Omaha Public Power District
444 South 16th Street Mall
Omaha, NE 68102-2247

10 CFR 50.55a

LIC-15-0066
May 9, 2015

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

Fort Calhoun Station, Unit No. 1
Renewed Facility Operating License No. DPR-40
NRC Docket No. 50-285

Subject: Relief Request Number RR-14, Request for Relief from Paragraph -3142.1(c) of ASME Code Case N-729-1 for Reactor Vessel Head Penetration Nozzle Welds

Reference: 1. Letter from OPPD (L. P. Cortopassi) to NRC (Document Control Desk), "Revised Fourth Ten-Year Inservice Inspection (ISI) Interval," dated November 3, 2014 (ML14308A658) (LIC-14-0099)

Pursuant to 10 CFR 50.55a(z)(2), the Omaha Public Power District (OPPD) hereby requests NRC approval of the attached relief request for the Fort Calhoun Station (FCS), Unit No. 1 Inservice Inspection (ISI) Program, fourth ten-year interval. The attachment identifies the affected components, applicable American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME) Code requirements, reason for request, proposed alternative, and basis for proposed alternative. The alternatives are proposed to be applied during the next operating cycle and will conclude in the refueling outage scheduled to begin the fall 2016.

During the current twenty-seventh refueling outage the reactor vessel head penetration nozzles were cleaned and examined in accordance with the FCS developed work instructions (Reference Work Order (WO) 551120-01) and with the FCS ISI Program. OPPD identified evidence of leakage from mechanical connections above the reactor vessel head that travelled down onto the reactor vessel head and penetrations. Based on chemical testing and visual examination, OPPD concluded none of the observed deposits came from the penetration nozzles and no corrosion of the reactor vessel head material has occurred. As described in the attachment, OPPD is requesting relief from the requirement for a supplemental examination due to hardship without a compensating increase in the level of quality or safety. Relief is requested in accordance with 10 CFR 50.55a(z)(2).

The provisions of this relief are applicable to the fourth ten-year ISI interval for FCS, which commenced on September 26, 2003 and will end on June 6, 2017 (Reference 1).

OPPD requests approval of this request by May 14, 2015 to support restart from the current refueling outage.

There are no regulatory commitments contained in this submittal.

If you should have questions, please contact Mr. Bill Hansher at (402) 533-6894.

Respectfully,

Louis P. Cortopassi
Site Vice President and CNO

LPC/TJH/mle

Attachment: 10 CFR 50.55a Request Number RR-14

c: M. L. Dapas, NRC Regional Administrator, Region IV
C. F. Lyon, NRC Project Manager
S. M. Schneider, NRC Senior Resident Inspector

10 CFR 50.55a Request Number RR-14

Relief Requested
In Accordance with 10 CFR 50.55a(z)(2)

Hardship or Unusual Difficulty without Compensating
Increase in Level of Quality or Safety

1. ASME Code Component(s) Affected

Component:	Reactor Vessel Head (RVH) Nozzles
Code Class:	Class 1
Examination Category:	B-P
Code Item Number:	B4.30 (Code Case N-729-1, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1)
Description:	Control Element Drive Mechanism (CEDM) Nozzles In-Core Instrumentation (ICI) Nozzles RVH Vent Nozzle
Size:	7.48 Inch (Nominal Outside Diameter) 4.33 Inch (Nominal Outside Diameter) ¾ Inch (Nominal Pipe Size)
Material:	RVH SA-508 Grade 3 Class 1 Nozzle SB 167 N06690 (Alloy 690) Alloy 52/152 weld material

2. Applicable Code Edition and Addenda

- American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Section XI, 1998 Edition through 2000 Addenda
- Code Case N-729-1 as modified by 10 CFR 50.55a(g)(6)(ii)(D)

3. Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(D)(1) requires that examinations of the reactor vessel head be performed in accordance with ASME Code Case N-729-1 subject to the conditions specified in paragraphs 10 CFR 50.55a(g)(6)(ii)(D)(2) through (6).

Paragraph -3142.1 (c) of Code Case N-729-1 states:

A nozzle whose VE indicates relevant conditions indicative of possible nozzle leakage shall be unacceptable for continued service unless it meets the requirements of -3142.2 or -3142.3.

Paragraph -3142.2 of Code Case N-729-1 states:

A nozzle with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if the results of supplemental examinations [-3200(b)] meet the requirements of -3130.

Paragraph -3142.3 of Code Case N-729-1 states:

(a) A component with relevant conditions not indicative of possible nozzle leakage is acceptable for continued service if the source of the relevant condition is corrected by a repair / replacement activity or by corrective measures necessary to preclude degradation.
(b) A component with relevant conditions indicative of possible nozzle leakage shall be acceptable for continued service if a repair / replacement activity corrects the defect in accordance with IWA-4000.

Paragraph -3200(b) of Code Case N-729-1 states:

(b) The supplemental examination performed to satisfy -3142.2 shall include volumetric examination of the nozzle tube and surface examination of the partial-penetration weld, or surface examination of the nozzle tube inside surface, the partial penetration weld, and nozzle tube outside surface below the weld, in accordance with Fig. 2, or the alternative examination area or volume shall be analyzed to be acceptable in accordance with Appendix I. The supplemental examinations shall be used to determine the extent of the unacceptable conditions and the need for corrective measures, analytical evaluation, or repair / replacement activity.

4. Reason for Request

A visual examination requires a qualified inspector to examine the bare metal RVH upper head surface to look for relevant conditions that could cause degradation of the upper head or indicate potential nozzle penetration leakage. The Code Case states "relevant conditions shall be evaluated to determine the extent, if any, of degradation. The boric acid crystals and residue shall be removed to the extent necessary to allow adequate examinations and evaluation of degradation, and a subsequent VE of the previously obscured surface shall be performed prior to return to service, and again in the subsequent refueling outage."

Based on the visual examination and results of chemical analyses, the source of deposits were dispositioned to be leakage from mechanical connections above the RVH that traveled down onto the RVH and penetration nozzles. Areas recorded as relevant were cleaned to the extent necessary and re-inspected with acceptable results, and the RVH was subsequently cleaned. It has been determined that some areas were inappropriately recorded as not relevant. The RVH has been cleaned in excess of what is permissible prior to performing a subsequent VE in accordance with the Code Case -3142.1(b)(1). Although no degradation has been detected, the areas inappropriately recorded as not relevant can no longer have a VE performed that meets the Code Case.

The other allowable options per -3142.1(c) are supplemental examination per -3142.2 or a repair/replacement activity per -3142.3.

The reactor head was replaced in 2006 with nozzles and partial penetration welds made with Primary Water Stress Corrosion Cracking (PWSCC) resistant materials. Alloy 690 and corresponding weld metals Alloy 52 and 152 have demonstrated PWSCC resistance. This is demonstrated by no reported PWSCC indications in up to 24 calendar years of service for thousands of Alloy 690 steam generator tubes, and more than 22 calendar years of service for thick-wall and thin-wall Alloy 690 applications.

There is no evidence of a flaw in any of the RVH nozzles or partial penetration welds; therefore, performing emergent supplemental examination and/or repair / replace ment of the nozzles does not result in a compensating increase in the level of quality or safety.

The FCS RVH is currently located on the headstand. The headstand has been modified with scaffolding to hold temporary shielding until after the head is returned to the reactor vessel. FCS does not have the internal resources to conduct a volumetric examination as required by Code Case N-729-1 -3200(b). A third party vendor would be required to perform the examinations. The radiation shield for the RVH currently does not allow for vendor tooling to access underneath without modification. The RVH would be required to be lifted to allow removal of the scaffolding, staging of vendor equipment, and subsequently returned to the head stand. Movement of the RVH is a Heavy Load Lift as defined by MM-RR-RC-0305 “Removal of Reactor Vessel Closure Head, Hold Down Ring, and Upper Guide Structure” Rev. 38 and MM-RR-RC-0314 “Reactor Vessel Closure Head Installation” Rev. 28. Work performed within containment inherently includes industrial safety risk and radiological concerns. These risks are exacerbated when lifting of heavy loads and working within limited spaces.

The FCS Control Element Assembly (CEA) rack extensions are installed in the penetrations. At this time, the double CEA rack extensions in CEDM Penetration numbers 14, 15, 16, 17, 30, 31, 32, 33, 34, 35, 36, and 37 would interfere with existing vendor inspection tooling to access the required nozzle locations due to the shape of the extension. To complete the examination as required, the double CEA racks would need to be removed from the penetrations. Clearance for vendor tooling with respect to the installed CEA racks would require the head to be elevated from its current position. Elevation of the head on the headstand requires potential physical modification(s) and reanalysis for the increased height.

There is no qualified ultrasonic (UT) examination technique for the FCS ICI nozzles due to its thickness. Surface inspections may need to be performed by manual methods due to tooling and qualification status. The RVH vent line would be required to be manually tested by eddy current and/or dye penetrant examination techniques due to size. These inspection techniques would cause an increase in dose and time due to the lost efficiency of automated examinations. These examinations require personnel accessing the underside of the head, which is highly contaminated and has elevated dose rates thus challenging radiological safety.

It is estimated that a supplemental examination will require two Rem and five weeks to mobilize and complete the examinations. This will extend the outage duration by five weeks.

5. Proposed Alternative and Basis for Use

Repairs to identified boric acid leakage sources will be made on RC-100, CEDM RC-10-24, and ICI Port #44 during the current refueling outage. Repairs to identified component cooling water (CCW) leakage sources will be made on CEDMs RC-10-03, RC-10-12, RC-10-14, RC-10-27, RC-10-35, and RC-10-39 during the current refueling outage. This corrects the known sources of leakage. Procedure OP-ST-RC-3007, Periodic Reactor Coolant System Integrity Test, which specifies inspection areas inside containment including the ICI Ports, CEDM housings, and RC-100, will be performed. This surveillance test will verify the correction of the identified leaks and confirm the absence of new leaks.

FCS has performed a detailed review of the photographic record of the as found condition. This review did not provide any new insights relative to the potential for nozzle leakage. This inspection will be re-performed following final cleaning to provide additional confidence that nozzle degradation is not present.

The site will continue to monitor Reactor Coolant System (RCS) Leak-rate daily in accordance with Technical Specifications Section 3.2 Table 3-5 Item 8a. The allowable RCS leak-rate is further constrained by the provisions of ER-AP-331-1003, RCS Leakage Monitoring and Action Plan, which implements the provisions of WCAP-15988-NP, Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors.

The RVH was inspected on May 3, 2015 by a qualified VT-2 inspector that also had an additional four hours of boric acid training to meet the requirements set forth in Code Case N-729-1. The qualified inspector determined that no penetration had signs of degradation or evidence of nozzle leakage. The boric acid on the head was identified as originating from leaking mechanical connections on ICI Port #44, RC-100 packing (reactor vessel RC-1 head vent to Reactor Coolant Gas Vent System (RCGVS) isolation valve), and CEDM RC-10-24 mechanical connections. There was also CCW leakage seen on the head that is attributed to a cracked return hose on CEDM RC-10-03. RC-100 has been repaired under Work Order (WO) 551054, ICI Port #44 flange will be replaced under WO 550760, and RC-10-24 temperature element will be repaired under WO 550424. CEDM RC-10-03 will be repaired under WO 552344. All of these repairs will be completed before startup. CR 2015-05836 has an action to verify the correct as-left torque values are part of the work instructions with sign-offs for craft and QC, This CR will drive necessary procedure and/or process changes. Additionally, the CEDM Seal Housing on RC-10-24 was eddy current (ET) tested under WO 485681-22 with satisfactory results and no detectable indications. Eight other seal housings were eddy current tested as a repetitive task for refueling outages with no detectable indications.

Deposit samples /smears were obtained from RC-100 (smear), ICI Port #44 (smear), upper CEDM seal housing RC-10-24 (sample), CEDM RC-10-03 (sample), CEDM RC-10-08 (sample), CEDM RC-10-11 (sample), CEDM RC-10-38 (sample) for chemical analysis to characterize the deposit / smear and determine the source of the leakage. Deposit samples were also taken on the RVH insulation and on the head itself. Based on chemical analysis, the predominant chemicals identified in the samples were tolytriazole (CorrShield NT 4207) with minor concentrations of boron. These chemicals are consistent with the constituents of the chemical additive, CorrShield NT 4207, which is used in the CCW system. Isotopic comparisons between the RCS samples and the samples taken from the RVH were evaluated for Cobalt 58 and Cobalt 60. The comparison of the isotopic ratios from the samples substantiated the deposits were predominantly from CCW leakage with some minor contamination from leakage from the RC-100 packing and ICI Port #44 mechanical connection.

RCS leak-rate trended down over Cycle 27 and was comparable to Cycles 23, 24, and 25. Cycle 26 leak-rate was higher than previous cycles due to valve leakage on three Chemical and Volume Control System (CVCS) valves. These valves were subsequently repaired during FCR26 and following maintenance the leak-rate returned to a value consistent with previous cycles. The final quarter of Cycle 27 (1st Quarter 2015) the average total leak-rate was 0.084 gpm.

A review of Condition Reports and inspection history on the containment ventilation coolers was conducted. This review determined that the amount of boric acid deposits is consistent from cycle to cycle back to 1998. No changes have been required for filter Preventative Maintenance (PM) frequencies nor have any change outs from condition based monitoring for HEPA or Charcoal filters been required.

The FCS replacement RVH was fabricated by Mitsubishi Heavy Industries (MHI) and replaced the previous RVH in the fall of 2006. The RVH currently has accumulated 5.23 effective full power years (EFPY) of service. The RVH is constructed as a one piece forging of SA-508 Grade 3 Class 1 with SB-167 Alloy 690 penetrations and Alloy 52/152 J-groove welds. The FCS RVH operating temperature is 588°F. The RVH was hydrostatically tested on October 11, 2005 by MHI per MHI procedure UGS-L5-030146 Rev.3. The test pressure of 3147 psig was held for 18 minutes and documented as "Acceptable." MHI performed dye penetrant examinations on the J-groove welds during the welding process. The CEDM penetrations were dye penetrant tested after the first pass, 25%, 50%, 75%, and 100% completion. The ICI penetrations were dye penetrant tested after the first pass, approximately every 1/7th of the weld thickness up to 100% completion and tests were documented as "Acceptable". The pre-service inspection was performed in November of 2005, in accordance with the First Revised NRC Order EA-03-009, MRP-2003-13, and ASME Section XI and documented as "Acceptable".

The ET/UT techniques were performed in accordance with Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) demonstrated ET/UT techniques applicable at the time. Eddy current, ultrasonic, and manual dye penetrant testing was performed on all penetrations and J-groove welds with no recordable indications found.

As documented in MRP-375, the resistance of Alloy 690 and corresponding weld metals Alloy 52 and 152 is demonstrated by the lack of PWSCC indications reported in these materials, in up to 24 calendar years of service for thousands of Alloy 690 steam generator tubes, and more than 22 calendar years of service for thick-wall and thin-wall Alloy 690 applications.

This operating experience includes service at pressurizer and hot-leg temperatures higher than those at FCS and includes Alloy 690 wrought base metal and Alloy 52/152 weld metal. This experience includes ISI volumetric or surface examinations performed in accordance with ASME Code Case N-729-1 on 13 of the 41 replacement RVH currently operating in the U.S. fleet. This data supports a factor of improvement in time to detectable PWSCC flaw initiation of at least five to 20 when compared to service experience of Alloy 600 in similar applications worldwide.

Based upon the visual inspection already performed on the FCS RVH and the operating time of the RVH, it is not likely that the observed leakage is coming from a flaw in any of the nozzles or partial penetration welds. Therefore, performing emergent supplemental examination of the nozzles does not result in a compensating increase in the level of quality or safety.

6. Duration of Proposed Alternative

The proposed alternative will be utilized until the end of operating cycle 28 or such time a degraded RVH nozzle penetration is detected.

7. References

1. ASME Boiler and Pre ssure Vessel Code C ase N-729-1 "Alternative Exa mination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1"
2. WO 551120-01 "RC-6 Clean Boric Acid and Inspect Reactor Vessel Closure Head"
3. OP-ST-RC-3007 "Reactor Coolant System Integrity Test Following Opening, Repair"
4. Materials Reliability Program: Technical Ba sis for Reexamination Interval Extension for Alloy 690 PWR Rea ctor Vessel Top Head Penetration Nozzles (MRP-375) 2014 Technical Report
5. CR 2015-05836 Rollup CR for Leaks identified on the Reactor Vessel Head
6. Work Order (WO) 551120 CR 2015-05255 - Clean RV Head To Remove Boric Acid Residue
7. Work Order (WO) 551054 Packing Leak On RC-100
8. ICI Port#44 flange will be replaced under WO 550760 ICI (Port #44): Clean, Inspect And Repair Boric Acid Leak
9. RC-10-24 temperature element WO 550424 CR 2015-04170 - Clean Disassemble And Repair Threaded Fitting
10. CEDM RC-10-03 WO 552344 Check Tightness Of CCW Fittings On CEDMs 3, 8, And 19.
11. CEDM Seal Housing RC-10-24 WO 485681-22 RC-10-24: Perform Eddy Current Testing On Seal Housing
12. CEDM RC-10-12 WO 550266 Check Tightness of CCW Fittings On CEDM 12
13. CEDM RC-10-14 WO 550267 Check Tightness of CCW Fittings On CEDM 14
14. CEDM RC-10-21 WO 550265 Check Tightness of CCW Fittings On CEDM 21
15. CEDM RC-10-35 WO 550264 Check Tightness of CCW Fittings On CEDM 35
16. CEDM RC-10-39 WO 524684 Check Tightness of CCW Fittings On CEDM 39
17. MM-RR-RC-0305 "Removal of Reactor Vessel Closure Head, Hold Down Ring, and Upper Guide Structure" Rev. 38
18. MM-RR-RC-0314 "Reactor Vessel Closure Head Installation" Rev. 28