant system water is analyzed for dissolved oxygen and halides three times per week with no interval between analysis to exceed 72 hours. Analysis for dissolved oxygen is not required when the reactor coolant system Tavg is less than or equal to 250 degrees F. Analysis for halides is not required when all fuel is removed from the reactor vessel and the reactor coolant system Tavg is less than 140 degrees F. The reactor coolant system water is analyzed for sulfate ions at least once per 7 days. The analysis results for the last two cycles were reviewed and no transients were identified.*

*(Previous paragraph provided in Reference 6, Request Item 7)*

2. During the outage in which the half nozzle or MNSA repairs are scheduled to be implemented, licensees adopting the TR's stress corrosion crack growth arguments will need to review their plant specific RCS coolant chemistry histories over the last two operating cycles for their plants and confirm that these conditions have been met over the last two operating cycles.

   *FPL Response: The above contaminant limits have been maintained at steady state operation during the past two cycles.*

This Relief Request applies to all previous repairs to alloy 600 small bore nozzles on the hot leg reactor coolant piping that have left a remnant nozzle in place and all similar future repairs that will leave a remnant nozzle in place.

In conclusion, the ASME Code requirement, IWB-3132.2, is to repair/replace material containing a flaw. The proposed alternative is to not remove the material containing the flaw but show by analysis that the material and the presence of the flaw will not be detrimental to the pressure retaining function of the reactor coolant piping. Analyses, references 1 and 3, have shown that allowing the material containing a flaw to remain in place and in service would not result in a reduction of the level of quality or safety.

## 6. Duration of Proposed Alternative

Relief is requested for the fourth inspection interval for St. Lucie Unit 1.

## 7. References

1) Westinghouse Electric Company LLC Calculation Note Number CN-CI-02-69 Rev. 0 "Evaluation of Fatigue Crack Growth Associated with Small Diameter Nozzles for St. Lucie 1 & 2," dated 10/9/02

2) FPL letter to NRC, Letter L-2002-222, Supplemental Responses to NRC Requests for Additional Information for Review of the St. Lucie Units 1 and 2 License Renewal Application, November 27, 2002

3) WCAP-15973-P-A, Rev 0 (NRC approved version of WCAP-15973-P, Revision 1 with SER and resolved questions) "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Programs," Westinghouse Electric Company LLC, February 2005

4) NRC letter dated January 12, 2005, Subject: Final Safety Evaluation for Topical Report WCAP-15973-P, Rev 01 "Low-Alloy Steel Component Corrosion Analysis Supporting Small-Diameter Alloy 600/690 Nozzle Repair/Replacement Program" (TAC No. MB6805)

5) FPL letter to NRC, Letter L-2005-099, St. Lucie Unit 1 Docket No. 50-335 Inservice Inspection Plan - Third 10 Year Interval Relief Request 26 – Repair of Alloy 600 Small Bore Nozzles Without Flaw Removal, April 29, 2005

6) FPL letter to NRC, Letter L-2005-189, St. Lucie Unit 1 Docket No. 50-335 Inservice Inspection Plan - Third 10 Year Interval Request for Additional Information Relief Request 26 – Repair of Alloy 600 Small Bore Nozzles Without Flaw Removal, August 25, 2005

7) FPL letter to NRC, Letter L-2005-220, St. Lucie Unit 1 Docket No. 50-335 Inservice Inspection Plan - Third 10 Year Interval Second Request for Additional Information Response Relief Request 26 – Repair of Alloy 600 Small Bore Nozzles Without Flaw Removal

8) NRC letter dated November 22, 2005, Subject: St. Lucie Nuclear Plant, Unit 1 – Safety Evaluation for Relief Request No. 26 Regarding Reactor Coolant Piping Hot Leg Alloy – 600 Small Bore Nozzles (TAC No. MC6944)

9) NUREG-0800, U. S. Nuclear Regulatory Commission Standard Review Plan, Branch Technical Position 5-3