Florida Power & Light Company, P. O. Box 14000, Juno Beach, FL 33408-0420

FPL

SEP - 4 2001

L-2001-198
10 CFR 50.4
10 CFR 50.54(f)

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555-0001

RE: St. Lucie Units 1 and 2 and Turkey Point Units 3 and 4
Docket Nos. 50-335, 50-389, 50-250, and 50-251
Response to NRC Bulletin 2001-01

On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." Bulletin 2001-01 requires licensees to submit a written response, within 30 days of the date of the bulletin, that includes the requested information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities, including the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

In accordance with the NRC request, attached is FPL's response to Bulletin 2001-01, for Turkey Point Units 3 and 4 (attachment 1), and St. Lucie Units 1 and 2 (attachment 2). FPL's plans for addressing the bulletin have been developed as an integrated and graded approach to timely action for FPL's four nuclear units which takes into consideration the susceptibility of each of the units to the material cracking phenomenon addressed by the bulletin. The graded approach results in FPL conducting 100% inspections of the vessel head penetrations at each of the units in the order of relative susceptibility. The graded inspection plan calls for either a 100% visual or non-destructive volumetric examination (pending qualification of examination techniques) by spring 2003 for all four units. As described in the attached response, FPL's inspection plan begins with Turkey Point Unit 3 in October 2001 and continues with Turkey Point Unit 4 in spring 2002, St. Lucie Unit 1 in fall 2002, and St. Lucie Unit 2 in spring 2003. Additionally, the FPL unit which is least susceptible to the phenomenon addressed by the bulletin (i.e., St. Lucie Unit 2) has a near term outage to the issuance date of the bulletin (fall 2001). FPL will access a segment of the reactor pressure vessel head through the installed insulation to determine the feasibility of inspecting easily accessible penetrations on the periphery of that reactor pressure vessel head during the fall 2001 outage. This approach provides for a confirmation of the structural integrity of the vessel head penetrations in a timely manner while at the same time keeping personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy.

This response is provided pursuant to the requirements of Section 182a of the Atomic Energy Act of 1954, as amended, and 10 CFR 50.54(f).

Should there be any questions, please call me at (561) 694-4848.

Very truly yours,

R. S. Kundalkar

R. S. Kundalkar
Vice President
Nuclear Engineering

Attachments
cc: Regional Administrator, Region II, USNRC
Senior Resident Inspector, St. Lucie Plant
Senior Resident Inspector, Turkey Point Plant

A088

STATE OF FLORIDA )
) ss.
COUNTY OF PALM BEACH )

R. S. Kundalkar being first duly sworn, deposes and says:

That he is Vice President, Nuclear Engineering, of Florida Power and Light Company, the Licensee herein;

That he has executed the foregoing document; that the statements made in this document are true and correct to the best of his knowledge, information and belief, and that he is authorized to execute the document on behalf of said Licensee.

Rajiv S. Kundalkar

R. S. Kundalkar

Subscribed and sworn to before me this

4th day of September, 2001.

Roberta S. Economy

Name of Notary Public (Type or Print)

Roberta S. Economy
MY COMMISSION # DD047295 EXPIRES
June 1, 2005
BONDED THRU TROY FAIN INSURANCE, INC.

R. S. Kundalkar is personally known to me.

## Turkey Point Units 3 and 4
## Response to NRC Bulletin 2001-01

On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The Bulletin requests licensees to provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities. The requested data includes the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that their plans for future inspections will ensure compliance with applicable regulatory requirements.

Florida Power and Light Company (FPL) has been working with the Electric Power Research Institute (EPRI) Material Reliability Program (MRP) and the Nuclear Energy Institute (NEI) to provide industry information to the NRC concerning the VHP issue, such as the MRP-44 Part 2 Safety Assessments Report[1] referenced in the NRC Bulletin. The EPRI MRP has also prepared an additional report, MRP-48[2], that assembles data from all licensees for reference. MRP-48 was transmitted to the NRC in a letter from A. Marion, NEI to Dr. Brian Sheron, NRC[3] on August 21, 2001. The information in these reports, hereafter referred to as MRP-44, Part 2 and MRP-48, will be referenced as part of this response.

FPL hereby responds to the questions posed in the Bulletin with respect to Turkey Point Units 3 and 4.

***NRC Question 1:*** All *addressees are requested to provide the following information:*

***NRC Question 1a:*** the *plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

**FPL Response to NRC Question 1a:** Turkey Point Unit 3 and 4 have been analyzed for susceptibility relative to Oconee 3 using the time at temperature model and the plant specific input data reported in MRP-48.

This evaluation showed that it would take Turkey Point Units 3 and 4 an additional 6.3 and 6.4 effective full power years (EFPYs) of operation respectively, to reach the same time at temperature as Oconee 3 (ONS 3) at the time that the ONS 3 condition was discovered on March 1, 2001. This ranking puts Turkey Point Units 3 and 4 in the NRC subpopulation of plants having moderate susceptibility to PWSCC or greater than 5 EFPY but less than 30 EFPY from the ONS3 condition.

The different periods of head operating temperatures for Turkey Point Units 3 and 4 in the MRP-48 report reflect upper head temperatures that changed due to flow analyses resulting from steam generator replacements.

***NRC Question 1b:*** *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

**FPL Response to NRC Question 1b:** A description of the Turkey Point Unit 3 and 4 VHP nozzles is included in MRP-48 as well as Table 1 below. All of the 65 four-inch diameter VHPs are of the same size, regardless of their function. The VHPs are in a grid pattern as shown in the MRP-44 Part 2 report (Figure A-2a). The VHPs are on 11.97" centers as identified in MRP-48, which results in a minimum spacing between penetrations of 7.97" when accounting for the penetration radius. The head vent is in the center of a group of 4 VHPs with its center being 8.466" from the center of any of the 4 adjacent VHPs. Accounting for the 2" radius of the VHP and 1.050"/2 radius of the head vent, the minimum spacing for the head vent is 5.94".

**Table 1: Turkey Point VHP Description Details**

| Turkey Point Unit | VHP Type | VHP Qty | VHP Size ID/OD | Min Spacing* | VHP Material Type |
|---|---|---|---|---|---|
| Unit 3 | CRDM/Spares/ Part Length/ Instr Column | 65 | 2.750"/4.000" | 7.97" | SB-167-600 |
| Unit 3 | Vent Line | 1 | ¾" NPS Sch 80 (1.050") | 5.94" | SB-166-600 |
| Unit 4 | CRDM/Spares/ Part Length/ Instr Column | 65 | 2.750"/4.000" | 7.97" | SB-167-600 |
| Unit 4 | Vent Line | 1 | ¾" NPS Sch 80 (1.050") | 5.94" | SB-166-600 |

*** Centerline distance less two VHP radii.**

***NRC Question 1c:*** *a description of the RPV head insulation type and configuration;*

**FPL Response to NRC Question 1c:** The Reactor Vessel Head permanent insulation, i.e., within the Control Rod Drive Mechanism (CRDM) shroud, for Turkey Point Units 3 and 4 consists of blankets fabricated from flexible, high temperature fiberglass encased in a jacket of woven fiber glass fabric. This insulation is secured with velcro fasteners. The insulation is placed in two layers, the bottom layer in one direction with the top placed at right angles. These layers are placed between the rows of CRDMs, with cutouts for the CRDMs. The air gap between the insulation and shroud is maintained to provide the required ventilation for the CRDMs. The materials for the insulation conform to the requirements of NRC Regulatory Guide 1.36.

***NRC Question 1d:*** *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

**FPL Response to NRC Question 1d:** An examination inside the RV head shroud for evidence of leakage (without insulation being removed), is performed every time a unit is taken from Mode 2 to Mode 3 (prior to returning to Mode 2) if an inspection has not been performed in the last 30 days. The acceptance criterion, by procedure is "no observable leakage." This inspection is typically performed by a VT qualified examiner with the limitation that the RV head insulation is in place. There have been no leaks identified in the alloy 600 VHPs to date.

On January 27, 2001 at Turkey Point Unit 4, three periphery CRDM welds on one CRDM, including the bi-metallic alloy 600 weld at the VHP were dye penetrant inspected as part of the Section XI program. This activity also included a VT-1 examination of the general area around this CRDM and was performed with the insulation on the head being noted as a limitation. These exams were performed to written procedures by a qualified PT and VT examiner, and the results were acceptable.

In the last 4 years, examinations of the VHP nozzles and area inside the RV head shroud have been limited to examination for evidence of leakage, and the ASME Section XI Code examination described above. None of these examinations resulted in the insulation being removed to view the bare metal head surface.

***NRC Question 1e:*** *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

**FPL Response to NRC Question 1e**: There are 66 reactor vessel head penetrations for each of the reactor vessels at Turkey Point Units 3 and 4. Of these, 45 are active CRDMs, 2 are used for the reactor vessel level monitoring system (RVLMS), 4 are for the core exit thermocouple (CET) columns, 6 are part length CRDMs with the lead screws retracted and pinned in the housing, 8 are spare CRDM nozzle penetrations with no current use, and there is one head vent penetration. Figure 1 shows a representation of the elevation view depicting the various components above the reactor head.

As shown in Figure 1, each of the CRDMs including the part length CRDMs are supported laterally at the 58'-4 7/8" elevation. The support is provided by capturing the square plates at the top of the control rod position coil stack forming a checker board configuration within the perimeter of the reactor head lifting rig box beam. Using a series of jacking screws along with dummy plates at those locations not containing a CRDM, these plates are captured with a clearance of approximately 1/4" between sides of adjacent plates. This checker board/ lifting rig assembly is then secured to the reactor cavity wall with four equal spaced rigid struts. The spare penetrations are fitted with a short, ventilation flow restrictor box that has no additional lateral restraint. For the CET and RVLMS penetrations, the assemblies terminate just above the nozzles with cable connections.

The cables for the CETs and the RVLMS are gathered, tie-wrapped and secured to the vertical legs of the reactor head lifting rig. The cables continue to the upper area of the reactor head lifting rig, around the inside of the lifting rig box beam and terminate at two adjacent connector panels on the side of the lifting rig box beam. These cables then exit the reactor cavity across two cable trays. For the CRDMs, the cables rise a short distance as shown in Figure 1, are draped over and then tie-wrapped to a series of support elements that are part of the CRDM ventilation duct support. The cables are routed across these support elements to four quadrant points where the cables enter cable trays to exit to the plant.

The reactor vessel missile barrier consists of two rectangular reinforced concrete slabs each 29' x 11' x 2'6" and weighing approximately 120,000 lbs. The two barrier slabs are centered over the reactor head and overlap along their inner edge. A cut out, approximately 8'6" x 6', is formed on each end of the joined slabs to allow passage of the CRDM ventilation ducts. This results in the center 12'6" of the inner 29' edge overlapping. The Updated Final Safety Analysis Report (UFSAR) states that this removable reinforced concrete frame is provided to block any missile that could be generated by a CRDM ejection event.

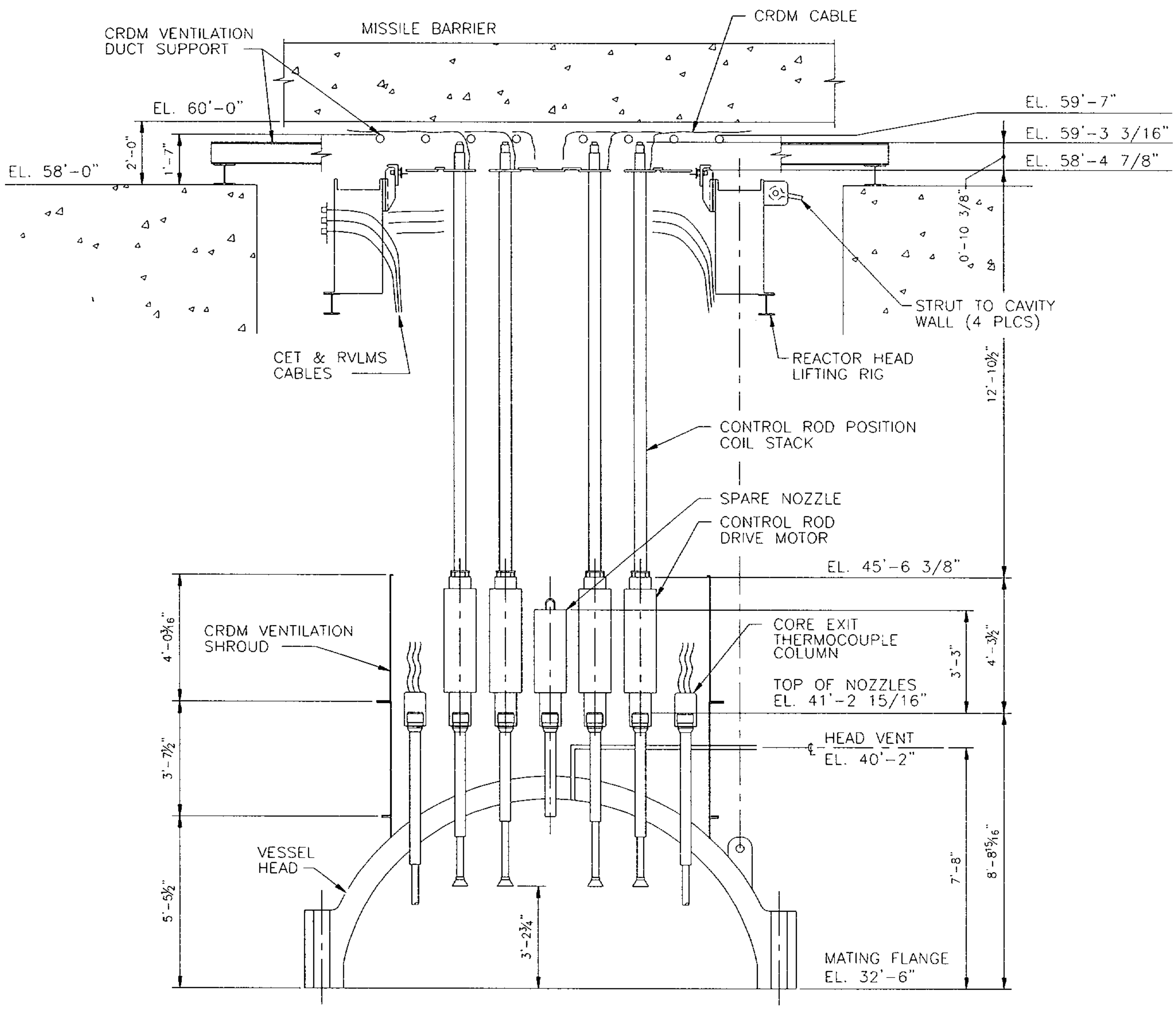



**Figure 1:** Turkey Point Units 3 & 4 Nominal Elevation Schematic of Components Above the Head

**FPL Response to NRC Question 2**: Question 2 is not applicable since FPL has not identified previous VHP cracking or leakage due to PWSCC at Turkey Point Units 3 and 4.

**FPL Response to NRC Question 3**: Question 3 is notapplicable since the susceptibility ranking for Turkey Point Units 3 and 4 are greater than 5 EFPYand less than 30 EFPY of ONS3.

***NRC Question 4**: If the susceptibility rankingfor your plant is greater than 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:*

***NRC Question 4a:** your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

**FPL Response to NRC Question 4a**: FPL is planning a visual inspection (VT-2) of the bare metal on the top of the reactor vessel heads at Turkey Point Units 3 and 4. The blanket insulation will be moved or removed from the RV head as necessary to conduct the examination. The VT-2 examination will be conducted using direct or remote methods including boroscopes and cameras to record the area of interest for any evidence of leakage. The scope is planned for essentially 100% of the surface (as implied in 10 CFR 50.55a; more than 90% of the examination volume of each weld or item, where the reduction in coverage is due to interferences) at the interface between the RV head and the 66 VHPs of both units. The Turkey Point Unit 3 inspection is scheduled for the October 2001 refueling outage. The Turkey Point Unit 4 inspection is scheduled for the spring 2002 refueling outage.

The personnel qualification will be in accordance with the requirements of IWA-2300 of the 1989 ASME Section XI for visual examiners. The acceptance criteria will be no leaks from the VHPs.

Experience from Oconee 1, 2, 3, and ANO-1 inspections indicates that although past leakage (from leaking jointed connections) may result in boric acid residue on the head and insulation, the characteristic of leakage that has clearly initiated at the VHP nozzle is boric acid crystal deposits that appear to have been pushed out of the annulus between the nozzle and the vessel head. This unique condition should be detectable even in the presence of some quantity of boric acid from other sources. Any evidence of boric acid deposits will be documented and evaluated in accordance with the Turkey Point corrective action program.

***NRC Question 4b**: your basis for concluding that the inspections identified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

1) *If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

2) *The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

**FPL Response to NRC Question 4b**:The technical basis for concluding that the regulatory bases are met for Turkey Point Units 3 and 4 is provided in the Regulatory Requirements Section of MRP-48. The following is a supplement to that response with plant specific clarification.

The "Applicable Regulatory Requirements" section of BL 2001-01 list specific General Design Criteria (GDC) of 10 CFR 50 Appendix A applicable to the vessel head penetration nozzle cracking issue. The GDC identified include GDC 14, GDC 31, and GDC 32. Due to the plant vintage, Turkey Point Units 3 and 4 are committed to the 1967 Proposed General Design Criteria as specifically addressed in various sections of the Turkey Point UFSAR. The 1967 Proposed General Design Criteria 9, 34, and 36 contain requirements for reactor coolant pressure boundary, reactor coolant pressure boundary rapid propagation failure prevention and reactor coolant pressure boundary surveillance requirements similar to the current requirements in GDC 14, 31, and 32. Regardless, the requirements established for design, fracture toughness, and inspectability were satisfied during Turkey Point Units 3 and 4 initial licensing review, and will continue to be satisfied during operation by performance of the visual inspections identified in the response to question 4a.

The visual inspections identified in the response to question 4a will meet the requirements of 10CFR50 Appendix B Criterion V and Criterion IX because all examinations will be performed by qualified personnel using qualified procedures in accordance with written acceptance criteria as previously identified in the response to question 4a. Although the subpopulation of plants with greater than 5 EFPYs and less than 30 EFPYs from the ONS3 condition requires only an "effective visual examination," FPL intends to perform examinations that meet the criteria of a "qualified visual examination." FPL has obtained the as-built fabrication records for the Turkey Point Units 3 and 4 VHP interferences. The data was provided as mean fit since multiple readings were taken on each penetration and bore. The mean interference fit for Turkey Point Unit 3 is 0.0001" and 0.0015" for Unit 4. This data is bounded by the 0.0005" to 0.0015" diametral interference identified in the MRP-44 Part 2 Report on the high end, but the Unit 3 data actually has a smaller (looser) interference fit than specified. Field experience at Oconee Units 1, 2, and 3 and ANO-1 with penetrations having an identified interference of 0.0014" (as identified in MRP-2001-050[4,5]) have shown leakage. In addition, the field experience at Bugey 3, with a tighter interference fit[6] of 0.0031"-0.0035" (80μm-90μm) and a through wall PWSCC flaw pressurized to 3000 psi resulted in a visible 1 liter/hour leak[7]. The Bugey experience suggests that even tighter interference fits can result in detectable leakage. The Turkey Point Units 3 and 4 interference fit data is representative of that of other B&W fabricated vessels which have field experience of showing leakage at normal operating pressures when a through wall PWSCC flaw exists. Therefore, based on the field experience to date and the plant specific interference fit data, the Turkey Point Units 3 and 4 visual examinations meet the criteria of a "qualified visual" examination identified in the Bulletin.

The visual inspections identified in the response to question 4a will also meet the requirements of 10CFR50 Appendix B Criterion XVI since these visual inspections are being planned at the next refueling outages. This action is prompt when considering the Turkey Point Units' ranking of susceptibility.

Should leakage be detected during the examinations identified in the response to question 4a, corrective action will be conducted to identify the source. If the leakage is identified as pressure boundary leakage and confirmed to be coming from the VHP annulus region or other component, additional inspection techniques would be used to locate and characterize the flaw. NDE methods would likely include eddy current, ultrasonic, dye penetrant or a combination of these examination methods. Following flaw characterization, the flaw would be removed and/or repaired with an ASME Code or NRC approved method. This corrective action would occur prior to returning to a mode of operation in which the Technical Specification 3/4.4.6.2 requirement for pressure boundary leakage is applicable.

***NRC Question 5**: Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

***NRC Question 5a:** a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

***NRC Question 5b:**if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

**FPL Response to NRC Question 5**: FPL will provide the requested information within 30 days after plant restart following the next refueling outage.

## Turkey Point Units 3 and 4 References

[1] "PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations," Electric Power Research Institute (EPRI), Palo Alto, CA: 2001. TP-1001491, Part 2, Interim Report dated May 2001.

[2] "PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," Electric Power Research Institute (EPRI), Palo Alto, CA: 2001. TP-1006284, dated August 2001.

[3] Nuclear Energy Institute (NEI) Letter, "Generic Information for Use by Licensees in Response to NRC Bulletin 2001-01, Project Number: 689," From A. Marion, NEI to Dr. Brian Sheron, NRC, August 21, 2001.

[4] "Response to NRC Review Comments Transmitted by letter Dated June 22, 2001 to the NEI Relating to MRP-48, Part 2," Electric Power Research Institute (EPRI) MRP Report MRP-2001-050, dated June 29, 2001.

[5] Nuclear Energy Institute (NEI) Letter, "Response to June 22, 2001, letter from Dr. Brian Sheron, (NRC) to Mr. Alex Marion (NEI) transmitting NRC staff questions on EPRI Interim Report TP-1001491, Part 2 (MRP-48)," From A. Marion to Dr. Brian Sheron, NRC, June 29, 2001.

[6] "Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs," Electric Power Research Institute (EPRI), Palo Alto, CA: 1993. TR-103345, p.E1-5.

[7] "Proceedings: 1992 EPRI Workshop on PWSCC of Alloy 600 in PWRs," Electric Power Research Institute (EPRI), Palo Alto, CA: 1993. TR-103345, p.B1-2.

## St. Lucie Units 1 and 2
## Response to NRC Bulletin 2001-01

On August 3, 2001, the NRC issued Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." The Bulletin requests licensees to provide information related to the structural integrity of the reactor pressure vessel head penetration (VHP) nozzles for their respective facilities. The requested data includes the extent of VHP nozzle leakage and cracking that has been found to date, the inspections and repairs that have been undertaken to satisfy applicable regulatory requirements, and the basis for concluding that licensee's plans for future inspections will ensure compliance with applicable regulatory requirements.

Florida Power and Light Company (FPL) has been working with the Electric Power Research Institute (EPRI) Material Reliability Program (MRP) and the Nuclear Energy Institute (NEI) to provide industry information to the NRC concerning the VHP issue, such as the MRP-44 Part 2 Safety Assessments Report [1] referenced in the Bulletin. The EPRI MRP has also prepared an additional report, MRP-48 [2], which assembles data from all licensees for reference. MRP-48 was transmitted to the NRC in a letter from A. Marion, NEI to Dr. Brian Sheron, NRC [3] on August 21, 2001. The information in these reports, hereafter referred to as MRP-44, Part 2 and MRP-48, will be referenced as part of this response.

FPL hereby responds to the questions posed in the Bulletin with respect to St. Lucie Units 1 and 2.

***NRC Question 1:*** *All addressees are requested to provide the following information:*

***NRC Question 1a:*** *the plant-specific susceptibility ranking for your plant(s) (including all data used to determine each ranking) using the PWSCC susceptibility model described in Appendix B to the MRP-44, Part 2, report;*

**FPL Response to NRC Question 1a:** St. Lucie Units 1 and 2 have been ranked for the potential for primary water stress corrosion cracking (PWSCC) of the reactor pressure vessel (RPV) top head nozzles using the time at temperature model and plant-specific input data reported in MRP-48 (The time at temperature model is the same as described in MRP-44). As shown in Table 2-1 of MRP-48, this evaluation indicates that it will take St. Lucie Units 1 and 2, 10.3 and 11.3 effective full power years (EFPYs) respectively, of additional operation from March 1, 2001, to reach the same time at temperature that Oconee Nuclear Station Unit 3 (ONS3) had at the time that its leaking nozzles were discovered in February 2001.

Using the criteria stated in the NRC Bulletin 2001-01, St. Lucie Units 1 and 2 fall into the NRC subpopulation of plants having moderate susceptibility to PWSCC of greater than 5 EFPYs but less than 30 EFPYs until reaching the ONS3 time at temperature.

The periods of different head operating temperatures for St. Lucie Unit 1 in the Table 2-2 of MRP-48 reflect upper head temperatures that changed due to flow analyses resulting from uprating the unit early in cycle 5 and steam generator replacement in 1998.

***NRC Question 1b:*** *a description of the VHP nozzles in your plant(s), including the number, type, inside and outside diameter, materials of construction, and the minimum distance between VHP nozzles;*

**FPL Response to NRC Question 1b:** St. Lucie Unit 1 has 78 VHPs. A description, quantity, inside and outside diameter, and minimum centerline spacing (except for the head vent) of all the VHP nozzles is included in MRP-48, Table 2-3. The material type for each penetration is provided in Table 1 below. The VHPs are mostly in a grid pattern as shown in the MRP-44 Part 2 report (Figure A-6a). The head vent is in the center of two control element drive mechanism (CEDM) penetrations, which are 16.360 inches apart, making the vent center 8.18 inches from the center of an adjacent CEDM. (CEDM is the Combustion Engineering NSSS terminology for control rod drive mechanism or CRDM). For consistency, the term CRDM will be used hereafter in this response to be consistent with the terminology used in the Bulletin). Accounting for the radius of the CRDM (3.850 inches/2) and radius of the head vent (1.050 inches/2), the minimum spacing for the head vent is 5.73 inches.

St. Lucie Unit 2 has 102 VHPs. A description, quantity, inside and outside diameter, and minimum centerline spacing (except for the head vent) of all the VHP nozzles is included in MRP-48, Table 2-3. The material type for each penetration is provided in Table 1 below. The VHPs are in a grid pattern as shown in the MRP-44 Part 2 report (Figure A-7a). The head vent is in the same grid as the CRDMs with a centerline spacing of 11.57 inches. Therefore, due to the larger diameter of the CRDMs the minimum spacing between the head vent is greater than that of two adjacent CRDMs.

**Table 1: St. Lucie VHP Description Details**

| St. Lucie Unit | VHP Type | VHP Qty | VHP Material Type |
|---|---|---|---|
| **Unit 1** | CRDM / Part Length | 69 | SB-167-600 |
| **Unit 1** | ICI Instr Column | 8 | SB-167-600 |
| **Unit 1** | Vent Line | 1 | SB-167-600 |
| **Unit 2** | CRDM | 91 | SB-166-600 |
| **Unit 2** | ICI Instr Column | 10 | SB-167-600 |
| **Unit 2** | Vent Line | 1 | SB-167-600 |

***NRC Question 1c:*** *a description of the RPV head insulation type and configuration;*

**FPL Response to NRC Question 1c:** The insulation type at St. Lucie Unit 1 is metal reflective encapsulated mineral wool. The insulation conforms to the head as do the lower CRDM pressure housings. The insulation is made up of 33 panels, 21 of which encircle the 69 CRDM nozzles. The encircling panels were installed over the lower CRDM pressure housing flanges during initial construction. The gap between the insulation panels and the VHPs are filled with plug rings (containing asbestos insulation), that were installed prior to the panels and secured with a metal band. Figure 1 shows the general arrangement of the insulation.

The insulation type at St. Lucie Unit 2 is metal reflective encapsulated fibrous "cerablanket" material. The insulation conforms to the head and is made up of 17 panels which encircle the 101 VHP nozzles (excluding the vent line). The encircling panels were installed over the lower CRDM pressure housing flange during initial construction. Figure 2 shows the general arrangement of the insulation. The gap between the insulation panels and the VHPs are filled with metal and blanket insulation plug rings that were installed prior to the panels. The rings are secured with a metal band and have bottom and top metal end cap "flanges" that cover the hole in each panel. The bottom end cap flange is tack welded to the plug ring and is larger than the hole in the panel such that it cannot be removed without first removing the panel. Figure 3 shows the insulation plug ring as it was installed during construction.

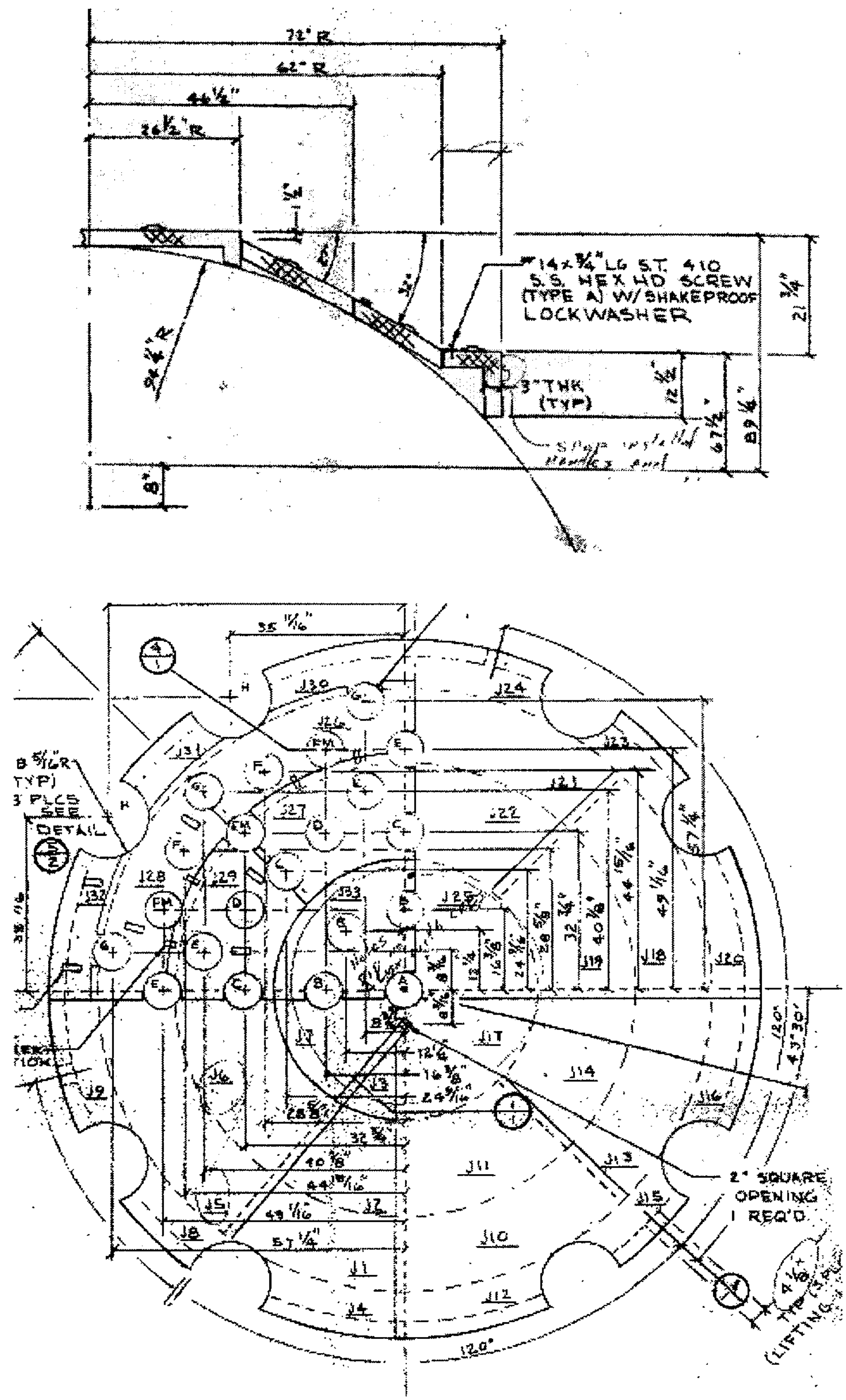
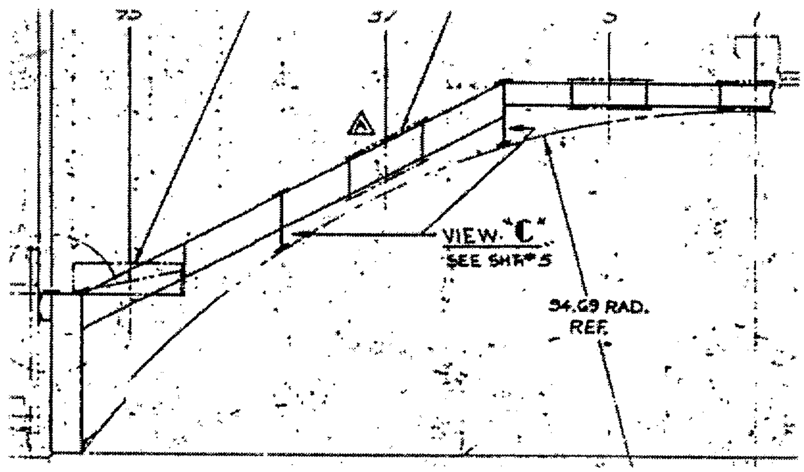



**Figure 1: Plan and Cross Sectional View of the St. Lucie Unit 1 RPV Head Insulation**



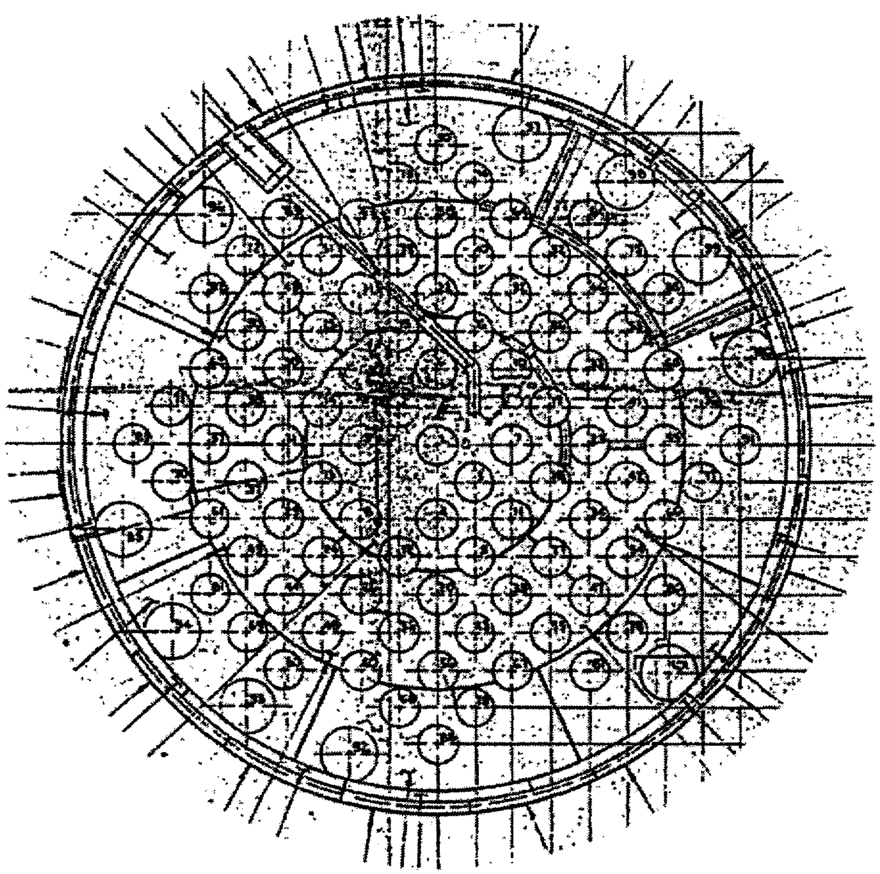



**Figure 2: Plan and Cross Sectional View of the St. Lucie Unit 2 RPV Head Insulation**

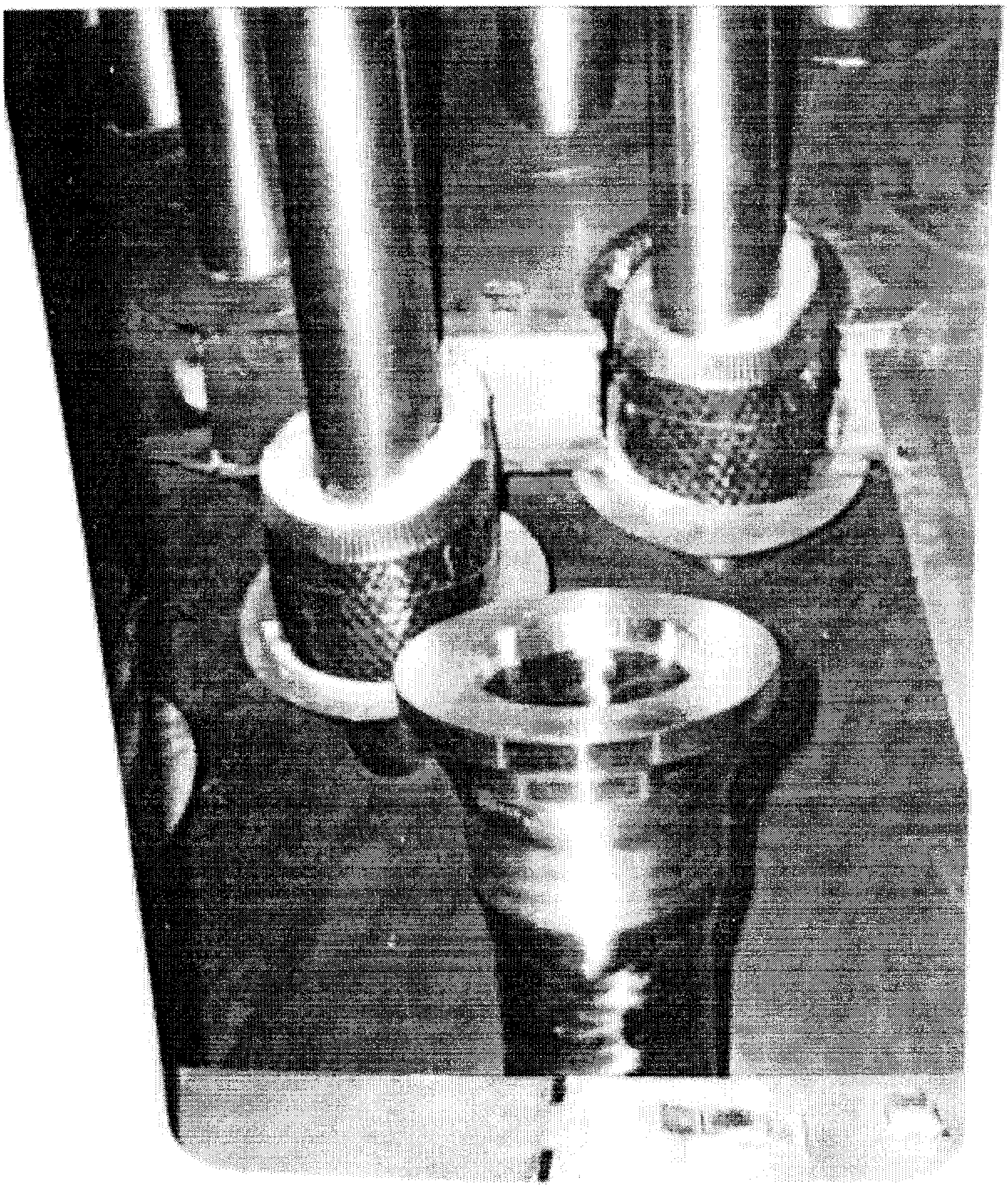


**Figure 3: St. Lucie Unit 2 RPV Head Insulation Initial Installation Showing Plug Rings with Lower End Cap**

***NRC Question 1d:*** *a description of the VHP nozzle and RPV head inspections (type, scope, qualification requirements, and acceptance criteria) that have been performed at your plant(s) in the past 4 years, and the findings. Include a description of any limitations (insulation or other impediments) to accessibility of the bare metal of the RPV head for visual examinations;*

**FPL Response to NRC Question 1d:** During the April 2001 refueling outage at St. Lucie Unit 1, a walkdown of the RPV upper head area was performed in preparation for an upcoming unrelated Section XI exam. During that walkdown, it was noted that the bare metal RPV head surface was accessible at two CRDM penetrations due to the absence of the 2 insulation plug rings that normally fill the gap between the panels and the CRDM penetrations. This access was photographed and documented in a plant condition report. The limitations were that the 3 inch thick surrounding insulation panel had an 8 inch access hole to view the 2 VHPs and the RPV base metal. No indication of leakage was visible.

All other examinations of the VHP nozzles and RPV head general area in the last 4 years have been limited to implementation of the reactor coolant system leak test procedures at St. Lucie Units 1 and 2. These leakage examinations are performed to written procedures that require examinations on every cooldown for refueling and all heatups as well as ASME Code required system pressure tests. These examinations do not require insulation removal. All evidence of leakage is documented for disposition by procedure.

***NRC Question 1e:*** *a description of the configuration of the missile shield, the CRDM housings and their support/restraint system, and all components, structures, and cabling from the top of the RPV head up to the missile shield. Include the elevations of these items relative to the bottom of the missile shield.*

**FPL Response to NRC Question 1e:**

<u>**Configuration of the Missile Shield:**</u>
**St. Lucie Unit 1** - The missile shield is a rectangular concrete slab with overall dimensions of 27 feet x 31 feet-7inches x 3 feet thick. It is fabricated in three 9 feet x 31 feet-7 inches x 3 feet sections and is steel lined on the bottom and sides. Each section weighs approximately 140,000 lbs. The missile shield is located above the CRDMs. The missile shield combined with the primary and secondary shield walls protect the containment vessel, reactor coolant system and parts of the main steam system from missiles generated from the pressure retaining components. FSAR Table 3.5-1 identifies the missile shield to be the barrier against the following internal missiles: reactor vessel closure head nut, reactor vessel closure head nut and stud, instrumentation assembly, instrumentation from flange up, instrument flange stud, and control rod drive assembly.

**St. Lucie Unit 2** - The Unit 2 missile shield has the same configuration as Unit 1. FSAR Table 3.5-4 identifies the missile shield to be the barrier against the following internal missiles: reactor vessel closure head nut, reactor vessel closure head nut and stud, incore detector instrumentation assembly, and control rod drive assembly.

<u>**CRDM Housing and Support/ Restraint System:**</u>
**St. Lucie Unit 1** - There are 78 VHPs in the reactor closure head as previously identified, of which 69 penetrations are attached to CRDMs (61 active CRDMs plus 8 that were previously attached to part-length control element assemblies or CEAs). The 69 CRDMs are restrained laterally by the reactor head cooling shroud orifice plate and reactor head lifting rig assembly. The cooling shroud orifice plate is 1.75 inches thick and is located just above the CRDMs at elevation 48 feet 1-5/16

inches.

**St. Lucie Unit 2** - There are 102 CEA nozzle penetrations in the reactor closure head as previously identified, of which 91 VHPs are used for CRDMs. The CRDMs are laterally restrained within the plenum orifice plate and the reactor head lift rig assembly. The plenum orifice plate is 2 inches thick. The plate is located just above the reactor head insulation at 44 feet 2-3/16 inches.

**RPV Head Details, Spacing & Height Relative to Missile Shield:**

**St. Lucie Unit 1** - The bottom of the missile shield is at elevation 67.0 feet. The flange face of the RPV head, or RPVH is at elevation 36.0 feet. The separation from the reactor head flange face to the bottom of the missile shield is 31.0 feet. There are 69 CRDM nozzle penetrations, 8 ICI instrument nozzles, and one head vent penetration. The relative height of each is shown in the attached Unit 1, Figure 4.

**St. Lucie Unit 2** - The bottom of the missile shield is at elevation 68.0 feet. The flange face of the RPVH is at elevation 36.0 feet. The separation from the reactor head flange face to the bottom of the missile shield is 32.0 feet. There are 91 CRDM nozzle penetrations, 10 instrument nozzles, and one head vent penetration. The relative height of each is shown in the attached Unit 2, Figure 5.

**RPVH Vent Pipe Assembly, Elevations and Restraints:**

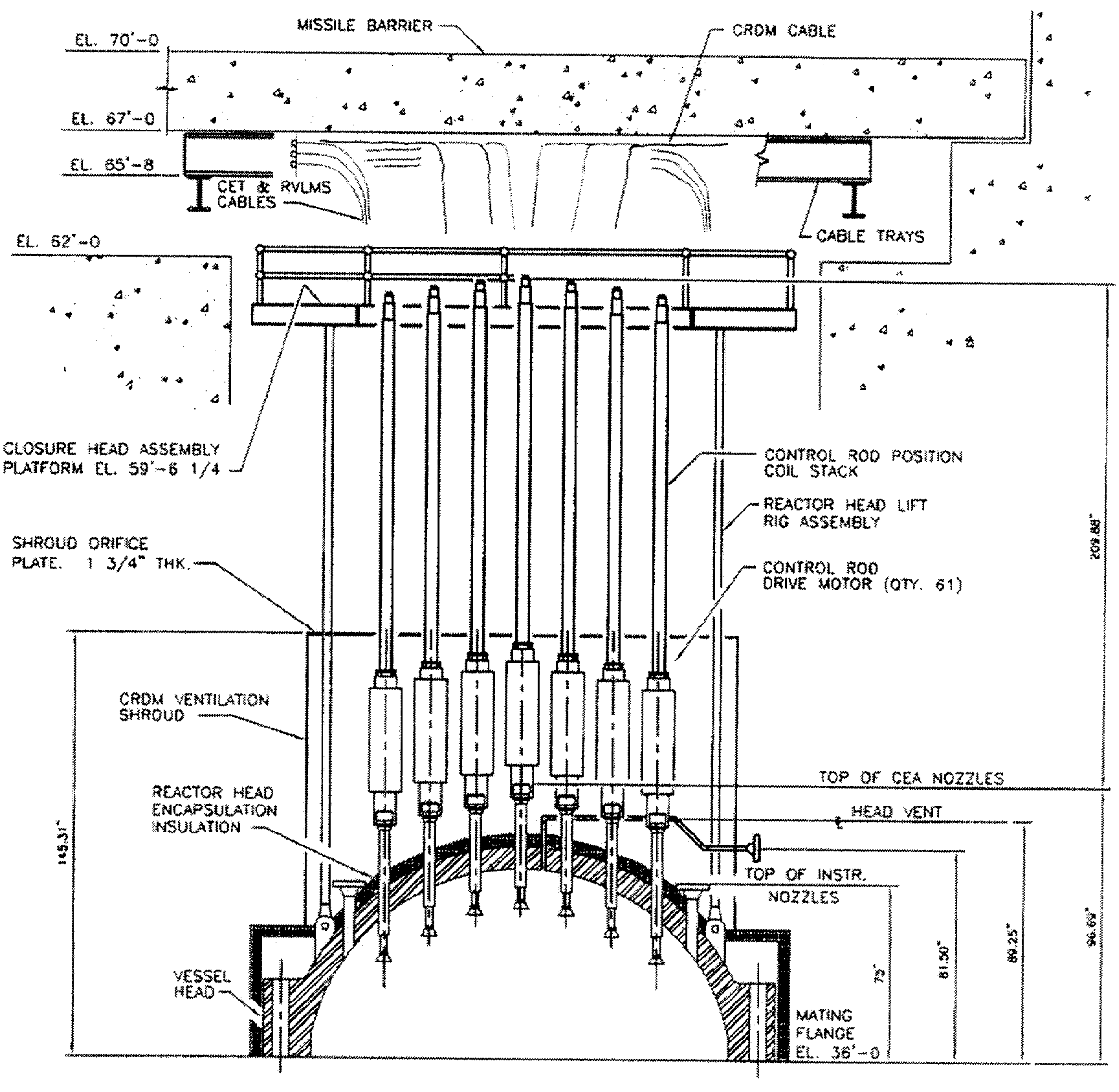
**St. Lucie Unit 1** - The reactor vessel head vent is a ¾ inch, schedule 160, stainless steel pipe that is attached to the alloy 600 head vent nozzle penetration. The vent extends from the reactor head penetration outward approximately 75 inches through the head ventilation shroud to a flanged connection and the remainder of the head vent system. A rigid pipe support is located above the reactor head to provide a restraining function in the horizontal plane. Additional rigid supports, located downstream of the flange connection, also provide restraining functions in all three translational and rotational directions.

**St. Lucie Unit 2** - The reactor vessel head vent is a ¾ inch, schedule 160, stainless steel pipe that is attached to the alloy 600 head vent nozzle penetration. The vent extends from the reactor head penetration outward approximately 96-5/8 inches through the head ventilation shroud to a flanged connection and the remainder of the head vent system. A rigid restraint is located above the reactor head to provide restraining functions in both the horizontal and vertical planes. Additional rigid supports, located downstream of the flange connection, provide restraining functions in all three translational and rotational directions.

**Cabling:**

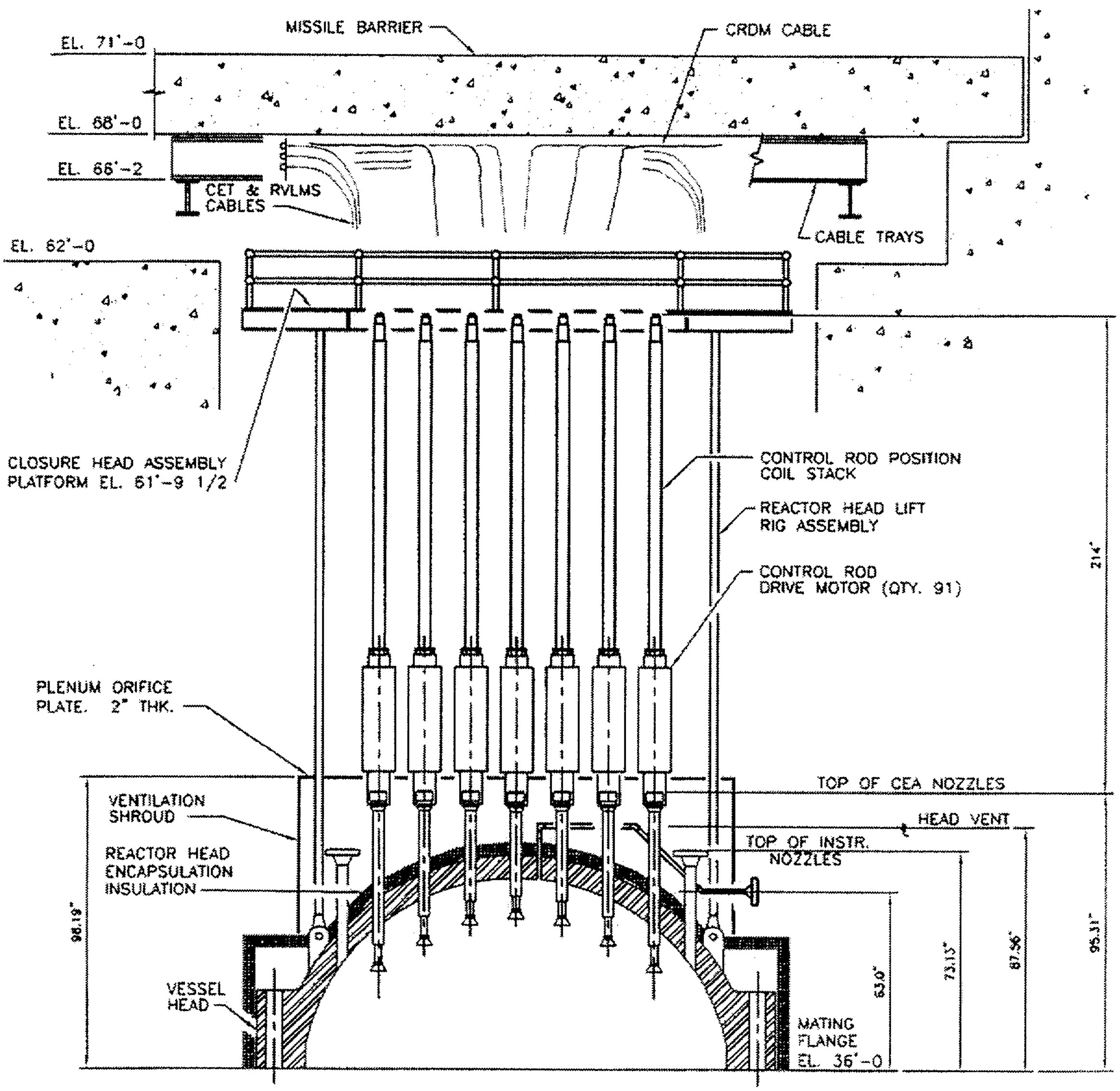
**St. Lucie Unit 1** - The cabling between the reactor head and the reactor missile shield includes the CRDM power and position cables, and the head cable assemblies for the core exit thermocouples (CET), heated junction thermocouples (HJTC), and self-powered nuclear detectors (SPND). The CRDM power and reed switch position cables are jacketed, high temperature, radiation resistant cables. The head cables are OEM provided, mineral insulated (MI), steel jacketed cables. All the cables are fastened and routed through channels to disconnect panels and then to the cable trays located just below reactor missile shield.

**St. Lucie Unit 2** – The configuration of the Unit 2 cables is equivalent to the Unit 1 cabling.




St. Lucie Unit 1

**Figure 4:** St. Lucie Unit 1 Nominal Elevation Schematic of RPVH Components and The Missile Shield Above the Head (not to scale)




St Lucie Unit 2

**Figure 5:** St. Lucie Unit 2 Nominal Elevation Schematic of RPVH Components and The Missile Shield Above the Head (not to scale)

**FPL Response to NRC Question 2:** Question 2 is not applicable since FPL has not identified previous VHP cracking or leakage due to PWSCC at St. Lucie Units 1 and 2.

**FPL Response to NRC Question 3:** Question 3 is not applicable since the susceptibility ranking for St. Lucie Units 1 and 2 are greater than 5 EFPY and less than 30 EFPY of ONS3.

***NRC Question 4:*** *If the susceptibility ranking for your plant is greaterthan 5 EFPY and less than 30 EFPY of ONS3, addressees are requested to provide the following information:*

***NRC Question 4a:*** *your plans for future inspections (type, scope, qualification requirements, and acceptance criteria) and the schedule;*

**FPL Response to NRC Questions 4a**: For St. Lucie Unit 1, FPL is planning a visual inspection (VT-2) of the bare metal on the top of the reactor vessel head during the fall 2002 refueling outage (SL1-18). A visual examination (VT-2) will be conducted using direct or remote methods including boroscopes and cameras to record the area of interest for any evidence of leakage. The scope is planned for essentially 100% of the surface (as implied in 10 CFR 50.55a; more than 90% of the examination volume of each weld or item, where the reduction in coverage is due to interferences) at the interface between the RPV head and the 78 VHPs of St. Lucie Unit 1.

For St. Lucie Unit 2, FPL is planning to perform an essentially 100% examination under the RPV head of the 102 VHPs during the spring 2003 refueling outage (SL2-14). Examinations under the RPV head may be performed with surface, near surface, volumetric or a combination of NDE techniques. FPL will work with industry groups and vendors to develop inspection equipment and methodology including qualification and acceptance criteria. The actual NDE methods will be determined based on equipment availability, capability and qualification prior to the SL2-14 outage. This examination will provide meaningful inspection results and a more comprehensive assessment of the structural integrity of the VHPs as requested in the Bulletin.

In the near term, FPL will perform a partial visual examination (VT-2) of the bare metal on the top of the St. Lucie Unit 2 reactor vessel head during the fall 2001 refueling outage (SL2-13). This visual examination will be performed by accessing a segment of the RPV head through the installed insulation to determine the feasibility of inspecting easily accessible penetrations on the periphery of the RPV head during the fall 2001 outage. This effort is intended to view as many VHPs as practical when access is gained under the insulation by removing one panel or obtaining access under or through the shroud. This effort will allow FPL to: properly assess the feasibility of removing the tight fitting restrictive insulation; confirm the initial assessment that access to perform an under the insulation inspection is limited without insulation removal; and to verify the large man-hour effort assumed in the dose estimate of approximately 64 Rem to remove the insulation. The spring 2003 (SL2-14) inspection schedule will allow adequate time to plan the activities associated with the under the RPV head examination of the VHPs and keep personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy.

FPL has determined removal of the St. Lucie Unit 2 restrictive metal insulation (described in the response to question 1c), without adequate time for planning would result in a large impact of dose and outage schedule. MRP-2001-050 [4] reported dose and schedule impacts to perform a bare metal visual inspection for a "typical plant" in response to the NRC questions in Reference 1 as 6 Rem and 2 days duration with no schedule impact. For St. Lucie Unit 2, the dose for this destructive insulation removal is estimated at approximately 64 Rem. The actual insulation removal duration from a plant with similar restrictive insulation that performed this removal in 1989 (at a plant with 25% less VHPs) was approximately 16 days, not including the documentation effort of a VHP examination. The St. Lucie Unit 2 insulation removal has been estimated at approximately 2000 person hours. With adequate planning for this mostly manual effort, the dose and schedule impact to destructively remove the insulation is expected to be reduced somewhat. However, a more quantifiable dose reduction would be achieved by using a largely automated examination technique under the RPV. Actual dose to perform an automated eddy current exam (ECT) inside the VHPs from under the RPV head have been approximately 5 Rem or less. Although the exam area of interest identified in the Bulletin (inside, outside and weld surface area of the VHPs) is larger than those exams performed in the 1990s, the dose would not be expected to increase significantly for this largely automated exam. The dose reduction for performing an automated under the RPV head inspection with a qualified NDE technique could be as much as 60 Rem, or approximately a 90% reduction. Therefore, with a properly planned and implemented automated examination of the VHPs, personnel exposures will be kept as low as reasonably achievable, which is consistent with the NRC ALARA policy.

Qualification test blocks and requirements are being developed by industry and EPRI-MRP for ECT and ultrasonic testing (UT) methods. These test blocks will be used to verify that techniques are qualified, and the abilities and limitations are known and understood before they are used in the field. These qualifications are expected to be similar to those demonstrated for ECT and UT of the ID surface of the VHPs for the industry integrated examination program provided in response to GL97-01. However, this qualification program will not be available in time to perform under the RPV head examinations at the moderately susceptible St. Lucie Unit 2 in the fall of 2001. In addition, the limited equipment capability that currently exists is directed toward those plants that have demonstrated the existence of PWSCC in their VHPs or plants within the high susceptibility subpopulation (less than 5 EFPYs from the ONS3 condition).

This approach of a partial inspection in the fall of 2001 and essentially 100% in the spring of 2003 for St. Lucie Unit 2 is prompt because this plant has 11.3 EFPYs of margin before reaching the time at ONS3 and is in the moderate susceptibility subpopulation. Also, because of the timing of the Bulletin, and some plants operating on 24 month refueling cycles, four plants in the moderate susceptibility category as identified in table 2-1 of MRP48, with 10.2, 10.8, 14.5 and 16.1 EFPYs from ONS3, will be performing their first inspections in 2003.

Therefore, performing an essentially 100% examination of the St. Lucie Unit 2 VHPs during the spring 2003 refueling at the same schedule as other similarly ranked plants, will not result in an increase in risk, since St. Lucie Unit 2 will be performing a more comprehensive assessment of the integrity of its VHPs within the same time as other similarly ranked units covered by the Bulletin.

Finally, FPL's plans for addressing the inspection requirements of the bulletin have been developed as an integrated and graded approach to timely action for FPL's four nuclear units which takes into consideration the susceptibility of each of the units to the material cracking phenomenon addressed by the bulletin. The graded approach results in FPL conducting 100% inspections of the vessel head penetrations at each of the units in the order of relative susceptibility. The graded inspection plan calls for either a 100% visual or non-destructive volumetric examination (pending qualification of examination techniques) by spring 2003 for all four units, which is within the same time frame as other similarly ranked plants first refueling outage schedule identified in MRP-48. Additionally, the FPL unit which is least susceptible to the phenomenon addressed by the bulletin (i.e., St. Lucie Unit 2) has a near term outage to the issuance date of the bulletin (fall 2001). FPL will access a segment of the reactor pressure vessel head through the installed insulation to determine the feasibility of inspecting penetrations easily accessible on the periphery of that reactor pressure vessel head during the fall 2001 outage. This approach provides for a confirmation of the structural integrity of the vessel head penetrations in a timely manner while, at the same time keeping personnel exposures as low as reasonably achievable (ALARA), consistent with the NRC ALARA policy.

The safety assessment has also been addressed by MRP–44 (Reference 1 and 3) and provides the basis that there is no significant near-term impact on plant safety in the presence of potential CRDM nozzle PWSCC. The main points supporting this are:

The three Oconee plants and ANO-1 are among the lead units in the United States from the standpoint of operating time and vessel head temperature (see Section 4 of MRP-44).

Several other plants with long operating times and high head temperatures have already performed inspections of 1) the top surface of their vessel heads for leaks, or 2) the inside surfaces of the nozzles near the welds for cracks. In addition to the B&W units, top of the head inspections have been performed at eight plants subsequent to the leakage being discovered at Oconee 1. Individuals performing these inspections have been advised of the need to detect small amounts of leakage. There have been no significant findings in any of these inspections (see Section 4 of MRP-44).

The CE Owners Group was asked to evaluate the need to address operator actions and training for scenarios involving rod ejection(s), small-, medium- and large-break LOCAs and rod-insertion failure(s). The results of this evaluation can be summarized as follows:

LOCA(s) resulting from head penetration failure(s) would be bounded by existing design basis analyses and therefore the core would remain covered by borated water, which would provide adequate cooling. Core internals would remain in a coolable geometry, and the condition of the containment following the scenario(s) would not require implementation of the severe accident management guidelines (SAMGs).

Existing EOPs provide guidance for the full range of LOCAs and include coverage for multiple events including reactivity excursions that might occur during the course of an accident. Existing guidelines provide adequate directions to mitigate the transient induced by one or more CRDM penetration failures.

Existing guidelines also cover the reactivity insertion event during periods of highest rod worth, resulting in the same conclusion.

The personnel qualification for visual examiners will be in accordance with the requirements of IWA-2300 of the 1989 ASME Section XI. The acceptance criteria will be no leaks from the VHPs. Any indication of leakage will be evaluated using pictures available from the Oconee 1, 2, 3 and ANO-1 inspections. Experience from Oconee 1, 2, 3, and ANO-1 indicate that although past leakage (from leaking jointed connections) may result in boric acid residue on the head and insulation, the characteristic of leakage that has clearly initiated at the VHP nozzle is boric acid crystal deposits that appear to have been pushed out of the annulus between the nozzle and the vessel head. The unique appearance of boric acid that has been pushed out of the annulus between the nozzle and the vessel head should be detectable even in the presence of some quantity of boric acid from other sources. Any evidence of boric acid deposits will be documented and evaluated in accordance with our corrective action program.

***NRC Question 4b:*** *your basis for concluding that the inspectionsidentified in 4.a will assure that regulatory requirements are met (see Applicable Regulatory Requirements section). Include the following specific information in this discussion:*

1) *If your future inspection plans do not include a qualified visual examination at the next scheduled refueling outage, provide your basis for concluding that the regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.*

2) *The corrective actions that will be taken, including alternative inspection methods (for example, volumetric examination), if leakage is detected.*

**FPL Response to NRC Questions 4b**: The technical basis for concluding that the regulatory bases are met for St. Lucie Units 1 and 2 are provided in the Regulatory Requirements Section of MRP-48. The following is a supplement to that response with plant specific clarification.

The visual and under the RPV head inspections identified in the response to question 4a will meet the requirements of 10CFR50 Appendix B Criterion V and Criterion IX because all examinations will be performed to written procedures by qualified personnel using qualified procedures in accordance with written acceptance criteria as previously identified in the response to question 4a.

The visual and under the RPV head inspections identified in the response to question 4a will also meet the requirements of 10CFR50 Appendix B Criterion XVI since these visual inspections are being planned at near term refueling outages that allow sufficient time for proper planning so that personal exposure is kept low consistent with the NRC ALARA Policy. This action is prompt when considering the St. Lucie Units ranking of susceptibility and the discussion provided in the response to question 4a.

Should leakage be detected during the examinations identified in the response to question 4a, corrective action will be conducted to identify the source. If the leakage is identified as pressure boundary leakage and confirmed to be coming from the VHP annulus region or other component, additional inspection techniques would be used to locate and characterize the flaw. NDE methods would likely include eddy current, ultrasonic, dye penetrant, or a combination of these examination methods. Following flaw characterization, the flaw would be removed and/or repaired using an ASME Code or NRC approved method. This corrective action would occur prior to returning to a mode of operation that the St. Lucie Units 1 and 2 Technical Specification 3/4.4.6.2 requirement for pressure boundary leakage is applicable.

***NRC Question 5:*** *Addressees are requested to provide the following information within 30 days after plant restart following the next refueling outage:*

***NRC Question 5a:*** *a description of the extent of VHP nozzle leakage and cracking detected at your plant, including the number, location, size, and nature of each crack detected;*

***NRC Question 5b:*** *if cracking is identified, a description of the inspections (type, scope, qualification requirements, and acceptance criteria), repairs, and other corrective actions you have taken to satisfy applicable regulatory requirements. This information is requested only if there are any changes from prior information submitted in accordance with this bulletin.*

**FPL Response to NRC Questions 5a and b:** FPL will provide the requested information from St. Lucie Unit 1 within 30 days after plant restart following the next refueling outage (SL1-18).

FPL will also provide the scope and results of the partial visual inspection from St. Lucie Unit 2 within 30 days after plant restart following the fall 2001 refueling outage (SL2-13) and the results of the essentially 100% under the head inspection within 30 days after plant restart following the spring 2003 refueling outage (SL2-14).

## St. Lucie Units 1 and 2 References

[1] "PWR Materials Reliability Program Interim Alloy 600 Safety Assessments for US PWR Plants (MRP-44): Part 2: Reactor Vessel Top Head Penetrations", Electric Power Research Institute (EPRI), Palo Alto, CA: 2001. TP-1001491, Part 2, Interim Report dated May 2001.

[2] "PWR Materials Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," Electric Power Research Institute (EPRI), Palo Alto, CA: 2001. TP-1006284, dated August 2001.

[3] Nuclear Energy Institute (NEI) Letter, "Generic Information for Use by Licensees in Response to NRC Bulletin 2001-01, Project Number: 689," From A. Marion, NEI to Dr. Brian Sheron, NRC, August 21, 2001.

[4] "Response to NRC Review Comments Transmitted by letter Dated June 22, 2001 to the NEI Relating to MRP-48, Part 2," Electric Power Research Institute (EPRI) MRP Report MRP-2001-050, dated June 29, 2001.