Florida Power
A Progress Energy Company

Crystal River Nuclear Plant
Docket No. 50-302
Operating License No. DPR-72

Ref: 10 CFR 50.54(f)

March 28, 2002
3F0302-11

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

Subject: Crystal River Unit 3 - Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

References:

1. FPC to NRC Letter, 3F1101-04, dated November 19, 2001, "Information Requested in Item 5 of NRC Bulletin 2001-01, Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"
2. FPC to NRC Letter, 3F1101-03, dated November 28, 2001, "Licensee Event Report (LER) 50-302/01-004-00"
3. FPC to NRC Letter, 3F0801-06, dated August 30, 2001, Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles"

Dear Sir:

Pursuant to 10 CFR 50.54(f), Florida Power Corporation (FPC) is hereby providing the information requested in NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," for Crystal River Unit 3 (CR-3). Attachment A provides the CR-3 15-day response to the information required by Items 1.A through 1.E of the Bulletin. CR-3 has concluded that the existing inspection and maintenance programs will ensure continued compliance with applicable regulatory requirements. Attachment B contains a list of regulatory commitments contained in this submittal. Responses to Items 2 and 3 of the Bulletin will be addressed by separate submittals.

CR-3 experienced degradation of one Control Rod Drive Mechanism (CRDM) nozzle which was detected and repaired during refueling outage 12R (Fall 2001). The details of this activity were addressed in References 1 and 2 above. In addition to the corrective actions to repair the CRDM nozzle discussed in References 1 and 2, FPC is planning to replace the CR-3 reactor vessel head in refueling outage 13R, scheduled for Fall 2003. The replacement reactor vessel head is being designed to minimize the concerns for CRDM cracking and leakage.

Although not required by CR-3 procedures, this Bulletin response has been reviewed and approved by the Plant Nuclear Safety Committee.

If you have any questions regarding this submittal, please contact Mr. Sid Powell, Supervisor, Licensing and Regulatory Programs at (352) 563-4883.

Sincerely,

Dale E. Young
Vice President, Crystal River Nuclear Plant

DEY/pei

Attachments:

A. Crystal River Unit 3 Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

B. List of Regulatory Commitments

xc: NRR Project Manager
Regional Administrator, Region II
Senior Resident Inspector

STATE OF FLORIDA

COUNTY OF CITRUS

Dale E. Young states that he is the Vice President, Crystal River Nuclear Plant for Progress Energy; that he is authorized on the part of said company to sign and file with the Nuclear Regulatory Commission the information attached hereto; and that all such statements made and matters set forth therein are true and correct to the best of his knowledge, information, and belief.

____

Dale E. Young
Vice President
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this _____ day of ________________, 2002, by Dale E. Young.

____

Signature of Notary Public
State of Florida

____

(Print, type, or stamp Commissioned
Name of Notary Public)

Personally Known _______ -OR- Produced Identification _______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

ATTACHMENT A

Crystal River Unit 3
Response to NRC Bulletin 2002-01
“Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity”

NRC Bulletin 2002-01 stated that the following information was required to be submitted within 15 days of the date of the Bulletin:

1.A a summary of the reactor pressure vessel head inspection and maintenance programs that have been implemented at your plant,

Response to 1.A

On March 17, 1988, the Nuclear Regulatory Commission issued Generic Letter (GL) 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." GL 88-05 dealt with concerns regarding the possible corrosive degradation of carbon steel parts due to the leakage of borated water and the resulting loss of the reactor coolant pressure boundary. In response to GL 88-05, Florida Power Corporation developed a program to identify and correct any leakage of borated water and the resulting corrosion damage. Preventative Maintenance procedure, PM-168, "Boric Acid Corrosion Inspection And Evaluation Program," was developed to implement GL 88-05. This program was fully implemented for the first time in 1990 during refueling outage 7R. The program includes extensive walkdowns for all components located inside the Reactor Building containing primary reactor coolant.

GL 97-01, "Degradation of CRDM/CEDM Nozzle and Other Vessel Closure Head Penetrations," raised additional concerns about boric acid corrosion specific to the Reactor Vessel Head (RVH). Crystal River Unit 3 (CR-3) responded that these concerns would be addressed by the visual inspections performed in accordance with the GL 88-05 program.

Further industry experience indicated that additional measures were required to address phenomena related to boric acid corrosion. In 2001, the NRC issued NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles." In response to this Bulletin (Reference 3 to this letter), CR-3 stated that an effective bare metal visual inspection of all Control Rod Drive Mechanism (CRDM) nozzles would be performed (this inspection met the criteria for a qualified visual inspection as defined by the Bulletin). For any CRDM nozzle identified as potentially leaking, confirmatory ultrasonic or penetrant testing would be performed. All confirmed leaking nozzles would be repaired during refueling outage 12R. The results of these inspection and repair processes was detailed in References 1 and 2 to this letter.

The configuration of the CR-3 reactor head and Control Rod Drive Service Structure (CRDSS) provide two types of openings that can be used for leakage inspection. There are nine twelve-inch diameter round inspection ports located around the perimeter of the CRDSS that allow for direct visual inspection of the top of the RVH. The inspection ports have doors that are normally bolted in the closed position and are normally covered in mirror insulation. These inspection ports were installed in 1990 to facilitate the GL 88-05 inspection program. There are also small (approximately 4 5/8(W) x 7(H) inches) holes around the perimeter of the CRDSS called "mouse holes." The mouse holes were cut or ground out of the cylindrical section of the lower CRDSS skirt after the skirt was welded to the reactor closure head.

During refueling outage 10R (February 1996), CR-3 performed a 100% bare metal detailed inspection (i.e., through the access ports). Boric acid residue was noted on the RVH and was attributed to leaking CRDM flanges. Each CRDM nozzle ends in a flanged connection that is bolted to the CRDM motor tube assembly. The flanges are located above the RVH and are part of the RCS pressure boundary. During 10R, CR-3 repaired the CRDM flanges and installed new hardware to prevent future leakage. CR-3 also cleaned the RVH during 10R.

During refueling outage 11R (October 1999), CR-3 performed a visual inspection through the mouse holes for evidence of CRDM flange leakage. No leakage was observed.

During refueling outage 12R (October 2001), a 100% bare metal visual inspection of the RVH was performed both before and after the RVH was cleaned. The inspection was performed in accordance with plant procedures SPS VT-N14 Revision 5, "Visual Examination of System Pressure Testing ASME Code Section XI," and SPS VA-N11, Revision 4, "Visual Acceptance of System Pressure Testing ASME Section XI." The inspectors were VT-2 qualified with special qualifications and training related to CRDM nozzle leakage observation and boric acid detection. Inspections conducted in 12R indicated that no active flange leaks were present. One leaking CRDM nozzle was found and repaired.

1.B an evaluation of the ability of your inspection and maintenance programs to identify degradation of the reactor pressure vessel head including, thinning, pitting, or other forms of degradation such as the degradation of the reactor pressure vessel head observed at Davis Besse,

Response to 1.B

The visual examination programs utilized at CR-3 have shown that through-wall leakage in CRDM nozzles is readily detectable via boric acid crystal accumulation at the nozzle to head interface. The locations of leaking nozzles can be identified via visual bare metal inspections of the outside surface of the closure head. The Non-Destructive Examination (NDE) techniques used to characterize the nozzle flaws, as discussed below, would readily detect the extent, if any, of void formation, thinning, or any other form of wastage in the carbon steel.

This experience is supported by an Engineering calculation that has been performed using plant specific as-built CRDM nozzle interference fit data which concluded that through-wall pressure boundary leakage will produce visible boric acid crystal deposits on top of the RVH. The 100% bare metal visual examinations are performed of the RVH by personnel specifically trained and qualified for identifying and characterizing boric acid deposits and localized corrosion. If boric acid deposits are detected, the source of the leakage is located. Boric acid deposits are then removed by cleaning and a second visual examination is performed looking for evidence of corrosion. Pitting or material wastage due to the boric acid crystal deposits would be detected during the post-cleaning examination. The post-cleaning examination provides a baseline for future examinations.

Automated ultrasonic examinations of the nozzles using Framatome's Top-Down-tooling are capable of detecting voids in the head at the nozzle outside diameter (OD) by analysis of the zero degree backwall data. During the Framatome CRDM repair process, the grinding sequence removes the cracked portions of the nozzle, exposing the head material inside the nozzle bore. Visual inspection and liquid penetrant examination of the exposed nozzle bore act as a further, and more direct confirmation that no laminations or voids exist in the carbon steel in that area. Finally, head thickness UT measurements taken from the top of the head adjacent to the nozzle are also capable of detecting deeper voids and laminations, as well as thinning of the head material.

The worst-case corrosive environment for RVH degradation would be repetitive cycles of wetting/drying and replenishing of boric acid from reactor coolant in direct contact with carbon steel. This mechanism could lead to degradation of the carbon steel of the RVH. One potential source of boric acid would be reactor coolant leakage from CRDM flanges onto the RVH. Based on the 12R inspections, this potential source of boric acid leakage is not present. Generalized surface degradation has not been observed on the RVH at CR-3.

Another potential source of boric acid would be CRDM nozzles exhibiting active leakage via through-wall cracking. Experience at CR-3 and throughout the industry has shown that leakage through CRDM nozzle cracks would result in boric acid crystals being present on the head surrounding the leaking nozzle before significant head corrosion occurs. Therefore, if no degradation is detected by the visual and NDE methods at leaking nozzles, it can be concluded that no void formation is occurring at non-leaking nozzles.

1.C a description of any conditions identified (chemical deposits, head degradation) through the inspection and maintenance programs described in 1.A that could have led to degradation and the corrective actions taken to address such conditions,

Response to 1.C

Video inspections of the CRDM nozzles prior to 10R have historically shown buildup of boric acid deposits on some nozzles. The deposits generally take the form of streaked residue on the flange, nut ring and nozzle, occasionally extending down the nozzle to the reactor head. The primary corrective action has been to replace the flange gasket to eliminate the source of leakage. Boric acid deposits were removed only to the extent necessary to disassemble the flange and replace the gasket. The video inspection in 12R showed no indications of active flange leakage.

The 100% bare metal visual inspection of the head performed as part of the boric acid inspection program during 10R identified boric acid deposits on the reactor head. Loose deposits were removed and no RVH surface degradation was detected.

The 100% as-found bare metal inspection of the head performed on October 1, 2001, identified streaks of boric acid characteristic of leakage from sources above the head, and "popcorn" boric acid deposits at nozzle #32 characteristic of through-wall CRDM nozzle leakage. The streaked deposits had a "baked" appearance and were attributed to old CRDM flange leaks. No fresh deposits indicative of active flange leaks were identified. The deposits at nozzle #32, shown in the photograph below, were similar in nature to those previously observed at leaking CRDM nozzle locations at the Oconee and Arkansas Nuclear One (Unit 1) plants.

Boron Deposits at CR-3 RVP Nozzle #32 During Refuel 12

Following the as-found bare metal inspection, the head was cleaned to remove all boric acid deposits, after which the head was subjected to a second 100% visual inspection for evidence of corrosion. No pitting or wastage of the carbon steel was detected.

In response to the discovery of "popcorn" deposits around nozzle #32, the presence of through-wall nozzle cracking was confirmed to be circumferential in nature through ultrasonic testing (UT) of the portion of the nozzle extending from the inside surface of the head to the outside surface. The nozzle was repaired by removing the lower (cracked) portion of the nozzle and welding the remainder of the nozzle directly to the carbon steel nozzle bore. As part of the repair process, penetrant testing (PT) was performed on the exposed portion of the carbon steel bore, additional UT and PT were performed following the nozzle repair, and UT was performed to measure the head thickness local to nozzle #32.

The PT results indicated no evidence of voids or laminations in the carbon steel bore. Circumferential nozzle UT results showed zero degree backwall data that confirmed that there were no voids in the nozzle, weld or head material. The UT thickness measurement showed that actual head thickness exceeded nominal, and showed no evidence of thinning or internal voids. This collective NDE data demonstrates conclusively that the carbon steel base metal in the vicinity of nozzle #32 was unaffected by through-wall leakage in the Alloy 600 nozzle material.

1.D your schedule, plans, and basis for future inspections of the reactor pressure vessel head and penetration nozzles. This should include the inspection method(s), scope, frequency, qualification requirements, and acceptance criteria,

Response to 1.D

CR-3 is planning to replace the RVH in refueling outage 13 (R13), scheduled for Fall 2003. The new RVH service structure is being designed to facilitate periodic visual inspections of the CRDM nozzles and the RVH material. Industry guidance is being developed for the inspection criteria for replacement RVHs. CR-3 will utilize this guidance in the development of the RVH inspection schedule. Inspections will continue to be performed in accordance with the GL 88-05 Boric Acid Corrosion Inspection and Evaluation program.

1.E your conclusion regarding whether there is reasonable assurance that regulatory requirements are currently being met (see the Applicable Regulatory Requirements, above). This discussion should also explain your basis for concluding that the inspections discussed in response to item 1.D will provide reasonable assurance that these regulatory requirements will continue to be met. Include the following specific information in this discussion:

(1) If your evaluation does not support the conclusion that there is reasonable assurance that regulatory requirements are being met, discuss your plans for plant shutdown and inspection.

(2) If your evaluation supports the conclusion that there is reasonable assurance that regulatory requirements are being met, provide your basis for concluding that all regulatory requirements discussed in the Applicable Regulatory Requirements section will continue to be met until the inspections are performed.

Response to 1.E

DESCRIPTION OF REGULATORY REQUIREMENTS

The general design criteria (GDC) as stated in the bulletin, for nuclear power plants (Appendix A to 10 CFR 50), came into effect after the licensing of CR-3. CR-3 has been designed and constructed taking into consideration the proposed 10 CFR 50.34 Appendix A, "General

Design Criteria for Nuclear Power Plant Construction Permits," as published in the Federal Register (32FR10213) on July 11, 1967, and which are applicable to this unit. The GDC in 10 CFR 50, Appendix A and the corresponding CR-3 criteria are provided for comparison.

10 CFR 50, Appendix A
Criterion 14 – Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.

Corresponding CR-3 Criterion 9 – Reactor Coolant Pressure Boundary (Category A)

The reactor coolant pressure boundary shall be designed and constructed so as to have an exceedingly low probability of gross rupture or significant leakage throughout its design lifetime.

Discussion

The Reactor Coolant (RC) system pressure boundary at CR-3 meets the criterion through the following:

a. Material selection, design, fabrication, inspection, testing, and certification in accordance with ASME and USA Standards (USAS) codes.
b. Manufacture and construction in accordance with approved procedures.
c. Inspection in accordance with ASME and USAS code requirements plus additional requirements imposed by the manufacturer.
d. System analysis to account for cyclic effects of thermal transients, mechanical shock, seismic loadings, and vibration loadings.
e. Selection of reactor vessel material properties to give due consideration to neutron flux effects and the resultant increase of the Nil-Ductility Transition Temperature (NDTT).
f. Quality Assurance program described in Sections 1.6 and 1.7 of the CR-3 Final Safety Analysis Report.
g. Advances in the field of fracture mechanics have been used to analytically demonstrate that large bore pipe, such as that used in the reactor coolant loop, will not rupture catastrophically.

The original materials and methods of construction are not changed or altered as a result of the potential for CRDM nozzle cracking and RVH carbon steel wastage. No carbon steel degradation or wastage was observed at the site of the leaking CRDM nozzle or on any other portion of the CR-3 RVH. The CRDM nozzle materials are flaw tolerant and will exhibit degradation through small leakages visible on the RVH and not through sudden rupture or catastrophic failure. The small amount of observed leakage from the degraded CR-3 CRDM nozzle demonstrates that leakage from RVH penetration cracks can be detected before they lead to gross failure or result in wastage of RVH carbon steel. The leaking CRDM nozzle was repaired and determined to be acceptable. As part of that determination, an Engineering Calculation was generated to show that the portion of the carbon steel nozzle bore exposed to the reactor coolant as a result of the repair process would be subject to corrosion at a rate conservatively estimated at 0.0030 inch/year. This figure accounts for a 24-month operating cycle plus a 2-month refueling cycle. The low corrosion rate is attributed primarily to the lack of an aerated environment within the reactor during power operation. This corrosion rate is supported by industry operating experience and corresponds to a wastage depth of approximately 1/8" over 40-years. Therefore, CR-3 concludes that the GDC continues to be met.

10 CFR 50, Appendix A
Criterion 31 – Fracture Prevention of Reactor Coolant Pressure Boundary

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a non-brittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Corresponding CR-3 Criterion 34 – Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (Category A)

The reactor coolant pressure boundary shall be designed to minimize the probability of rapidly propagating type failures. Consideration shall be given (a) to the notch-toughness properties of materials extending to the upper shelf of the Charpy transition curve, (b) to the state of stress of materials under static and transient loadings, (c) to the quality control specified for materials and component fabrication to limit flaw sizes, and (d) to the provisions for control over service temperature and irradiation effects which may require operational restrictions.

Discussion

The reactor coolant pressure boundary design meets this criterion by the following:

a. Development of reactor vessel plate material properties opposite the core to a specified Charpy-V-notch test result of 30 ft/lb or greater at a nominal low nil ductility transition temperature (NDTT).
b. Determination of the fatigue usage factor resulting from expected static and transient loading during detailed design and stress analysis.
c. Quality Control procedures including permanent identification of materials and non-destructive testing.

d. Operating restrictions to prevent failure towards the end of design vessel life resulting from increase in the NDTT due to neutron irradiation, as predicted by a material irradiation surveillance program (CR-3 FSAR Section 4.4.5).
e. Surveillance capsules for BWOG Material surveillance are located in CR-3 reactor vessel providing representative samples of actual fluence induced degradation.

The CR-3 RVH carbon steel has not experienced degradation that could impact flaw propagation.

10 CFR 50, Appendix A
Criterion 32 – Inspection of Reactor Coolant Pressure Boundary

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

Corresponding CR-3 Criterion 36 – Reactor Coolant Pressure Boundary Surveillance (Category A)

Reactor coolant pressure boundary components shall have provisions for inspection, testing, and surveillance by appropriate means to assess the structural and leaktight integrity of the boundary components during their service lifetime. For the reactor vessel, a material surveillance program conforming to ASTM-E-185-66 shall be provided.

Discussion

The reactor coolant pressure boundary components at CR-3 meet this criterion. Access is provided for non-destructive examination during plant shutdown. A reactor pressure vessel material surveillance program conforming to this criterion has been established as described in CR-3 FSAR Section 4.4.5. The present reactor vessel surveillance program is described in B&WOG Topical Report BAW-1543.

As described above, the requirements established for design, fracture toughness, and inspectability in GDC 14, 31, and 32 were satisfied during the initial licensing review of CR-3, and continue to be satisfied during operation, even in the presence of the potential for primary water stress corrosion cracking (PWSCC) of the CRDM nozzle penetrations of the RVH. In part, the selection of Alloy 600 materials provide excellent corrosion resistance and extremely high fracture toughness of the reactor coolant pressure boundary. CR-3, in the original design of the CRDSS, had the capability to perform required ASME Code visual examinations. The CR-3 CRDSS has been modified to provide additional access to the bare metal interface of the CRDM nozzle and the RVH to improve examination capabilities during ASME Code required visual examinations.

10CFR50.55a Codes and Standards

ASME Class I components (which include RVH and CRDM nozzles) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Table IWB-2500-1 of Section XI provides examination requirements for CRDM nozzles and references IWB-3522 for acceptance standards.

<u>Discussion</u>

CR-3 has performed inspections of the RVH during previous refueling outages using both direct observation and indirect observation for leakage. The direct inspection is conducted through the access openings in the CRDSS and is a bare metal inspection. The indirect inspection is performed through the observation of evidence of leakage; i.e., signs of boric acid accumulation. These visual inspections meet the requirements of Section XI Table IWB-2500-1 and IWB-3522. The visual inspections also meet the requirements of NRC Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." Compliance with the requirements of Section XI is implemented through the CR-3 Inservice Inspection Program. If the VT-2 examinations detect the conditions described in IWB-3522.1, as not meeting the acceptance of IWB-3142, then the corrective actions required would be performed in accordance with IWA-5250 (Corrective Measures) and the CR-3 Corrective Action Program. During 12R, one CRDM nozzle was identified and confirmed as leaking from the visual inspections of the RVH. The CRDM nozzle was repaired prior to restart from the refueling outage. No degradation of the RVH carbon steel was identified.

CR-3 Improved Technical Specifications (ITS)

CR-3 ITS 3.4.12, "RCS Operational LEAKAGE," LCO 3.4.12a states, "RCS operational LEAKAGE shall be limited to: No pressure boundary LEAKAGE."

<u>Discussion</u>

Identification of RCS pressure boundary (RCPB) leakage would require the plant to shutdown. With the plant in the shutdown condition, leakage should reduce due to the lower pressure. When the unidentified plant leakage approaches the plant administrative limits, appropriate actions will be taken to identify leakage sources to ensure that continued degradation of the RCPB does not continue.

Additionally, monitoring and various leakage detection systems are available that provide diverse methods of detection to the plant operator to ensure appropriate corrective actions are taken in accordance with ITS. Visual inspections conducted during refueling outages provide the opportunity to access areas/components within the plant that are normally not accessible during plant operations.

Performance of a 100% visual inspection during 12R identified leakage from one CRDM nozzle. The leaking nozzle was repaired as described earlier. All repairs were completed prior to return to service. NDT was performed on a sample of other CRDM nozzles to verify that the degradation was not widespread. This event was reported to the NRC per 10 CFR 50.73 in Reference 2 to this letter.

10 CFR 50, Appendix B "Quality Assurance Requirements"

Criterion IX – Control of Special Processes

Measures shall be established to assure that special processes, including welding, heat-treating, and nondestructive testing, are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria and other special requirements.

Discussion

Activities related to inspection and repair of the CRDM nozzles are controlled as required by the FPC Quality Assurance Program for CR-3. Personnel, processes and procedures will be used as required. The visual inspections of the CRDM nozzles and RVH were conducted by qualified inspectors using approved procedures. The inspectors were specifically qualified for CRDM nozzle leakage observations. Additional processes and procedures required for nondestructive examination (NDE) and other repair activities such as machining and welding are controlled in accordance with the QA program.

Criterion V – Instructions, Procedures, and Drawings

Activities affecting quality shall be prescribed by documented instructions, procedures, or drawings of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings. Instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished.

Discussion

Activities for visual inspection of the RVH, NDE and repair of CRDM nozzles are performed in accordance with the Florida Power QA Program for CR-3. The procedures, instructions and drawings are subject to preparation, review and approval requirements imposed through the QA Program. The QA Program meets the requirements of 10 CFR 50 Appendix B.

Criterion XVI – Corrective Actions

Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective actions taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken shall be documented and reported to the appropriate levels of management.

Discussion

The identification and confirmation of a leaking CRDM nozzle required that the issue be appropriately identified and entered into the CR-3 Corrective Action Program (CAP) in a timely manner. In the case of a significant adverse condition, the CAP requires determination of the cause of the failure and assignment of appropriate corrective actions to preclude recurrence. The CAP implemented at CR-3 meets the requirements of 10 CFR 50 Appendix B, Criterion XVI. The CAP effectively addressed the CRDM nozzle leak and repair in 12R. The existing repaired CRDM nozzle continues to meet all applicable criteria. The corrective action to prevent recurrence is the replacement of the RVH, scheduled for 13R.

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50 - 302 / LICENSE NUMBER DPR - 72

Attachment B

List of Regulatory Commitments

List of Regulatory Commitments

The following table identifies those actions committed to by Florida Power Corporation in this document. Any other actions discussed in the submittal represent intended or planned actions by Florida Power Corporation. They are described to the NRC for the NRC's information and are not regulatory commitments. Please notify the Supervisor, Licensing and Regulatory Programs, of any questions regarding this document or any associated regulatory commitments.

ID Number	Commitment	Commitment Date
3F0302-11-01	CR-3 will replace the reactor vessel head during refueling outage 13R.	Refueling outage 13R currently scheduled for Fall 2003.