Progress Energy

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

Ref: 10 CFR 50.54(f)

January 28, 2003
3F0103-03

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

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

References:

1. NRC to FPC letter, 3N1102-06, dated November 22, 2002, Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," 60-Day Response for Crystal River Unit 3 Request for Additional Information (TAC No. MB4539)

2. FPC to NRC letter, 3F0502-01, dated May 15, 2002, Crystal River Unit 3 – 60-Day Response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

3. 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:

Reference 1 contains nine questions regarding the Crystal River Unit 3 (CR-3), 60-Day Response to Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity." Answers to those questions are provided in the Attachment to this letter.

The Attachment provides the basis for concluding that CR-3's 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 to be evaluated and enhanced, as needed, incorporating industry experience and best practices.

This letter establishes no new regulatory commitments.

A095

Progress Energy Florida, Inc.
Crystal River Nuclear Plant
15760 W. Powerline Street
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Although not required by CR-3 procedures, this Request for Additional Information response has been reviewed by the Plant Nuclear Safety Committee and comments were incorporated.

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

Dale E. Young
Vice President,
Crystal River Nuclear Plant

DEY/lvc

Attachment: Response to Request for Additional Information, Items 1 Through 9 Regarding Crystal River Unit 3, 60-Day Response for 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

PROGRESS ENERGY FLORIDA, INC.

CRYSTAL RIVER UNIT 3

DOCKET NUMBER 50-302/LICENSE NUMBER DPR-72

ATTACHMENT

Response to Request for Additional Information, Items 1 Through 9 Regarding Crystal River Unit 3, 60-Day Response for NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

Introduction:

The NRC staff's review of the licensees' responses to Bulletin 2002-01 resulted in the following Request for Additional Information (RAI). In accordance with NRC's request, Progress Energy Florida, Inc. is providing the NRC questions and the responses for Crystal River Unit 3 (CR-3) to the RAI. The information provided below, in conjunction with information previously provided, constitute the basis for concluding that CR-3's boric acid inspection program is providing reasonable assurance of compliance with the applicable regulatory requirements discussed in Generic Letter 88-05 and Bulletin 2002-01.

Question 1:

Provide detailed information on, and the technical basis for, the inspection techniques, scope, extent of coverage, and frequency of inspections, personnel qualifications, and degree of insulation removal for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary. Include specific discussion of inspection of locations where reactor coolant leaks have the potential to come in contact with and degrade the subject material (e.g., reactor pressure vessel bottom head).

Response:

The technical bases for examination of Alloy 600 pressure boundary material and dissimilar metal Alloy 82/182 welds and connections in the reactor coolant pressure boundary (RCPB) is consistent with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, and Generic Letter 88-05.

In addition to those Alloy 600 components to be examined under the rules of ASME Section XI, CR-3 has augmented the Inservice Inspection (ISI) Non-Destructive Examination (NDE) Program to include Reactor Pressure Vessel (RPV) penetrations, Control Rod Drive Mechanisms (CRDM) nozzles, and other Alloy 600 components. The Babcock & Wilcox Owners Group (BWOG) Materials Committee has performed Primary Water Stress Corrosion Cracking (PWSCC) susceptibility reviews. CR-3 has used that ranking as a basis for the Augmented Alloy 600 Program examinations. The augmented inspection encompasses the interfaces where reactor coolant leaks have the potential to come in contact and produce RCPB degradation. Augmented components are VT-2 examined for evidence of leakage, including boric acid residue, following insulation removal. The augmented component examinations are scheduled to coincide with the 10-year ASME XI ISI component examination schedules. ASME Code, and augmented examinations, are performed by ASNT-TC-1A qualified and certified examiners. Examiners performing augmented VT-2 examination receive additional training to recognize the characteristics of small volume boric acid leakage. CR-3 has completed approximately 50% of the augmented Alloy 600 weld examinations (bare metal) for this interval. The results of the visual inspection of the CRDM nozzle penetrations performed during Refueling Outage 12 (fall 2001) and the corrective actions taken as a result of leakage from a CRDM nozzle were provided in Florida Power Corporation (FPC) to NRC letter, 3F1101-04, dated November 19, 2001, Crystal River Unit 3 – Information Requested in Item 5 of NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles."

The following table identifies the susceptible Alloy 600 pressure boundary components and Alloy 82/182 welds at CR-3. Also included in the table are the inspection techniques, frequencies, degree of insulation removal and types of insulation.

Component (Alloy 600 pressure boundary material and Alloy 82/182 welds)	Quantity	Inspection Techniques	Extent of Coverage	Frequency	Degree of Insulation Removal	Insulation Type
Core Flood Tank (CFT) Level Sensing Nozzles	2/Tank	Visual (VT-2)	100%	1 / 40 months per Section XI	Not insulated	N/A
Core Flood Tank (CFT) Make-up Nozzle	1/Tank	VT-2	100%	1 / 40 months per Section XI	Not insulated	N/A
Core Flood Tank (CFT) Outlet Weld	1/Tank	VT-2	100%	1 / 40 months per Section XI	Not insulated	N/A
Core Flood Tank (CFT) Pressure Relief Nozzle	1/Tank	VT-2	100%	1 / 40 months per Section XI	Not insulated	N/A
Core Flood Tank (CFT) Pressure Sensing Nozzle	2/Tank	VT-2	100%	1 / 40 months per Section XI	Not insulated	N/A
Core Flood Tank (CFT) Sample Connection	1/Tank	VT-2	100%	1 / 40 months per Section XI	Not insulated	N/A
Once Through Steam Generator (OTSG) Primary Drain	1/OTSG	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Pressurizer Lower Level Sensing Nozzle	3	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Pressurizer Pressure Relief Nozzle weld	3	Penetrant Test (PT)	100%	1 / 120 months per Section XI	100%	Fiberglass
Pressurizer Sample Nozzle	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Pressurizer Spray Nozzle Safe End	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Fiberglass
Pressurizer Surge Nozzle Weld	1	Ultrasonic Examination (UT) / PT	100%	1 / 120 months per Section XI	100%	Reflective
Pressurizer Thermowell	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Pressurizer Upper Level Sensing Nozzle	3	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Pressurizer Vent Nozzle	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Fiberglass
Reactor Coolant System (RCS) Decay Heat Nozzle	1	UT / PT	100%	1 / 120 months per Section XI	100%	Reflective

Component (Alloy 600 pressure boundary material and Alloy 82/182 welds)	Quantity	Inspection Techniques	Extent of Coverage	Frequency	Degree of Insulation Removal	Insulation Type
Reactor Coolant System (RCS) Drain Nozzle	3	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Drain Nozzle Safe End	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Flow Meter Nozzle	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) High Pressure Injection Nozzle Weld	4	UT / PT	100%	1 / 120 months per Section XI	100%	Reflective
Reactor Coolant System (RCS) Lower Cold Leg Resistive Temperature Element Mounting Boss	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Lower Cold Leg Temperature Connections	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Piping Reactor Coolant Pump Inlet / Outlet Welds	8	UT / PT	100%	1 / 120 months per Section XI	100%	Reflective
Reactor Coolant System (RCS) Piping Surge Nozzle Welds	1	UT / PT	100%	1 / 120 months per Section XI	100%	Reflective
Reactor Coolant System (RCS) Hot Leg Pressure Tap Nozzle	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Cold Leg Pressure Tap Nozzle	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Resistive Temperature Element Mounting Boss	4	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Temperature Connection	2	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Coolant System (RCS) Vent Nozzle	2	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Vessel Core Flood Weld	2	UT from Inside Diameter (ID)	100% scan performed (Actual calculated coverage is 86%)	1 / 120 months per Section XI	Not required	N/A

Component (Alloy 600 pressure boundary material and Alloy 82/182 welds)	Quantity	Inspection Techniques	Extent of Coverage	Frequency	Degree of Insulation Removal	Insulation Type
Reactor Vessel Control Rod Drive Mechanism (CRDM) Motor Tube Welds (2)	2/Tube (136)	PT	6 motor tubes examined	1 / 120 months per Section XI	100%	Reflective
Reactor Vessel Control Rod Drive Mechanism (CRDM) Nozzle to Head J-Groove Weld	1/Nozzle (69)	VT-2	100%	Each Refueling Outage	100%	Reflective
Reactor Vessel Control Rod Drive Mechanism (CRDM) Nozzle Welds (2)	2/Nozzle (138)	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
CRDM Nozzle Forgings	1/Nozzle (69)	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective
Reactor Vessel Incore Instrumentation Nozzles	52	VT-2	25% (See Response to Question 2)	1 / 120 months per Section XI	33%	Reflective
Reactor Vessel Monitor Tap Weld	1	VT-2	100%	1 / 120 months per Augmented Program	100%	Reflective

Question 2:

Provide the technical basis for determining whether or not insulation is removed to examine all locations where conditions exist that could cause high concentrations of boric acid on pressure boundary surfaces or locations that are susceptible to primary water stress corrosion cracking (Alloy 600 base metal and dissimilar metal Alloy 82/182 welds). Identify the type of insulation for each component examined, as well as any limitations to removal of insulation. Also include in your response actions involving removal of insulation required by your procedures to identify the source of leakage when relevant conditions (e.g., rust stains, boric acid stains, or boric acid deposits) are found.

Response:

CR-3 removes all the insulation from the Alloy 600 and Alloy 82/182 welds to perform the examinations listed in the table in Question 1. The exception has been the bottom head Reactor Vessel Incore Instrumentation Nozzles. These nozzles have been determined to be low probability of failure per the B&W Owners Group (BWOG). Based on this, the examinations to date have consisted of examining the insulation while installed and general area reviews when the insulation is removed to perform scheduled ISI on the Reactor Vessel Support Skirt. VT-3 examination is performed on the interior of the reactor vessel support at three positions along the circumference of the support located 120 degrees apart. Approximately 33% of the Reactor Vessel bottom insulation is removed to perform this exam. There has not been an observed accumulation of boric acid crystals in this area. Based on recent Operating Experience (OE), CR-3 has scheduled the bottom head incore instrumentation nozzles for a complete VT-2 examination during the next refueling outage scheduled for fall 2003 (R13). The insulation will be removed to provide 100% access to the nozzles.

In addition, the CR-3 Boron Corrosion Control Procedure (PM-168) contains the following requirement when performing evaluations of components with observed boric acid deposits:

"NOTE: For any components where boron crystals prevent inspections of component parts a reinspection may be required after the boric acid has been removed from the component. This is dependent upon the presence of known or suspected low-alloy or carbon steel parts. If they are present, an inspection of the component a second time after removal of boric acid will be required to determine if degradation has occurred to these component parts."

Question 3:

Describe the technical basis for the extent and frequency of walk-downs and the method for evaluating the potential for leakage in inaccessible areas. In addition, describe the degree of inaccessibility, and identify any leakage detection systems that are being used to detect potential leakage from components in inaccessible areas.

Response:

The walk-downs performed to detect evidence of leakage from borated systems are scheduled in a manner consistent with 10 CFR 50.65, 10 CFR 50.55 and CR-3 Improved Technical Specifications (ITS). 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 at the beginning of refueling outages and during plant shutdown (MODE 3). Plant walk-downs in the Reactor Building include the entire RCPB. Focused exams of specific areas (i.e., RPV Head, scheduled ISI) are performed during refueling outages when access is permitted. This methodology results in RCPB being completely accessible for examination.

These inspections are performed by a team of engineers who are briefed by the Boric Acid Corrosion Control program engineer on the requirements of GL 88-05 and PM-168. These engineers are trained to perform walk-downs.

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.).

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 schedule.

The training includes discussions of recent OE and requirements of PM-168 at the pre-job brief.

During each refueling outage, certified VT-2 inspectors perform visual exams on all the bolted connections on the Reactor Coolant System (RCS) as required by the ASME Code, Section XI. 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. Additionally, the source of the leakage is located and corrected, and the surrounding area is assessed in accordance with PM-168 (Response to Question 4). The

ASME Code-required 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, track, investigate and correct adverse conditions. At CR-3, OE is controlled under Action Tracking, apart of the CAP.

The radiation protection personnel performing decontamination evolutions are aware of boric acid corrosion impact and have instructions to document, via the CAP, any signs of boric acid accumulation observed on external surfaces of components. Operators are also performing walk-downs to determine leakage amounts and cleanliness per the applicable surveillance procedures (SP-317, "RCS Water Inventory Balance," and SP-324, "Containment Inspection").

CR-3 does not have an installed local leakage detection system, however, very small amounts of leakage can be detected. CR-3 currently monitors, tracks, and trends RCS leakage to hundredth of a gallon per minute. As described in the CR-3 Final Safety Analysis Report (FSAR), Reactor Coolant Pressure Boundary 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 is used to determine the appropriate corrective actions.

Question 4:

Describe the evaluations that would be conducted upon discovery of leakage from mechanical joints (e.g., bolted connections) to demonstrate that continued operation with the observed leakage is acceptable. Also describe the acceptance criteria that were established to make such a determination. Provide the technical basis used to establish the acceptance criteria. In addition,

a. if observed leakage is determined to be acceptable for continued operation, describe what inspection/monitoring actions are taken to trend/evaluate changes in leakage, or

b. if observed leakage is not determined to be acceptable, describe what corrective actions are taken to address the leakage.

Response:

When evidence of leakage (i.e., boric acid crystals, water, etc.) is found, plant procedure PM-168 and the plant's CAP 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 (for non-RCPB leaks), 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. This approach was developed based on the guidance contained in EPRI Report, TR-1027485, "Boric Acid Corrosion Guidebook," and is contained in PM-168. The evaluation of the component will include any reinspection/monitoring requirements.

Steps from the boric acid corrosion control (BACC) program procedure include the following considerations or requirements:

EVALUATIONS

When evaluating components due to boric acid concerns, consider the need for a Design Change or Operating Procedure change. The goal of the BACC Program is to eliminate the leak source to reduce future inspections and maintenance activities.

Evaluation of an active leak and a justification for continued operations (JCO) and/or startup from a shutdown with a borated system leak shall include the following:

- Characterize the leakage and degradation.
- Predict leakage and degradation until repair can be implemented.
- Assess future degradation versus code requirements for full qualification of component.
- Establish subsequent inspection requirements.
- Determine most probable failure mechanism AND predict effects of this failure to plant operations.

- IF the leak is excessive (i.e., large cleanup activity, high probability of excessive wastage, etc.),
  THEN recommend immediate corrective action.
- Evaluate leaks found during a shutdown to include the following:
  Determine if the component can be repaired online.
  All inaccessible components should be repaired/replaced prior to startup unless a JCO has been performed for the condition.
  Initiate a work request (WR) to clean, repair and/or replace component(s) if required.
  If leak is determined to be active, ensure a corrective action document has been initiated to stop the leak (i.e., adjust packing, initiate WR, initiate CAP Document, etc.).
  Active leaks are tracked.

If leakage is not determined to be acceptable, the schedule for repair/replacement activities is established.

Question 5:

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. The NRC has had a concern with the bottom reactor pressure vessel head incore instrumentation nozzles because of the high consequences associated with loss of integrity of the bottom head nozzles. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response:

The Boric Acid Control Program at CR-3 will detect low levels of reactor coolant leaks that may result from through-wall cracking in the bottom reactor pressure vessel head incore instrumentation nozzles through visual observation. The incore nozzles have been partially observed when the insulation was removed to perform scheduled ISI on the RV support skirt. Although the PWSCC susceptibility of the Incore Monitoring Instrumentation (IMI) nozzles is believed to be low due to low operating temperature (See response to Question 2), the entire area is currently scheduled for 100% bare metal inspection during the next refueling outage (R13). During R13, the insulation will be removed to provide 100% access to the penetrations and the area will be VT-2 examined. The results will be documented and evaluated per the BACC Program (BACC Program Evaluation requirements are described in the response to Question 4 above, these requirements include the establishment of subsequent inspection requirements) and the CAP. The BACC Program Evaluation requirements also include an assessment of components that are in the leak path, that may be impacted by the leakage.

Question 6:

Explain the capabilities of your program to detect the low levels of reactor coolant pressure boundary leakage that may result from through-wall cracking in certain components and configurations for other small diameter nozzles. Low levels of leakage may call into question reliance on visual detection techniques or installed leakage detection instrumentation, but have the potential for causing boric acid corrosion. Describe how your program would evaluate evidence of possible leakage in this instance. In addition, explain how your program addresses leakage that may impact components that are in the leak path.

Response:

As explained in the response to Question 4 above, when evidence of leakage (i.e., boric acid crystals, water, etc.) is found, plant procedure 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 (for non-through-wall leaks), 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. This approach was developed based on the guidance contained in EPRI Report TR-1027485 "Boric Acid Corrosion Guidebook," and is contained in plant procedure PM-168. The evaluation of the component will include any reinspection/monitoring requirements.

Question 7:

Explain how any aspects of your program (e.g., insulation removal, inaccessible areas, low levels of leakage, evaluation of relevant conditions) make use of susceptibility models or consequence models.

Response:

The inspection frequency of the Alloy 600/82/182 material components was based on the BWOG research contained in the Framatome Technologies Inc. (FTI) document 51-5003018-00 titled, "Program Plan for Alloy 600 PWSCC Life Cycle Management." This proprietary document uses susceptibility models and industry experiences to calculate the "Relative Time to Failure." CR-3 inspects the susceptible Alloy 600 pressure boundary components identified in the response to Question 1.

The inspections have been added to the ISI NDE Program as augmented exams and are performed coincident with the ASME Code required examinations. The examinations are performed by certified and qualified inspectors.

Question 8:

Provide a summary of recommendations made by your reactor vendor on visual inspections of nozzles with Alloy 600/82/182 material, actions you have taken or plan to take regarding vendor recommendations, and the basis for any recommendations that are not followed.

Response:

In addition to the BWOG research referenced in the response to Question 7, CR-3 has received documentation from Framatome ANP (log 4-02 file 205/T4.4 PSC 4-02) which includes the following statements of recommendation for the lower head IMI nozzles:

"Under ideal circumstances, NDE (UT/eddy current examination (ECT)/PT) would be the suggested corrective action. However, given the current state of qualified NDE techniques and accessibility concerns, a bare metal visual examination of the IMI nozzle/lower RV head interface area at all B&W-fabricated 177-FA reactor vessels should be performed at the earliest opportunity. Other actions may also be appropriate to consider."

Based on this recommendation and current industry experiences, CR-3 is planning a detailed VT-2 bare metal examination of the RPV bottom head IMI nozzle interface area during R13 currently scheduled for the fall of 2003. These exams have been added to the augmented ISI program and have been identified to outage management as required outage work.

Question 9:

Provide the basis for concluding that the inspections and evaluations described in your responses to the above questions comply with your plant Technical Specifications and Title 10 of the Code of Federal Regulations, Section 50.55(a), which incorporates Section XI of the American Society of Mechanical Engineers (ASME) Code by reference. Specifically, address how your boric acid corrosion control program complies with ASME Section XI, paragraph IWA-5250 (b) on corrective actions. Include a description of the procedures used to implement the corrective actions.

Response:

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.

IWA-5250 (b) states, "If boric acid residues are detected on components, the leakage source and the areas of general corrosion shall be located. Components with local areas of general corrosion that reduce the wall thickness by more than 10% shall be evaluated to determine whether the component may be acceptable for continued service, or whether repair/replacement activities will be performed." While this comes from a later edition of Section XI than the CR-3 ASME Code of record, CR-3 program essentially have the same requirements to locate leakage and assess the impact of any corrosion on the structural integrity of the affected component(s).

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 and Boric Acid residue. 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 on 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 Refueling Outage 12 (2001), one CRDM nozzle was identified and confirmed as leaking from the visual inspections of the Reactor Vessel Head (RVH). The CRDM nozzle was repaired prior to restart from the refueling outage. No degradation of the RVH carbon steel was identified.

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."

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 further degradation of the RCPB does not continue. Discovery of RCPB leakage would require the plant to shutdown.

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

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

A round-robin self-assessment of all three Progress Energy PWR's boric acid corrosion control programs has been performed. The results of the self-assessment will be used to enhance some aspects of the programs to ensure high standards, requirements, and operating experience are consistently implemented at each site.