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LIC-02-0034

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

- References:
1. Docket No. 50-285
 2. NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"(NRC-02-034)
 3. Letter from OPPD (R. P. Clemens) to NRC (Document Control Desk) dated August 31, 2001, Response to NRC Bulletin 2001-01, "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles" (LIC-01-075)

SUBJECT: Response to NRC Bulletin 2002-01, "Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity"

In accordance with Reference 2, the Omaha Public Power District (OPPD) is submitting the Fort Calhoun Station (FCS) response to Nuclear Regulator Commission (NRC) Bulletin 2002-01.

NRC Bulletin 2002-01 requires each addressee to supply information on the integrity of the reactor coolant pressure boundary (RCPB), inspections and the basis for distinguishing compliance with the applicable regulatory requirements. This information is presented in the Attachment.

Information related to cracking of reactor pressure vessel head penetration (VHP) nozzles was previously docketed in Reference 3. In that submittal, OPPD noted that the industry has developed a susceptibility ranking model based on time-at-temperature for VHP nozzle cracking, which ranks FCS as moderately susceptible to primary water stress corrosion cracking (PWSCC).

FCS intends to perform a 100% visual inspection of the reactor vessel head during the spring 2002 refueling outage (RFO). Following the completion of the 2002 RFO, FCS will submit the results of the inspection and associated corrective actions.

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I declare under penalty of perjury that the foregoing is true and correct. (Executed on April 1, 2002)

Sincerely,



Ross T. Ridenoure
Division Manager
Nuclear Operations

RTR/RLJ/fj

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
W. C. Walker, NRC Senior Resident Inspector
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Fort Calhoun Station's
Response to NRC Bulletin 2002-01

**Fort Calhoun Station's
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i. Overview of Reactor Coolant Pressure Boundary (RCPB) Management at FCS

The Davis-Besse experience of RCPB degradation (based on the preliminary assessment at Davis-Besse) suggests the requirement for reinforcement between penetration nozzles has exceeded the American Society of Mechanical Engineers code requirement. The Davis-Besse event would imply that the nuclear industry's institutionalized boric acid reliability programs for a reasonable assurance of RCPB structural integrity are being challenged.

The following discussion will demonstrate that the Davis-Besse condition is unlikely to occur at FCS, due to the ease of reactor vessel (RV) head accessibility, RV head inspections performed, and maintenance procedures implemented to manage the RCPB material reliability issues at FCS. These inspections and procedures provide a reasonable assurance for RCPB structural integrity and also provide compliance with the applicable regulatory requirements. Specifically, this conclusion is based on actions such as facilitating visual inspection of the RV head by replacing RV head insulation during the 1983 refueling outage (RFO) and performing a visual inspection of the RV head. Additionally, the FCS staff has shown awareness of the effects of boric acid corrosion by performing RV head insulation removal, RV head cleaning and a 100% visual inspection on the RV head during the 1992 RFO. This visual inspection was undertaken to address leakage of a spare control element drive mechanism upper housing assembly, which occurred during the previous operating cycle.

The Davis-Besse event stresses the necessity for accessibility to the RV head insulation above and below the RV head insulation. FCS has the ability of visually inspecting the RV head surface when the unidentified leakage rate trends above normal operating conditions. This leakage rate trending has been successfully implemented by FCS in the past, and is reflected in the use of daily FCS surveillance test OP-ST-RC-3001 "Reactor Coolant System (RCS) Leak Rate Test." FCS is also fortunate to have a CEDM enclosure, which was originally designed to improve accessibility through six access ports during Incore Instrumentation removal and re-installation at each RFO.

FCS will describe the reasons for stating that compliance is being met by material reliability management of the RCPB, which in turn provides reasonable assurance of structural integrity. In conclusion, the applicable regulatory requirements as presented in NRC Bulletin 2002-01 are being met at FCS.

1. Requested Information:

1.A. RV head Inspection and Maintenance Programs:

The Bulletin states:

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

RV Head Inspections

FCS replaced the original blanket insulation on the RV head with reflective stepped mirror insulation, during the 1983 RFO. This RV head insulation modification included cleaning, and inspection per the Inservice Inspection (ISI) program and no head degradation was reported. The RV head insulation was again removed and inspected during the 1992 RFO. The 1992 RFO inspection observed the high temperature aluminum RV head coating to be in excellent condition, before and after the head was cleaned with demineralized water. Partial insulation removal is also performed during each RFO for RV flange stud de-tensioning, removal, cleaning and inspection. This RV flange stud de-tensioning has not reported any significant boric acid degradation. No accumulation of boric acid was seen on the RV head or RV flange.

The following is a listing of RV head inspections at FCS:

EEAR No. FC-79-15, “Reactor Vessel Head Seismic Skirt Insulation”

Purpose: This modification removed the original blanket insulation and replaced it with reflective stepped mirror insulation as the result of Incore Instrumentation flange leakage. As part of the removal process an ISI was done on 1/13/83 per procedure step 6.9.

MWO No. 908165, 2/23/90

Purpose: Removed the RV head insulation per MP-RC-6-12-A.

MWO No. 910243, 3/24/92

Purpose: Removed the RV head insulation and cleaned the insulation with demineralized water before re-installing the insulation on the RV head. Also cleaned off any boric acid deposits on the RV head surface. Did not replace insulation until all inspections of head area were done.

MWO No. 938165, 10/5/93

Purpose: Removed the reactor vessel head insulation per PE-PP-RC-1000.

MWO No. 958165, 2/24/95

Purpose: Removed the reactor vessel head insulation per PE-PP-RC-1000.

MWO No. 968165, 10/10/96

Purpose: Removed the reactor vessel head insulation per PE-PP-RC-1000.

MWO No. 972135, 4/10/98

Purpose: Used maintenance procedure PE-RR-RC-1000 to remove the insulation on the RV head studs.

The above activities have not observed any significant accumulation of boric acid above or below the RV head insulation.

Maintenance Programs

The listed FCS maintenance procedures below provide input to the Boric Acid program, which was instituted in response to Generic Letter 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988 and with the guidance of the Electric Power Research Institute TR-104748 "Boric Acid Corrosion Guidebook: Recommended Guidance for Addressing Boric Acid Corrosion and Leakage Reduction Issues", dated 02/01/1995. Surveillance testing for assessing RCPB integrity is also performed at the end of each FCS RFO.

MM-RR-RC-1000, "Cleaning of Reactor Vessel Studs, Nuts, and Washers."

Purpose: This procedure provides instructions for cleaning of reactor vessel studs, nuts, and washers, using the Combustion Engineering stud cleaning machine and manual cleaning methods.

MM-RR-RC-0312, "Reactor Vessel Flange Cleaning and Preparation."

Purpose: The purpose of this procedure is to provide for safe and correct cleaning and preparation of the RV flange. In addition, the RV flange is prepared so installation of the Closure Head can take place, after RV Internals have been returned to RV and water level in RV pool has been lowered.

MM-RR-RC-0313, "Reactor Vessel Closure Head Flange Penetration."

Purpose: The purpose of this procedure is to provide for safe and correct preparation of the Reactor Vessel (RV) Closure Head flange. This procedure will normally be used for cleaning and preparing the mating surface of the Closure Head flange during refueling outages. This procedure also provides for replacement of the Closure Head O-Rings, and for replacement of O-Ring attachment hardware if required.

OP-ST-RC-3007, "Reactor Coolant System Integrity Test Following Opening, Repair or Modification."

Purpose: A visual examination to locate any leakage of the pressure retaining components in the Reactor Coolant System (RCS) located within containment after a closure, repair, or modification. This test is performed following each RCS closure following opening, modification, or repair. This test satisfies, after each RCS closure, the requirements of Technical Specification 3.4(1) and in part Technical Specification 3.3(1)a.

OP-ST-RC-3001, "Reactor Coolant System (RCS) Leak Rate Test"

Purpose: Satisfies the daily requirements of FCS Technical Specification 3.2, Table 3-5, Item 8. This test also satisfies, in part, the requirements of FCS Technical Specifications, Section 3.3(1)a., and the leakage test requirement for HCV-348, per Section 3.3(1)a. Finally, the test satisfies the conditions of FCS Technical Specification 2.1.4.

The above activities have not observed any significant accumulation of boric acid above or below the RV head insulation.

1.B. RV Head Inspection and Maintenance Programs Effectiveness:

The Bulletin states:

"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,"

At the time of issuance of this Bulletin a root cause evaluation was not available for assessing the cause of degradation of the reactor pressure vessel head at Davis-Besse for preparing the FCS response of RCPB material reliability management. The information available to FCS suggests three possible causes, which are as follows:

1. The occurrence of significant accumulation of crystallized boric acid on the RV head as reported by Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," has the potential to degrade the pressure boundary by reduction of the RV head thickness. This is due to a continuous flow of borated water producing localized wetting of the reactor head vessel surface in an oxygenated environment, which further exacerbates the corrosion process.

2. Boric acid leakage from a through-wall crack of the reactor head vessel nozzle near the J-groove partial penetration weld, has the potential to reduce the reinforcement support between the RV head nozzle penetrations. This concern is related to the potential effects of boric acid corrosion-erosion as a result of leakage through the annulus between the nozzle and RV head. The elevated wastage rate is due to the surface attack and the wall shears produced from borated water jet impingement.
3. The combined leakage of boric acid from a flange and RV head penetration through-wall crack that would accumulate at the same location and broaden the corrosion wastage rate previously proposed by the industry.

If FCS has significant accumulation of crystallized boric acid build-up on and/or below the RV head insulation, as reported by the Florida Power & Light's Turkey Point 4 plant experience from a leaking conoseal (Preliminary Notification of Event 11-87-19), this condition would cause RCPB degradation. However, this amount of crystallization would have been identified, due to accessibility of the FCS RV head insulation, the full and partial inspections undertaken with the insulation removed, and the instituted maintenance programs. In addition, the FCS Boric Acid program reports no significant accumulation of crystallized boric acid build-up has been observed on the RV head, nor has staining of the mirror insulation been detected. Leakage above the RV head has been minor, and corrected within an operating cycle before any boric acid has accumulated on the head.

The possibility of a through-wall leak causing corrosion-erosion has been assessed by the Combustion Engineering Owners Group (CEOG) as a result of Generic Letter 88-05 "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants," March 17, 1988 and Generic Letter 97-01 "Degradation of Control Rod Drive Mechanism Nozzle and Other Vessel Closure Head Penetrations," April 1, 1997.

An evaluation by CEOG Task 744, Final Report CE NPSD-949-P, "Evaluation of Boric Acid Corrosion of Reactor Vessel Heads Resulting from Leaking CEDM Nozzles" of the degraded condition from leaking CEDM nozzles was assessed for determining the amount of reinforcement reduction that could be allowed to maintain ASME code acceptance criteria. This report contains calculations directly applicable to FCS. A conservative estimate for maximum degradation was determined to be 10.47 in³ for a 0.1 gpm leak over 9.8 years and assuming a corrosion rate of 1.07 in³/yr. This degradation rate is consistent with NUREG/CR-6245 "Assessment of Pressurized Water Reactor Control Rod Drive Mechanism Nozzle Cracking," October 1994. Therefore, the FCS inspection during the 1992 RFO along with an unidentified leakage rate (per FCS procedure OP-ST-RC-3001) that has not been consistently above 0.1 gpm for 9.8 years, suggests reinforcement loss would be ASME code acceptable. In addition, the nuclear industry Inconel 600 cracking concern at RV head penetrations was assessed by CEOG

Task 1003 "RPV Head Nozzle Evaluations-CE Owners Group Units", December 1998 and provided a 34% probability definition for developing a flaw signature by 2020 for FCS, which is comparable to the NRC Bulletin 2001-01 definition of 17.9 effective full power years from the Oconee event during 2001. This industry information suggests the possibility of a through-wall crack leaking borated water and causing RCPB degradation is currently unlikely at FCS. Therefore, a possible condition where both a leak from above the FCS RV head and a significant FCS RV head nozzle through-wall leak would cause boric acid accumulation on the RV head surface and result in excessive RV head degradation is also unlikely.

In conclusion, based on the FCS Boric Acid program and a consistently low FCS RCS leakage rate, a low probability exists of a through-wall FCS RV head nozzle penetration leak. The RV head inspections and maintenance programs add further reasonable assurance that RCPB structural integrity is being met at FCS.

1.C. Identified Chemical Deposits and Head Degradation:

The Bulletin states:

"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,"

FCS tracks the status of the RV head through observation of boric acid wastage identified in the maintenance procedures cited in 1.A. These procedures are undertaken during each RFO, reviewed as part of OPPD's commitment to Generic Letter 88-05 and their observations are summarized in the Boric Acid Program. Any identified boric acid wastage on the reactor coolant pressure boundaries is dispositioned by cleaning, repair, replacement and/or evaluated based on the measured degradation impact of a component's safety function. In conclusion, no RV head degradation has been reported to the Boric Acid program.

1.D. RV Head Future Inspections:

The Bulletin states:

“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,”

The future inspection at FCS is an effective, 100% visual inspection of the RV head that is planned during the May 2002 Refueling Outage. This examination is a visual inspection of the RV head for indications of popcorn (crystallized) boric acid build-up in the vicinity of each nozzle penetration and also the RV head surface for wastage. The qualification requirements and acceptance criteria are given in the OPPD response to NRC Bulletin 2001-01. FCS will continue to monitor industry experience and take appropriate actions on RV head issues.

1.E. RV Head Inspection and Maintenance Programs Compliance:

The Bulletin states:

“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 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 inspection are performed.”

NRC Bulletin 2002-01 section entitled Applicable Regulatory Requirements cites the following regulatory requirements and plant commitments as providing the basis for the bulletin assessment:

1. Appendix A to 10 CFR Part 50, General Design Criteria for Nuclear Power Plants (see section 1.E.1)
 - Criteria 14 - Reactor Coolant Pressure Boundary
 - Criteria 31 - Fracture Prevention of Reactor Coolant Boundary, and
 - Criteria 32 - Inspection of Reactor Pressure Coolant Pressure Boundary
2. 10 CFR 50.55a, Codes and Standards, which incorporates by reference Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, of the ASME Boiler and Pressure Vessel Code (see section 1.E.2)
3. Appendix B of 10 CFR Part 50, Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants, Criteria V, IX, and XVI (see section 1.E.3)
4. Plant Technical Specifications (see section 1.E.4)
5. Generic Letter 88-05, "Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants" (see section 1.E.5)

This section discusses how FCS meets the cited regulatory requirements and plant commitments affect plant decisions relating to NRC Bulletin 2002-01.

1.E.1 Design Requirements: 10CFR § 50, Appendix A

The Bulletin states:

"The applicable GDC include GDC 14 (Reactor Coolant Pressure Boundary), GDC 31 (Fracture Prevention of Reactor Coolant Pressure Boundary), and GDC 32 (Inspection of Reactor Coolant Pressure Boundary). GDC 14 specifies that the reactor coolant pressure boundary (RCPB) have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture. GDC 31 specifies that the probability of rapidly propagating fracture of the RCPB be minimized. GDC 32 specifies that components which are part of the RCPB have the capability of being periodically inspected to assess their structural and leak tight integrity; inspection practices that do not permit reliable detection of degradation are not consistent with this GDC."

However, these referenced "General Design Criteria" were not part of FCS's licensing requirement. The criteria that are similar to GDC 14, 31 and 32 (per FCS USAR Appendix G) are the following:

- Criterion 9 - Reactor Coolant Pressure Boundary (similar to GDC 14)

"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."

- Criterion 34 - Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention (similar to GDC 31)

"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."

- Criterion 36 - Reactor Coolant Pressure Boundary Surveillance (similar to GDC 32)

"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 with ASTM-E-185-66 shall be provided."

The following information demonstrates how FCS complies with the design criteria for the reliability of reactor coolant pressure boundary integrity:

The FCS components are designed and constructed in accordance with ASME Boiler & Pressure Vessel Code, Section III. The combined static and transient stress is limited, whenever the RV temperature is below Nil-Ductility Temperature of +60°F, to sufficiently low values to make the probability of a rapidly propagating failure extremely remote. The required stress limits are maintained by operating restrictions. The test inspection requirements were to assure that flaw sizes will be limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the RV, on which tests and inspections are imposed to meet and exceed code requirements. In addition, ECT inspections of the Control Element Drive Mechanism (CEDM) seal housing assemblies have characterized an environmental condition (stagnancy) that provides FCS with an unique definition of VHP nozzles risk. This environmental characterization in conjunction with a comparatively low

material yield strength and design stress of the VHP nozzles, and overall design considerations are therefore adequate and manageable as prescribed by ASME XI table IWB-2500-1.

1.E.2 Inspection Requirements: 10 CFR. § 50.55a and ASME Section XI

The Bulletin states:

"NRC regulations at 10 CFR 50.55a state that American Society of Mechanical Engineers (ASME) Class 1 components (which includes the reactor coolant pressure boundary) must meet the requirements of Section XI of the ASME Boiler and Pressure Vessel Code. Various portions of the ASME code address reactor coolant pressure boundary inspection. For example, Table IWA-2500-1 [IWB-2500-1] of Section XI of the ASME Code provides examination requirements for reactor pressure vessel head penetration nozzles and references IWB-3522 for acceptance standards. IWB-3522.1(c) and (d) specify that conditions requiring correction include the detection of leakage from insulated components and discoloration or accumulated residues on the surfaces of components, insulation, or floor areas which may reveal evidence of borated water leakage, with leakage defined as "the through-wall leakage that penetrates the pressure retaining membrane." Therefore, 10 CFR 50.55a, through its reference to the ASME Code, does not permit through-wall degradation of the reactor pressure vessel head penetration nozzles.

For through-wall leakage identified by visual examinations in accordance with the ASME Code, acceptance standards for the identified degradation are provided in IWB-3142. Specifically, supplemental examination (by surface or volumetric examination), corrective measures or repairs, analytical evaluation, and replacement provide methods for determining the acceptability of degraded components."

Title 10 of the Code of Federal Regulations, Part 50.55a requires that in service inspection and testing be performed per the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, Inservice Inspection of Nuclear Plant Components. Section XI contains applicable rules for examination, evaluation and repair of code class components, including the reactor coolant pressure boundary.

Requirements for partial penetration welds attaching control rod drive mechanism housings to the RV head are contained in Table IWB-2500-1, Examination Category B-E, Pressure Retaining Partial Penetration Welds in Vessels, Items Numbers: B4.10, Partial Penetration Welds; B4.11, Vessel Nozzles; B4.12, CRDM Nozzles; and B4.13, Instrumentation Nozzles. The Code requires a VT-2 "visual examination" of 25% of the CRDM nozzles from the external surface. Since the head is insulated, and the nozzles do not represent a bolted flange, paragraph IWA-5242(b) permits these inspections to be performed with the insulation left in place.

The acceptance standard for the visual examination is found in paragraph IWA-5250, Corrective Measures. Paragraph IWA-5250 requires repair or replacement of the affected part if a through-wall leak is found and requires an assessment of damage, if any, associated with corrosion of steel components by boric acid.

Flaws identified by nondestructive examination (NDE) methods which are beyond current requirements are evaluated in accordance with the flaw evaluation rules for piping contained in Section XI of the ASME Code. This approach has been accepted by the NRC. Any flaw not meeting requirements for the intended service period would be evaluated by the program plan before returning it to service.

Industry repairs to RPV top head nozzles have been performed in accordance with Section XI requirements, NRC-approved ASME Code Case requirements, or an alternative repair or replacement method approved by the NRC.

FCS complies with these ASME Code requirements through implementation of the plant's Inservice Inspection program. If a VT-2 examination detects the conditions described by IWB-3522.1(c) and (d), then corrective actions per IWB-3142 would be performed in accordance with FCS's corrective action program. No new plant actions are necessary to satisfy the cited regulatory criteria.

1.E.3 Quality Assurance Requirements: 10 CFR. § 50, Appendix B

The Bulletin states:

"Criterion V (Instructions, Procedures, and Drawings) of Appendix B to 10 CFR Part 50 states that 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. Criterion V further states that instructions, procedures, or drawings shall include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished. Visual and volumetric examinations of reactor coolant pressure boundary are activities that should be documented in accordance with these requirements."

Criterion V is also a forward-looking criterion that applies should the bulletin response identify new inspections. It does not establish criteria for when or if inspections should be performed. If new inspections are performed, they will meet criterion V.

The Bulletin further states:

“Criterion IX (Control of Special Processes) of Appendix B to 10 CFR Part 50 states that special processes, including nondestructive testing, shall be controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Within the context of providing assurance of the structural integrity of reactor coolant pressure boundary for the degradation observed at Davis-Besse, special requirements for visual examination and/or ultrasonic testing methods would generally require the use of a qualified visual examination method and ultrasonic testing methods. Such a methods are ones that a plant-specific analysis has demonstrated would result in the reliable detection of degradation prior to a loss of specified reactor coolant pressure boundary margins of safety. The analysis would have to consider, for example, the as-built configuration of the system and the capability to reliably detect and accurately characterize flaws or degradation, and contributing factors such as the presence of insulation, preexisting deposits, and other factors that could interfere with detection of degradation.”

Criterion IX is a forward-looking requirement such that if inspections are performed they must be controlled and accomplished by qualified personnel. No action is required by a FCS to satisfy this criterion, unless a new inspection is proposed. However, if the Bulletin response identifies a new inspection then the response should identify how Criterion IX is satisfied

The last Appendix B criterion cited in the bulletin is:

"Criterion XVI (Corrective Action) of Appendix B to 10 CFR Part 50 states that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. For significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. For degradation of the reactor coolant pressure boundary, the root cause determination is important to understanding the nature of the degradation present and the required actions to mitigate future degradation. These actions could include proactive inspections and repair of degraded portions of the reactor coolant pressure boundary."

Criterion XVI has two attributes that should be considered by FCS in its response to the Bulletin.

The first attribute is that measures shall be established to assure that conditions adverse to quality are promptly identified and corrected. This criterion infers FCS's responsibility to be aware of industry experience, and has been interpreted in this manner in most plant's corrective action programs. FCS should determine if an industry experience applies to its plant and what, if any,

corrective actions are appropriate. This approach is consistent with the NRC's generic communication process for an Information Notice, which reports industry experience, but does not require a response to the NRC. FCS is expected to evaluate the applicability of the occurrence to the plant and document a record of the plant specific assessment for possible NRC review during inspections.

Criterion XVI provides the objectives and goals of the corrective action program, but licensees are responsible for determining a specific process to accomplish these goals and objectives. With regard to the bulletin response, Criterion XVI does not provide specific guidance as to what is an appropriate response, but rather, the FCS is responsible for determining actions necessary to maintain public health and safety. That is, the FCS must justify its actions for addressing the stress corrosion cracking of vessel head penetrations. Furthermore, the regulatory criteria of 10 CFR 50.109(a)(7), provides supporting evidence when it states that if there are two or more ways to achieve compliance then ordinarily the applicant or licensee is free to choose the way which best suits their purposes.

The second attribute of Criterion XVI that should be considered is that for significant conditions adverse to quality, the measures taken shall include root cause determination and corrective action to preclude repetition of the adverse conditions. The Bulletin suggests that for cracking of vessel head penetrations, the root cause determination is important in understanding the nature of the degradation and the required actions to mitigate future cracking. The FCS corrective action program would determine the cause of any degradation of the reactor coolant pressure boundary. However, if no known degradation in the reactor coolant pressure boundary is identified through reasonable quality assurance measures or inspection and monitoring programs, this criterion would not require specific action on FCS for remaining in compliance with the regulation.

In summary, the integrated industry approach to inspection, monitoring, cause determination, and resolution of the identified reactor coolant pressure boundary degradation concern is clearly in compliance with the performance-based objectives of Appendix B.

1.E.4 Operating Requirement: 10 CFR § 50.36 - Technical Specifications

The Bulletin states:

"Plant technical specifications pertain to the issue insofar as they do not allow operation with known reactor coolant system pressure boundary leakage."

Title 10 of the Code of Federal Regulations, Part 50.36 (10CFR 50.36) contains requirements for Plant Technical Specifications. Paragraphs 2 and 3 of 10CFR Part 50.36 are particularly relevant:

· 10CFR 50.36 (2) Limiting Conditions for Operation

"Limiting conditions for operation are the lowest functional capability or performance levels of equipment required for safe operation of the facility. When a limiting condition for operation of a nuclear reactor is not met, the licensee shall shut down the reactor or follow any remedial action permitted by the technical specifications until the condition can be met.

A technical specification limiting condition for operation of a nuclear reactor must be established for each item meeting one of the following criteria:

Criterion 3: A structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Criterion 4: A structure, system or component which operating experience or probabilistic risk assessment has shown to be significant to public health and safety."

· 10 CFR 50.36 (3) Surveillance Requirements

"Surveillance requirements are requirements relating to test, calibration, or inspection to assure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions will be met."

The reactor coolant pressure boundary provides one of the critical barriers that guard against the uncontrolled release of radioactivity. Therefore, plant technical specifications generally include a requirement and associated action statements addressing reactor coolant pressure boundary leakage. The limits for PWR reactor coolant pressure boundary leakage are typically stated in terms of the amount of leakage, e.g., 1 gallon per minute for unidentified leakage; 5-10 gpm for identified leakage; and no leakage from a non-isolable fault in the reactor coolant system pressure boundary.

Most leaks from reactor coolant pressure boundary leakage have been well below the sensitivity of on-line leakage detection systems. FCS has evaluated this condition and has determined that visual inspections of the reactor head for boric acid deposits during plant shutdowns or NDE examination of the CEDM housing are appropriate inspections. If leakage or unacceptable indications are found, then the defect shall be evaluated by the program plan before the plant resumes operation. If through-wall boundary leaks of CEDMs increase to the point where they are picked up by the on-line leak detection systems, then the leak shall be evaluated per the specified acceptance criteria, and corrective action as specified in FCS Technical Specification, section 2.1.4 'Reactor Coolant System Leakage Limits' will be taken.

1.E.5 Boric Acid Program - Generic Letter 88-05

The Bulletin states:

“Generic Letter 88-05, “Boric Acid Corrosion of Carbon Steel Reactor Pressure Boundary Components in PWR Plants,” pertains to this issue in that the staff concluded that in the absence of a program for addressing the corrosive effects of reactor coolant system leakage, compliance with General Design Criteria 14, 30 and 31 cannot be ensured.”

FCS has implemented a Boric Acid Program for compliance to NRC Generic Letter 88-05, which has been updated in accordance with RFO maintenance procedure summaries, audits, self-assessments and FCS Station License Renewal.

2. Reporting of Future Inspection Results

FCS will provide the NRC with the following information within 30 days after plant restart following the 2002 RFO:

- A. The inspection scope and results, including the location, size and nature of any degradation.
- B. The corrective actions taken and the root cause of the degradation.

3. Remaining RCPB Requested Information

The Bulletin states:

“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 preceding sections of NRC Bulletin 2002-01 are to be considered as a response to concluding that the FCS Boric Acid inspection program is in compliance with applicable regulatory requirements. The 60 day response is considered to be met through the submittal of this document based on the details of the FCS Boric Acid program, which have been provided. The FCS Boric Acid program is in compliance with NRC Generic Letter 88-05 and has been updated with the input from both RFO maintenance procedures and surveillance tests. FCS self-

assessments and the FCS License Renewal project have also provided independent assessments of the Boric Acid program and have made recommendations, which have been incorporated in order to enhance the Boric Acid program.