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February 17, 2003
LIC-03-0020

U. S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D. C. 20555

References:
1. Docket No. 50-285
2. NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Degradation and Vessel Head Penetration Nozzle Inspection Programs," August 9, 2002 (NRC-02-113)
3. Letter from OPPD (R. T. Ridenoure) to NRC (Document Control Desk), dated September 11, 2002, Response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Degradation and Vessel Head Penetration Nozzle Inspection Programs" (LIC-02-0106)

SUBJECT: Revised Response to NRC Bulletin 2002-02, "Reactor Pressure Vessel Head Degradation and Vessel Head Penetration Nozzle Inspection Programs"

The Omaha Public Power District (OPPD) is submitting a revision to the Fort Calhoun Station (FCS) response, Reference 2, to Nuclear Regulator Commission (NRC) Bulletin 2002-02, Reference 3.

This submittal revises the Effective Degradation Years (EDYs) and Effective Full Power Years (EFPYs) reported in the table on page 8 of the Attachment to Reference 2. The calculation of the EDYs and EFPYs has been refined using a detailed accounting of the FCS power history. The inspection plan has been changed to be consistent with these revisions. Changes are indicated by sidebars in the submittal.

This letter contains the following commitment:

> In accordance with the NRC Bulletin 2002-02, FCS will perform an under-the-head inspection of RPV head penetration nozzles using an approved MRP qualified NDE inspection method. (A/R 32031)

I declare under penalty of perjury that the forgoing is true and correct. (Executed on February 17, 2003).

If you have any questions or require additional information, please contact Dr. R. L. Jaworski of the FCS Licensing staff at (402) 533-6833.

A096

Sincerely,

R. T. Ridenoure
Division Manager
Nuclear Operations

RTR/RLJ/rlj

Attachment

c: E. W. Merschoff, NRC Regional Administrator, Region IV
A. B. Wang, NRC Project Manager
J. G. Kramer, NRC Senior Resident Inspector
Winston & Strawn

Fort Calhoun Station's Revised Response to NRC Bulletin 2002-02

Table of Contents

1.0 Summary

Fort Calhoun Station (FCS) has evaluated the material condition concerns with regard to Effective Full Power Years (EFPY) and Effective Degradation Years (EDY) calculated in accordance with EPRI MRP-48[1]. Additionally, these evaluations of the Reactor Vessel (RV) integrity per the EDY definitions have been shown to be conservative as a generic industry assessment in comparison to a FCS plant specific evaluation that provides a more comprehensive understanding of the long term viability of the RV Head material integrity at FCS. This industry approach, in conjunction with an aggressive FCS defense-in-depth philosophy that incorporates leakage monitoring and control element drive mechanism (CEDM) housing inspections, demonstrates a detailed understanding of the FCS RV Head material condition.

In conclusion, FCS is committed to the safe operation of the RV Head and maintenance of its integrity. The FCS inspection plan will exceed the requirements of the EPRI MRP plan by proposing more frequent (less than 5 EFPY) RV Head visual examinations, which will detect RV Head leakage before any possibility of RV control rod ejection or RV Head wastage can occur. FCS will begin an under the head inspection starting with the 2005 RFO that will comply with the criteria specified in EPRI MRP-48 and is described in section 4.0 of this response.

2.0 Technical Discussion 1.B Response for Non-Supplemental Inspection

2.1.1 Concern (1):

Circumferential cracking of CRDM nozzles was identified by the presence of relatively small amounts of boric acid deposits. This finding increases the need for more effective visual and non-visual NDE inspection methods to detect the presence of degradation in control rod drive mechanism (CRDM) nozzles before integrity is compromised.

2.1.2 *Response (1):*

This concern is focused on determining primary boundary integrity through detection of material degradation that would result in circumferential cracking of the reactor vessel head penetration nozzles. There are currently four Non-Destructive Examination (NDE) methods that are considered code acceptable. One is a volumetric examination (ultrasonic testing) and the other three are surface type examination (visual, dye-penetrant, and eddy current testing). However, there is currently only one NDE method that has consistently detected material degradation and that is the visual surface examination of a pressure boundary. These other NDE methods are recognized for their limitations in finding and characterizing flaws due to the characteristically tight crack morphology phenomen

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1 EPRI Document TR 1006284, "PWR Material Reliability Program Response to NRC Bulletin 2001-01 (MRP-48)," August 2001

known as primary water stress corrosion cracking (PWSCC). Therefore, the visual surface examination continues to be a reliable and effective inspection technique for ensuring regulatory and technical specification as affirmed in OPPD's responses to NRC Bulletin 2001-01[2] and 2002-01[3].

2.2.1 **Concern (2):**

Cracking of Alloy 82/182 weld metal has been identified in CRDM nozzle J-groove welds for the first time and can precede cracking of the base metal. This finding raised concerns because examination of weld material is more difficult than base material.

2.2.2 ***Response (2):***

This concern suggests an increased risk in pressure boundary integrity due to a crack in the area of a J-groove weld. An initiated crack in the j-groove weld has been observed in laboratory tests to have a high growth rate. However, a J-groove weld crack does not have any significantly different stress distributions that would pose any higher risk for circumferential cracking of the penetration nozzle. In addition, the visual examination frequencies as proposed by the MRP Inspection Plan[4] conservatively consider leakage from either the weld and/or base material.

2.3.1 **Concern (3):**

Through-wall circumferential cracking from the outside diameter of the CRDM nozzle has been identified for the first time. This raises concerns about the potential for failure of CRDM nozzles and control rod ejection, causing LOCA.

2.3.2 ***Response (3):***

This concern is based on a condition where the circumferential critical flaw size for a CRDM nozzle goes undetected. However, the industry's MRP Alloy 600 probability model for visual surface examination and pressure boundary leak rate monitoring is an accurate and reliable method of ensuring adequate primary pressure boundary integrity.

The lower bound EDY of the highest susceptibility classification (12 EDY) in NRC Bulletin 2002-02 is based on one CE plant's code acceptable artifacts (if shallow indications are not destructively and metallographically substantiated then they are artifacts) that were found in the three nozzles. The

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2 Response to NRC Bulletin 2001-01 "Circumferential Cracking of Reactor Pressure Vessel Head Penetration Nozzles," date August 31, 2001, Letter No. LIC-01-0075.

3 Response to NRC Bulletin 2002-01"Reactor Pressure Vessel Head Degradation and Reactor Coolant Pressure Boundary Integrity," date April 16, 2002, Letter No. LIC-02-0034.

4 Inspection Plan PWR Reactor Pressure Vessel (RPV) Upper Head Penetrations Inspection Plan (MRP-75), Revision 01, dated September 2002.

NRC Bulletin is using data from this one CE plant to justify the 12 EDY classification. This CE plant's findings did not challenge the primary pressure boundary integrity. In a conservative way these findings (even if shown to be real cracks) validate the industry's probabilistic model[5] (that considers all CE plants) in which this particular CE plant is the most susceptible of the CE fleet to nozzle cracking, and FCS is a moderately susceptible plant.

The FCS experience of pressure boundary leakage (see Ref. No. 3) provides confidence in the application of our leak rate monitoring program. In 1990, a through wall axial crack in a spare CEDM upper housing that resulted in a loss of RCS inventory of about 0.2 gpm was successfully identified.

These actions, implementations and inspections all demonstrate the industry's effectiveness through implementation of the MRP Alloy 600 material reliability program, leak rate monitoring and visual surface examination of the RCS pressure boundary. This approach has demonstrated success in detecting material degradation, and thus ensures regulatory and technical specification compliance.

2.4.1 **Concern (4):**

The environment in the CRDM housing/RPV head annulus will likely be more aggressive after any through-wall leakage because potentially highly concentrated borated primary water may become oxygenated. This raises concerns about the technical basis for current crack growth rate models.

2.4.2 ***Response (4):***

This concern suggests primary water leakage is a significant component in challenging the industry's currently defined crack growth rate models. The current model is based on two known phases of PWSCC, i.e. the development of an incubation period (material change) that can transition to a slip plane dislocation period (cracking). This incubation period is driven by an environmental condition that causes surface change to a susceptible material (Alloy 600), which then becomes embrittled. This embrittled surface has the potential for crack initiation in which the slip plane energy is sufficient to produce grain structure/boundary dislocations (cracks). The suggestion in this concern proposes that the incubation period is significantly reduced due to the presence of concentrated boric acid in the area of the nozzle. Alloy 600 PWSCC requires an environment where the material surface cycles through passivation and repassivation to develop a material change that would become embrittled. In the event of through-wall leakage, the incubation period is insignificant (environment) and the slip plane dislocation process is the dominant force of crack growth propagation. During this condition a large amount of boric acid residue is produced that is detectable through either a visual surface examination and/or the reactor coolant system leak rate monitoring program. Finally, the

[5] CEOG Task 1003 "RPV Head Nozzle Evaluations-CE Owners Group Units," December 1998, figure II-2.

current industry model[6] includes a safety factor of two on crack growth rate that is bounded by the uncertainty in the composition of the boric acid in the remaining residue.

2.5.1 **Concern (5):**

The presence of boron deposits or residue on the RPV head, due to leakage from mechanical joints, could mask pressure boundary leakage. This raises concerns that a through-wall crack may go undetected for years.

2.5.2 ***Response (5):***

This concern is related to boric acid residue from other mechanical joints in the vicinity of an actual through-wall leak masking pressure boundary leakage. The industry observation of boric acid residue rundown resulting in a light and broad dusting of the reactor vessel surface has not interfered with the capability of performing successful visual surface examinations in order to detect leaking CRDM nozzles. Also, the conclusion from the FCS 2002 refueling outage (RFO) 100% reactor vessel head penetration inspection validated the reactor vessel head integrity and general condition as being in compliance (see Refs. No. 2 and 3) and good condition. For example, an undetectable leakage of a 0.003 gpm[7] , could result in boric acid residue build-up of 100 in^3 and this volume is considered detectable by a visual surface examination of the pressure boundary.

2.6.1 **Concern (6):**

The causative conditions surrounding the degradation of the RPV head at Davis-Besse have not been definitively determined. The staff is unaware of any data applicable to the geometries of interest that support accurate predictions of corrosion mechanisms and rates.

2.6.2 ***Response (6):***

The concern is the boric acid wastage rate current experience is not understood since there is no accepted root cause determination. The visual surface inspection of the RV head has demonstrated that a very small amount of leakage can be located (e.g., 0.5 in^3 of deposits)[8]. Testing results have demonstrated (see Ref No. 6) the threshold for significant corrosion to occur over relatively large areas would require the surface temperature be reduced to boiling. At FCS, this condition needs a leak rate in the order of 0.1 gpm that would produce about 23 ft^3 of boric acid deposits. The time to reach this critical leak rate for a FCS head temperature of 588°F is about 2.4 EFPY (see Ref. No.

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[6] EPRI Document MRP-55, "Crack Growth Rates for Evaluation Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," July 2002.

[7] EPRI Document TR 100975, Boric Acid Corrosion Guide Book, Rev. 1, dated November 2001.

[8] EPRI Document TR 1006899, Visual Examination for Leakage of PWR Reactor Head Penetrations on Top of RPV Head, Rev. No 1, dated November 01 2001.

10). In addition, the Combustion Engineering Owners Group evaluation[9] 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 FCS specific calculations for code acceptance limits. A conservative estimate for the maximum degradation was determined to be 10.47 in^3 for a 0.1 gpm leak over 9.8 EFPY and assuming a corrosion rate of 1.07 in^3/yr (2.15 in/yr). It should be noted the NRC Augmented Inspection Team 'Summary of Findings'[10] for the Davis-Besse Nuclear Power Station event stated "The apparent rate of boric acid corrosion was consistent with certain industry data." This is a reference to the EPRI Boric Acid Corrosion Guidebook, Rev. 1 (see Ref. No. 7) for a corrosion rate at 2.37 in/yr. This information conservatively suggests that in order to reach code limits at FCS it would take 8.9 EFPY (factored: Davis Besse/CEOG wastage rate) of undetected leakage through the annulus between the reactor vessel head and a penetration nozzle which would result in about 120 ft^3 of boric acid deposits. The nuclear industry's test data, gap evaluations[11] and experience support the effectiveness of visual surface examinations and the FCS proposed inspection plan per section 4.0 is conservatively more aggressive than the MRP inspection plan's frequency definition of 2.0 EDY or less than 5 EFPY between visual inspections.

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9 CEOG Task 744, Final Report CE NPSD-949-P, "Evaluation of Boric Acid Corrosion of Reactor Vessel Heads Resulting from Leaking CEDM Nozzles," November 1994

10 NRC Report No. 50-346/02-03(DRS), "Davis-Besse Nuclear Power Station NRC Augmented Inspection Team - degradation of the Reactor Pressure Vessel Head", dated May, 3 2002.

11 Calculation No. CN-CI-02-38, Rev. No. 0, "Reactor Vessel Head Penetrations Leakage Study for Omaha Public Power District, Ft. Calhoun Station", prepared by Westinghouse, LLC, dated 6/18/02.

3.0 **FCS Reactor Vessel Head Integrity Basis**

The FCS RV Head material condition has been assessed (see Ref. No. 2) and verified (see Ref. No. 3) as in compliance. OPPD has provided information that described the justification for continuing plant operations per NRC Bulletin 2001-01 and verification of the material condition of the RV Head per NRC Bulletin 2002-01 at the completion of 100% bare head visual inspection performed during the 2002 Spring RFO.

The industry has recently reaffirmed the characteristics of RV Head leakage that result in material degradation from boric acid corrosion (see Refs. No. 7 and 10). These characteristics correlate well with the industry's understanding of corrosion rate and mechanism. In addition, the industry has an adequate understanding of crack growth rate as the material changes from a resistant to a susceptible condition. The time for nozzle through-wall crack propagation at FCS is about 7.5 EFPY[12], and RV Head degradation to code limits is approximately an additional 8.9 EFPY (factored: Davis Besse/CEOG wastage rate). This assessment shows that the time to through-wall crack propagation and boric acid corrosion to code limits at FCS takes about 16 EFPY. Current industry experience confirms this assessment as conservative.

EPRI MRP-48 proposes a more conservative time table based on Oconee's estimated through-wall crack propagation time of 12.8 EDY with a normalized RV Head temperature of 600°F. For FCS the Table below shows the outage values of EDY and EFPY[13]:

Refueling Outage	*Outage Date*	*Effective Degradation Years*	*Effective Full Power Years*
2001 Spring	3/01	10.40 (base)	20.06 (base)
2002 Spring	5/02	11.3	21.1
2003 Fall	9/03	11.85	22.4
2005 Spring	4/05	12.83	23.9
2006 Fall	9/06	13.72	25.3

The above durations represent a generic assessment of material condition of Alloy 600 penetration nozzles and should not be considered as a plant specific prediction. This generic assessment treats environment, geometry, nozzle material and weld properties, supplier, fabricator, construction process and operating conditions as equal. On the other hand, CEOG Task 1003 (see Ref. No. 5)

12 CEOG Task 769, "CEOG Program to Address Alloy 600 Cracking of CEDM Penetrations-Inspection Timing Model," Prepared for the C-E Owners Group, dated April 1994

13 Dominon Engineering, Inc., "Projection of EDY for Fort Calhoun RPV Head Through Cycle 21 (9/2002)," Task No 3914-00-3, Calculation No. C-3914-00-1, dated 2/7/03.

is a plant specific, quantitative risk assessment of all the RV Head Alloy 600 penetration nozzles, which has factored in all parameters in an attempt to predict material reliability. The limitation in all these predictions is the incubation period that is subject to an environmental circumstance that is dependent on operating conditions, RV Head and CEDM geometry, chemistry controls and venting practices. FCS has been actively pursuing an environmental definition of the CEDM housing assemblies through NDE methods and destructive examination. This information has provided FCS with a better understanding and possible historical characterization of the general material condition of these housings, as well as the RV Head penetration nozzles.

In summary, FCS has an ongoing, comprehensive RV Head integrity inspection program, which is based on the following:

a) The CEOG analyses, which establish times for RV nozzle leakage to grow to conditions where the boric acid corrosion will reduce the FCS RV Head thickness to the minimum allowable ASME code acceptance for RV Head reinforcement.

b) The CEOG analyses of RV Head wastage rates.

c) A FCS specific gap analysis, which assures the Plant of being able to detect small leaks through the annulus between RV Head and RV nozzle.

d) FCS specific analyses from CEOG work using known fabrication data to establish margins for RV nozzle cracking. At FCS, the CEOG risk assessment (see Ref. No. 5) predicts a 34% probability of a crack to ASME Section XI code depth by 2020.

e) FCS has reviewed original nozzle fabrication records for any abnormalities, such as excessive grinding or multi-repairs, which could elevate residual stresses. No nozzle abnormalities were found.

f) FCS has performed two 100% bare metal examinations of the RV Head in the last ten years with good results from both inspections. The last inspection was during the 2002 Spring Outage.

g) A fully implemented FCS Boric Acid Inspection 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 then supplemented 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.

h) The FCS experience of a leak in a spare upper CEDM Housing in 1990 has demonstrated the importance of closely monitoring the leak rate of the RCS. This leakage test[14] has since been updated to more stringent requirements by the Operations and System Engineering Departments. FCS Design Engineering has recently reviewed all the data since the 1990 leak and found no indication of any abnormal ongoing RCS leakage.

i) FCS specific data and analyses have demonstrated that cracking in CEDM Housings is a precursor to RV Head penetration cracking. FCS performs refueling outage NDE testing on the most limiting housings. The tests include eddy current, liquid penetrant and ultrasonic testing. The emphases of these tests is aging management program of the material condition for reactor vessel head and assemblies.

j) The FCS CEDM seal housings program plan has a goal of systematically and pro-actively detecting cracking with a focus of increasing the material reliability. This program was developed in response to industry operation experience in stress corrosion cracking and plant specific concerns during the 1996 and 1998 RFO reactor vessel gas bubble events. The lack of theoretical data on low temperature stress corrosion cracking of stainless steels in conjunction with environment uncertainties such as degree of stagnancy, contaminant concentrations, stress and temperature histories prompted this program. The program also provides comprehensive inspection goals for acceptance, rejection and re-inspection of the CEDM seal housings.

k) The FCS Alloy 600 Program Plan addresses plant specific concerns for material reliability and management of PWSCC issues. This program plan will consolidate industry data, include In-Service Inspection and fabrication records, prioritize site specific concerns, develop contingency plans, address Alloy 600 chemistry concerns and recommend inspection activities.

l) FCS will continue to closely monitor the results of industry inspections and EPRI MRP studies. These results will be used to continuously validate the analytical assumptions and margins in the supporting studies for the EPRI MRP inspection program and accordingly support the FCS RV Head inspection plan.

m) Surveillance testing for assessing reactor coolant pressure boundary integrity is performed at the end of each FCS RFO and evaluated by the Engineering department for material degradation concerns.

The FCS RV Head inspection program was instituted as a defense-in-depth approach to ensure RV Head integrity is maintained. The FCS program has achieved this goal and continues to ensure that material reliability is maintained.

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

4.0 FCS Reactor Vessel Head Inspection Plan

A FCS plant specific evaluation (see Ref. No. 5) provides reasonable justification for limited NDE of the RV Head until 2020. In 2020, the plant specific evaluation shows that the probability of reaching an ASME Section XI code acceptable RV head crack at FCS is still low. However, due to a small uncertainty in material condition, crack growth rates and current industry wastage concerns it is reasonable to consider increasing NDE inspections.

Furthermore, FCS recognizes that although the analyses recommended by the EPRI MRP Inspection Program currently support the use of FCS visual examinations of the RV Head penetrations at one of the next two refueling outages, the NRC in Bulletin 2002-02 has recommended a more conservative designation of susceptibility. The NRC susceptibility assessment places FCS into the region of moderate susceptibility for the FCS Fall 2003 refueling outage (as noted in the EDY projected times in the Table in Section 3 of this response). FCS considers that the FCS detailed plant specific analyses on cracking and wastage provide confidence in the margins to the expected RV head cracking, and that the planned visual inspections using EPRI guidelines[15] for the Fall 2003 refueling outage are therefore appropriate without supplement.

In accordance with the NRC Bulletin 2002-02 FCS will perform an under-the-head inspection of RPV head penetration nozzles using an approved MRP qualified NDE inspection method. However, the type of NDE examination technique chosen will be based on the best available technology, which will provide minimal operational risk with achievement of ALARA goals. The chosen NDE technique will be consistent with the methods recommended in Bulletin 2002-02 or EPRI MRP-48. It should also be noted that OPPD is currently completing our evaluation for purchase of a new reactor vessel head. This would be installed in the 2006 RFO if implemented.

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[15] EPRI document 1006269, "Visual Examination for Leakage of PWR Reactor Head Penetrations," Rev. 1, dated January 2002