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)

May 15, 2002
3F0502-01

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

Subject: Crystal River Unit 3 – 60-Day Response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Reference: FPC to NRC letter, 3F0302-11, dated, March 28 2002, "Crystal River Unit 3 – Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Dear Sir:

Pursuant to 10 CFR 50.54(f), Florida Power Corporation (FPC) hereby submits the Crystal River Unit 3 (CR-3) 60-Day response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." The CR-3 response to the information required in Items 1.A through 1.E of the Bulletin regarding the Reactor Vessel Head was provided to the NRC in the referenced letter. Item 3.A of the Bulletin requires that the following information related to the remainder of the reactor coolant pressure boundary be submitted to the NRC within 60 days of the date of the bulletin:

3.A. The basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

The attachment to this letter provides the information requested in Item 3.A. above. FPC concludes in the attachment that the CR-3 Boric Acid Corrosion, Inspection and Evaluation Program is in compliance with the applicable regulatory requirements discussed in GL 88-05 and NRC Bulletin 2002-01. Additionally, the program incorporates plant and industry operating experience. The program will continue being evaluated and enhanced, as needed, incorporating industry experience and best practices.

This letter establishes no new regulatory commitments.

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,

D.L. Roderick
Director Site Operations

DLR/lvc

Attachment: Response to Item 3.A. of NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation And Reactor Coolant Pressure Boundary Integrity"

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

STATE OF FLORIDA

COUNTY OF CITRUS

D.L. Roderick states that he is the Director Site Operations, 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.

D.L. Roderick
Director Site Operations
Crystal River Nuclear Plant

The foregoing document was acknowledged before me this 15th day of May, 2002, by D.L. Roderick.

Lisa A Morris

Signature of Notary Public
State of Florida

LISA A. MORRIS
Notary Public, State of Florida
My Comm. Exp. Oct. 25, 2003
Comm. No. CC 879691

LISA A MORRIS

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

Personally Known X -OR- Produced Identification______

FLORIDA POWER CORPORATION

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT

Response to Item 3.A. of NRC Bulletin 2002-01,
“Reactor Pressure Vessel Head Degradation And
Reactor Coolant Pressure Boundary Integrity”

NRC Bulletin 2002-01 requires that the following information related to the reactor coolant pressure boundary be submitted to the NRC within 60 days of the date of the bulletin:

3.A. The basis for concluding that your boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and this bulletin. If a documented basis does not exist, provide your plans, if any, for a review of your programs.

Response:

Crystal River Unit 3 (CR-3) Boric Acid Corrosion, Inspection and Evaluation Program is in compliance with the applicable regulatory requirements discussed in Generic Letter (GL) 88-05 and Bulletin 2002-01. Information regarding how CR-3 meets those requirements is described under the "Applicable Regulatory Requirements" section of this attachment.

The discussion below provides background information regarding the development of the CR-3 Boric Acid Corrosion, Inspection and Evaluation Program. The discussion also provides program definition, scope, responsibility, frequency and training. Additionally, the information addresses programmatic response to leakage, and discusses the incorporation of plant and industry operating experience (OE) into the program.

The program has assured that no significant wastage has ocurred as a resut of boric acid corrosion. The program will continue being evaluated and enhanced, as needed, incorporating industry experience and best practices.

Discussion

Prior to the NRC issuing GL-88-05, CR-3 had established a program to identify and correct borated water leakage from the reactor coolant pressure boundary (RCPB) and any resulting corrosion damage. Preventative Maintenance Procedure PM-168, titled, "Visual Observation Check for Boron Corrosion on Threaded and Flanged Connections on the High Pressure Primary Side," was implemented to establish a comprehensive boric acid corrosion inspection program.

On March 17, 1988, the NRC issued Generic Letter (GL) 88-05 titled, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants." This generic letter dealt with possible corrosive degradation of carbon steel parts due to the leakage of borated water and the resulting degradation of the reactor coolant pressure boundary. The NRC was requesting information regarding how utilities were meeting the requirements of 10 CFR 50, Appendix A, General Design Criteria 14, 30, and 31 - when fluid containing boric acid comes in contact with low alloy carbon steel. GL 88-05 requested assurance that a program had been implemented consisting of "systematic measures to ensure that boric acid corrosion does not lead to degradation of the assurance that the reactor coolant pressure boundary will have an extremely low probability of abnormal leakage, rapidly propagating failure, or gross rupture."

CR-3 fully implemented the boric acid corrosion program per PM-168. This program considered the programmatic requirements contained in GL 88-05, identified all potential sources of reactor coolant leakage and then implemented visual inspection of those locations. The program was designed to ensure that identified leakage paths would be thoroughly evaluated by Engineering for potential damage resulting from boric acid corrosion. A complete review of the inspection results and corrective actions taken would be performed. The purpose of this review would be to formulate a long-term plan for boric acid corrosion damage prevention. The inspection requirements of the program were based on the potential consequences of corrosion damage, previous inspection results, accessibility and personnel safety.

Currently, the Boric Acid Corrosion Inspection and Evaluation Program at CR-3 is integrated into other plant processes. The required inspections for this program are performed primarily by system engineers, during refueling outages and during plant shutdown (MODE 3). Plant walk-downs in the Reactor Building include the entire RCPB.

The inspections are performed by a team of engineers who are briefed on the requirements of GL 88-05 and PM-168. These engineers are trained to perform walk-downs. The training includes discussions of recent OE and requirements of PM-168 at the pre-job brief. Classroom training is accomplished by EPT-359, “Engineers Role in Equipment Aging Management,” which covers fundamentals of performing effective walk-downs. This allows engineers trained in performing walk-downs the opportunity to identify leakage as well as other issues that may require work during the outage, early enough to get the work properly planned and added to the current outage.

The training addresses the following elements for the engineering staff: how to look for deficiencies (i.e., obvious and subtle indicators, symptoms, causes and consequences, understanding equipment function, identifying aggressive environment and other potential causes of degradation etc.), when to perform the observations (i.e., Maintenance Rule frequency for (a)(1) and (a)(2) systems, opportunistic, system being opened, etc.), where to look (i.e., everywhere) and what to look for (i.e., unusual sounds, wetness, trash, etc.). The EPT-359 Training” provided to system engineers is used to assure specific issues are identified. This training is also used to sensitize the system engineers on any recent industry events occurring at other nuclear sites (i.e., Oconee, TMI, VC Summer, etc.).

During each refueling outage, certified VT-2 inspectors perform visual exams on all the bolted connections on the Reactor Coolant System (RCS). This exam is performed with the insulation removed. If boric acid residue is noted, the ASME code requires the bolting be removed and a VT-3 exam be performed on the bolting. The RCS system leak test is performed during plant startup (MODE 3) by certified visual inspectors. The inspection boundary includes the entire RCS. The CR-3 Corrective Action Program (CAP) is used to document and track conditions discovered that warrant further evaluation and it is used to distribute OE reports.

In a response to Industry Events, CR-3 has augmented the Inservice Inspection (ISI) Non Destructive (NDE) Program to include all the Reactor Pressure Vessel (RPV), Control Rod Drive Mechanisms (CRDM) nozzle welds and other Alloy 600 components. The Babcock &

Wilcox (B&W) Owners Group Materials Committee has performed susceptibility reviews and CR-3 has used that ranking to prioritize the inspections of these components. CR-3 has completed approximately 50% of the other Alloy 600 weld examinations (bare metal) for this interval. These items are scheduled with the remainder of the ISI exams and are performed by qualified and certified inspectors.

When evidence of leakage (i.e., boric acid crystals, water, etc.) is found, PM-168 and the plant's corrective action program require an assessment of the condition. PM-168 provides instructions for determining the amount of wastage, if any, and the impact on adjacent components. If the amount of wastage cannot be determined without the removal of the crystals, then guidance is provided to have the area cleaned and reevaluated after cleaning. The goal of the program is to have no leaks left in service, but if this is not achievable, guidance is provided to assess the leak, assess the estimated corrosion rate, determine the impact on adjacent components and then document the evaluation to allow continued service. These evaluations are documented in the CAP.

The health physics personnel performing decon evolutions are aware of boric acid corrosion impact and have instructions to document, via the Corrective Action Program, any abnormal indications of boron. Operators are also performing walk-downs to determine leakage amounts and cleanliness per the applicable surveillance procedures (SP-317 and SP-324).

APPLICABLE REGULATORY REQUIREMENTS

The following information provides a description of applicable regulatory requirements pertaining to GL 88-05 and Bulletin 2002-01.

GL 88-05 identified that a boric acid corrosion control program should include the following measures:

(1) A determination of the principal locations where leaks that are smaller than the allowable technical specification limit can cause degradation of the primary pressure boundary by boric acid corrosion. Particular consideration should be given to identifying those locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces.

Response:

As part of the Boric Acid Corrosion, Inspection and Evaluation Program, CR-3 identified the principal leakage locations. These locations were included in procedure, PM-168. An evaluation of the procedure was performed in 1999, which resulted in an increased observation of the entire RCPB, including Alloy 600 RCS components, and other systems interacting with the RCPB. The procedure was revised to inspect the entire RCPB (and other systems containing borated water).

(2) Procedures for locating small coolant leaks (i.e., leakage rates at less than technical specification limits). It is important to establish the potential path of the leaking coolant and the reactor pressure boundary components it is likely to contact. This

information is important in determining the interaction between the leaking coolant and reactor coolant pressure boundary materials.

Response:

When the MODE 3 walk-downs are performed, the results are communicated to Operations via Work Request, CAP or inter-office memo. Indications of leakage are investigated by Operations using surveillance procedure SP-317, "RC System Water Inventory Balance." This procedure is the basis for the identified and unidentified leakage at CR-3. In addition, each refueling outage, surveillance procedure SP-204, "Class 1 System, System Leakage Test for Inservice Inspection," is used to identify leakage.

(3) Methods for conducting examinations and performing engineering evaluations to establish the impact on the reactor coolant pressure boundary when leakage is located. This should include procedures to promptly gather the necessary information for an engineering evaluation before the removal of evidence of leakage, such as boric acid crystal buildup.

Response:

The methodology for conducting examinations and performing evaluations is contained in Preventative Maintenance procedure, PM-168. In addition, the CAP is used to identify, document, track, investigate, correct and trend adverse conditions.

(4) Corrective actions to prevent recurrences of this type of corrosion. This should include any modifications to be introduced in the present design or operating procedures of the plant that (a) reduce the probability of primary coolant leaks at the locations where they may cause corrosion damage and (b) entail the use of suitable corrosion resistant materials or the application of protective coatings/claddings.

Response:

CR-3 has implemented the Chesterton valve repack modification. This has eliminated a substantial amount of observed boric acid crystals on the RCPB. In addition, component bolting has been changed to a resistant material on a case-by-case basis. The CAP is used to determine the appropriate corrective actions to prevent recurrence of corrosion issues. The canopy seal modification for DHV-3 is an example of a recent modification to eliminate leakage that could cause RCPB degradation.

GENERAL DESIGN CRITERIA AND APPLICABLE REGULATORY GUIDANCE

The general design criteria (GDC), as stated in the Bulletin and the GL, 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."

<u>Discussion</u>

As described in the CR-3 Final Safety Analysis Report (FSAR), 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.

Therefore, CR-3 concludes that the design criterion continues to be met.

10 CFR 50, Appendix A
Criterion 30 – Quality of Reactor Coolant Pressure Boundary

"Components, which are part of the reactor coolant pressure boundary, shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage."

Corresponding CR-3 Criterion 16 – Monitoring Reactor Coolant PressureBoundary (Category B)

"Means shall be provided for monitoring the reactor coolant pressure boundary to detect leakage."

<u>Discussion</u>

As described in the CR-3 FSAR, RCPB integrity can be continuously monitored in the control room by the surveillance of variation from normal conditions for the following:

a. Reactor building sump level
b. Reactor building radioactivity levels
c. Condenser off-gas radioactivity levels (to detect steam generator tube leakage)
d. Decreasing makeup tank water level (indicating system leakage)

Gross leakage from the reactor coolant boundary will also be indicated by a decrease in pressurizer water level and rapid increase in the reactor building sump water level as described in FSAR Section 4.2.3.8.

Appropriate actions are taken to identify leakage sources. Once the source is identified, the CAP determines the appropriate corrective actions.

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

As described in the CR-3 FSAR, 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 influence induced degradation.

Therefore, CR-3 concludes that the design criterion continues to be met.

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 leak tight 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 leak tight 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-A.

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.

10 CFR 50.55a Codes and Standards

ASME Class I components (which include RCPB, Reactor Vessel Head (RVH) and Control Rod Drive Mechanism (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 welds and references IWB-3000 for acceptance standards.

Discussion

CR-3 has performed inspections of the RCPB and the RVH during previous refueling outages using volumetric, surface, and visual examination techniques. The visual examinations include direct observation and indirect observation for leakage. The direct inspection of the RVH is conducted through the access openings in the Control Rod Drive Service Structure (CRDSS) and is a bare metal inspection. Direct examinations are also performed in other Alloy 600 components and bolted connections. 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 CAP. During RFO 12, 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."

Discussion

Monitoring and various leakage detection systems are available that provide diverse methods of detection of unidentified leakage to the plant operator to ensure appropriate corrective actions are taken in accordance with ITS.

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. Discovery of RCPB leakage would require the plant to shutdown. With the plant in the shutdown condition, leakage should reduce due to the lower pressure.

Visual inspections conducted during refueling outages provide the opportunity to access areas/components within the plant that are normally not accessible during plant operations.

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

System engineers conduct the visual inspections performed to satisfy the requirements of GL 88-05. They are trained to perform walk-downs (EPT-359). Activities related to inspection and repair of RCPB components are controlled as required by the Florida Power Corporation (FPC) Quality Assurance (QA) Program for CR-3. Additional processes and procedures required for 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 RCPB, NDE and repair 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 nonconformance's 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

CR-3 meets this criterion. 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.