UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 2015

Mr. Louis P. Cortopassi
Site Vice President and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station
9610 Power Lane, Mail Stop FC-2-4
Omaha, NE 68008

SUBJECT: FORT CALHOUN STATION, UNIT NO. 1 – REQUEST FOR RELIEF RR-14, FROM CERTAIN REQUIREMENTS OF ASME CODE CASE N-729-1 FOR REACTOR VESSEL HEAD PENETRATION NOZZLE WELDS (TAC NO. MF6206)

Dear Mr. Cortopassi:

By letter dated May 9, 2015, as supplemented by letters dated May 13, 16, 17, and 18, 2015, Omaha Public Power District (OPPD, the licensee) submitted request for relief RR-14 to the U.S. Nuclear Regulatory Commission (NRC) for the inspection of reactor vessel head (RVH) nozzles at Fort Calhoun Station, Unit 1 (FCS). In RR-14, the licensee proposed to use alternative inspection requirements for RVH nozzles with respect to American Society of Mechanical Engineers (ASME) Code Case N-729-1, "Alternative Examination Requirements for PWR [Pressurized-Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1," as conditioned in Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(g)(6)(ii)(D) until the end of operating cycle 28 or until a degraded RVH nozzle is detected.

The NRC staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that RR-14 will provide reasonable assurance of the structural integrity of the RVH and attached nozzles. The NRC staff concludes that complying with the specified inspection in accordance with ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D), would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) and is in compliance with the requirements of ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). Therefore, the NRC authorizes the use of RR-14 at FCS until the end of operating cycle 28, which is currently scheduled for fall 2016, or until a degraded RVH nozzle is detected.

All other requirements of ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The NRC staff provided verbal authorization for RR-14 during a teleconference with your staff on May 17, 2015. If you have any questions, please contact Fred Lyon at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF RR-14 FROM CERTAIN REQUIREMENTS OF ASME CODE CASE

N-729-1 FOR REACTOR VESSEL HEAD PENETRATION NOZZLE WELDS

OMAHA PUBLIC POWER DISTRICT

FORT CALHOUN STATION, UNIT NO. 1

DOCKET NO. 50-285

1.0 INTRODUCTION

By letter dated May 9, 2015 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML15129A004), as supplemented by letters dated May 13, 16, 17, and 18, 2015 (ADAMS Accession Nos. ML15135A387, ML15136A002, ML15142A411, and ML15147A155, respectively), Omaha Public Power District (the licensee) submitted request for relief RR-14 to the U.S. Nuclear Regulatory Commission (NRC) for the inspection of reactor vessel head (RVH) nozzles at Fort Calhoun Station, Unit 1 (FCS). In RR-14, the licensee requested relief from the inspection of RVH nozzles per American Society of Mechanical Engineers (ASME) Code Case N-729-1 "Alternative Examination Requirements for PWR [Pressurized Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," as conditioned in Title 10 of the *Code of Federal Regulations* (10 CFR), paragraph 50.55a(g)(6)(ii)(D), "Reactor vessel head inspections."

Specifically, pursuant to 10 CFR 50.55a(z)(2), the licensee requested to use the proposed alternative in RR-14 on the basis that compliance with the specified ASME requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

On May 17, 2015 (ADAMS Accession No. ML15139A010), the NRC verbally authorized the use of RR-14 at FCS until the end of operating cycle 28, which is currently scheduled for fall 2016, or until a degraded RVH nozzle is detected. This safety evaluation describes the technical basis of the NRC's verbal authorization.

2.0 REGULATORY EVALUATION

The regulations in 10 CFR 50.55a(g)(4) require that ASME Code Class 1, 2, and 3 components will meet the requirements, except the design and access provisions and the pre-service examination requirements, set forth in the ASME Code, Section XI.

Enclosure

Pursuant to 10 CFR 50.55a(g)(6)(ii), the Commission may require the licensee to follow an augmented inservice inspection (ISI) program for systems and components for which the Commission deems that added assurance of structural reliability is necessary.

Licensees of PWRs are required by 10 CFR 50.55a(g)(6)(ii)(D to augment their ISI of the RVH nozzles in accordance with ASME Code Case N-729-1, with conditions, as a result of operating experience of primary water stress-corrosion cracking (PWSCC) in RVH nozzles which are fabricated with nickel-based Alloy 600/82/182 material.

The regulations in 10 CFR 50.55a(z) state that alternatives to the requirements of paragraph (g) of 10 CFR 50.55a may be used, when authorized by the NRC, if the licensee demonstrates that: (1) the proposed alternative provides an acceptable level of quality and safety, or (2) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request and the Commission to authorize the alternative requested by the licensee.

3.0 RELIEF REQUEST RR-14

3.1 ASME Code Components Affected

The affected components are control element drive mechanism (CEDM) nozzles, in-core instrumentation (ICI) nozzles, and a vent pipe nozzle of the RVH.

The affected components are ASME Class 1, Examination Category, B-P, Code Item Number, B4.30 per ASME Code Case N-729-1.

The nominal outside diameter of the CEDM and ICI nozzles are 7.48 inches and 4.33 inches, respectively. The vent pipe is of 3/4-inch, nominal pipe size. The RVH is made of SA-508, Grade 3, Class 1. The CEDM and ICI nozzles are made of SB 167, N06690 (Alloy 690) and the associated J-groove welds are made of Alloy 52/152 weld material. The RVH currently has accumulated 5.23 effective full power years (EFPY) of service. The RVH operating temperature is 588 degrees Fahrenheit (°F).

3.2 Applicable Code Edition and Addenda

The Code of record is the ASME Code, Section XI, 1998 Edition through 2000 Addenda.

3.3 Applicable Code Requirement

The regulations in 10 CFR 50.55a(g)(6)(ii)(D)(1) require that examinations of the RVH 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(b)(2) of Code Case N-729-1 states, in part, that "...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 [visual examination (VE)] of the previously obscured surface shall be performed prior to return to service, and again in the subsequent refueling outage."

Paragraph -3142.1(c) of Code Case N-729-1 states that "[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, in part, that "...[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, in part, that "...(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, in part, that "...(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."

3.4 Reason for Request

During refueling outage 27 (FCR27) in May 2015, the licensee detected residues on several CEDM, ICI, and vent pipe nozzles of the RVH. Based on the VE and results of chemical analyses, the licensee dispositioned the source of deposits to be leakage from mechanical connections above the RVH that traveled down onto the RVH and RVH penetration nozzles. The licensee cleaned areas recorded as relevant conditions and re-inspected with acceptable results, and the entire RVH was subsequently cleaned. The licensee determined that some areas were inappropriately recorded as having no relevant conditions. The licensee stated that the RVH was cleaned in excess of permissible prior to performing a subsequent VE in accordance with -3142.1(b)(1) of Code Case N-729-1. The licensee noted that the RVH areas inappropriately recorded as having no relevant conditions can no longer have a VE performed to meet requirements of Code Case N-729-1. The other options per -3142.1(c) are supplemental examination per -3142.2 or a repair/replacement activity per -3142.3.

In 2006, the licensee installed a new RVH with nozzles and partial penetration welds made with Alloy 690/52/152 material which is resistant to PWSCC.

Hardship Justification

The licensee stated that the RVH is currently located on the headstand. The radiation shield for the RVH currently does not allow for vendor tooling to access underneath without modification. The licensee noted that the Control Element Assembly (CEA) rack extensions are installed in the penetrations. At this time, the double CEA rack extensions in some CEDM penetration nozzles 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 RVH to be elevated from its current position. 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 which poses industrial safety risk and radiological concerns. These risks are exacerbated when lifting of heavy loads and working within limited spaces.

The licensee stated that it does not have the in-house expertise to conduct a volumetric examination as required by paragraph -3200(b) of Code Case N-729-1 during the current refueling outage in May 2015.

The licensee noted that there is no qualified ultrasonic (UT) examination technique at present for the ICI nozzles due to its thickness. The Electric Power Research Institute (EPRI), which has the expertise on UT, does not have a mock-up for ICIs with a 1-inch wall thickness. It is estimated for the completion of the design, manufacturing, and vendor qualification on a mock-up would require a lead time of approximately 9 to 12 months. Therefore, an emergent inspection of any ICI would require performance by manual surface examination. Manual surface examinations require significantly more time for an individual to be under the RVH increasing dose and inspection time.

As for the RVH vent line, the licensee explained that the vent line would be required to be manually tested by eddy current and/or dye penetrant examination techniques due to its 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. The licensee stated that the radiation dose estimate for completing supplemental UT/surface examinations of all RVH nozzles is 4.1 to 6.3 roentgen equivalent man (rem). The current dose estimate for the refueling outage is 67 rem.

The licensee stated that 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/replacement of the nozzles does not result in a compensating increase in the level of quality or safety.

3.5 Proposed Alternative

The licensee requested relief from the supplement examinations of the RVH nozzles as required by ASME Code Case N-729-1. In lieu of following Code Case N-729-1, the proposed alternative is not to perform the required supplemental examinations during the current refueling outage (FCR27) in May 2015. To support its request, the licensee (a) has demonstrated that the residue on the RVH nozzles is not from the inside of the reactor vessel and there is no evidence of a flaw in any of the RVH nozzles, (b) has performed a structural analysis to demonstrate the low probability of nozzle ejection, (c) has used PWSCC-resistant materials for the RVH penetration nozzles, (d) has a reliable leakage detection system, and (e) will perform compensatory examinations in the subsequent fuel cycle, cycle 28, under specified conditions and perform the required examinations per Code Case N-729-1 during the next refueling outage (FCR28).

3.6 Basis for Use

Inspection Results

On May 3, 2015, the licensee visually inspected the RVH and determined that no penetration had signs of degradation or evidence of nozzle leakage. The licensee identified boric acid residues on the RVH 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 isolation valve), and CEDM RC-10-24 mechanical connections. The licensee also detected component cooling water (CCW) leakage on the RVH that is attributed to a cracked return hose on CEDM RC-10-03. The licensee performed eddy current testing on the CEDM Seal Housing on RC-10-24 with satisfactory results and no detectable indications. The licensee also performed eddy current testing on eight other seal housings during the refueling outages with no detectable indications.

All the CEDM nozzle numbers are identified as: 1, 2, 3, 4, 5, 6, 7, 8, 10, 11, 12, 14, 15, 16, 17, 18, 19, 20, 21, 22, 23, 24, 25, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40, and 41. The nozzle numbers that are underlined are the CEDM nozzles that the licensee dispositioned as having "no relevant condition" during the initial evaluation but later changed to having a "relevant condition" after the re-evaluation. The nozzle numbers that are not underlined are the nozzles that the licensee dispositioned as having a relevant condition during the initial evaluation. Nozzle numbers 9 and 13 do not exist on the RVH.

The licensee did not identify a relevant condition at all six ICI nozzles during the initial evaluation but later classified all six ICI nozzle as having a relevant condition during re-evaluation. The six ICI nozzles are 42, 43, 44, 45, 46, and 47. However, the licensee noted that the ICI nozzles do not show any conditions indicative of possible nozzle leakage.

The licensee did not identify a relevant condition at the vent pipe nozzle (RC-100) during the initial evaluation but later classified the vent pipe nozzle as having a relevant condition during the re-evaluation.

Evaluation of Boric Acid Residue

The licensee took samples around the base of the penetration nozzles and removed deposits around selected areas to determine whether the residue is from the inside of the reactor vessel through a degraded CEDM nozzle or from the leaking mechanical connections located above the RVH. The licensee analyzed the presence of boron, pH, tolyltriazole (CorrShield NT 4207) and the isotopic ratio of Cobalt (Co)-58 and Co-60. Boric acid, used for reactivity control in the reactor coolant system (RCS), and the CCW corrosion inhibitor additive, CorrShield NT4204, both contain boron. The boric acid in the RCS contains elemental boron while the CCW additive contains boron as borate (i.e., sodium borate), which is not as soluble as elemental boron.

The licensee stated that its chemical analysis showed that the boric acid residue material on the RVH is inconsistent with RCS coolant chemistry. The licensee explained that the chemical analyses performed for boron cannot differentiate between boron and borate. Both analysis results are calculated in units of parts per million (ppm) boron. However, historically analytical results of known CCW deposits have shown boron up to approximately 200 ppm while analysis of known boric acid samples in RCS coolant range up to approximately 900 ppm and much greater.

The pH is another indicator of the type of boron (elemental boron or borate). The pH of samples with only boric acid are typically pH less than (<) 6. The licensee stated that all samples that had a sufficient quantity of deposits had pH values greater than (>) 9, which would be improbable if boric acid were the predominate species. The licensee further stated that the high pH (>9.0) indicates that the majority of the deposit was CCW-related.

The licensee performed an additional chemical analysis to identify tolyltriazole, a key compound of the CCW additive, CorrShield NT4204, on samples with sufficient quantity to conduct this analysis. These samples did identify tolyltriazole in them, indicating that CCW was a key contributor to the deposit. Based on the analysis performed, the licensee could not conclusively determine that the analyzed boron was a result of minor contamination from leakage of RC-100 packing and ICI Port #44 mechanical connection. The licensee stated that, however, the presence of tolyltriazole, combined with low levels of boron, is consistent with previously analyzed CCW leakage samples. The licensee noted that the sample shows that the predominant plant species in the deposits is not boric acid. If present, it would have to be a minor constituent.

The licensee noted that smear samples conducted on ICI nozzle penetrations had the following results:

(a) Visual evidence of the corrosion products hematite and magnetite. The smears were not analyzed for iron, since they would have to be dissolved, and a decision was made to maintain them.

(b) The licensee calculated short-term age using the ratio of the corrosion product isotopes, Nb-95 and Zr-95. The calculations yielded sample deposit ages in the range of 33- 82 days with an average of 61 days.

(c) The licensee determined that sample deposit ages ranging from 267- 547 days with an average of 379 days based on the Co-60/Co-58 ratios.

(d) The licensee determined that the sample deposit has a long-term age of between 2-4 years based on Co-60/Mn-54 ratios. The licensee further stated that intermediate and short-term aging data dictate some leakage from this cycle, probably from the last half to last third of cycle, is present. Long-term age analysis and high Cs-137 concentration dictate that longer term leakage has impacted the deposits/smears as well.

The licensee confirmed the source of the boron in the isotopic ratio of Co-58/60. The Co-58/60 ratio in the RCS liquid during this operating cycle was approximately 10. The isotopic ratios of the samples were in the range of 0.1 to 0.3, typical of the general contamination found in containment and provide additional confirmation that there is no evidence of fresh RCS leakage.

The licensee concluded that the chemical analysis of the sample shows that the predominate plant species in the deposits originate in the CCW, not from the coolant inside of the reactor vessel and that none of the RVH nozzles are degraded.

Examinations

In the construction of the replacement RVH in the late spring 2005, the vendor, Mitsubishi Heavy Industries (MHI), performed progressive dye penetrant examinations on the J-groove welds at CEDM during the welding process after the first pass, 25 percent, 50 percent, 75 percent, and 100 percent completion. The ICI penetrations were dye penetrant tested after the first pass, approximately every 1/7th of the weld thickness up to 100 percent completion and tests were documented as "Acceptable."

MHI also performed ultrasonic examination as part of RVH fabrication. MHI detected acceptable indications in various RVH nozzles as presented in licensee's May 13, 2015, response to NRC's request for additional information dated May 11, 2015 (ADAMS Accession No. ML15131A425). However, MHI identified an unacceptable UT indication on Nozzle #42 and dispositioned it as "Use-As-Is" in the post-weld ultrasonic inspection in July 2005. MHI attributed the cause of the unacceptable indication to oxide inclusion and eliminated other possible causes such as hot cracking, lack of fusion, or porosity. MHI stated that the unacceptable UT indication can be treated as a small group of indications. MHI further stated that if it is considered to be one large flaw and relevant for preservice inspection (PSI) or ISI activities, the flaw is still acceptable under ASME Code, Section XI acceptance standards. This includes extrapolating 40 years of operation where fatigue growth of the flaw would have to be considered.

The licensee stated that the RVH was hydrostatically tested on October 11, 2005. The test pressure of 3147 pounds per square inch gauge (psig) was held for 18 minutes and documented as "Acceptable."

The licensee performed the surface examinations of the replacement RVH and nozzles as part of the PSI in November 2005 after the hydrostatic test in accordance with the First Revised NRC Order EA-03-009, "Issuance of Order Establishing Interim Inspection Requirements for Reactor Pressure Vessel Heads at Pressurized Water Reactors," dated February 11, 2003

(ADAMS Accession No. ML030380470), MRP-2003-13,[1] and ASME Code, Section XI and documented as "Acceptable." The licensee also examined the RVH using ET/UT techniques in accordance with the EPRI Materials Reliability Program (MRP) demonstrated ET/UT techniques applicable at the time. The licensee also performed eddy current, ultrasonic, and manual dye penetrant testing on all penetrations and J-groove welds with no recordable indications found.

The last time the licensee performed a bare-metal visual inspection was on April 20, 2011, which was within the required every third refueling outage inspection interval after the RVH replacement in the fall of 2006. The licensee reported minor boric acid present at penetrations with the exception of CEDM penetrations 4, 5, 21, 25, and 28 and ICI penetrations 43, 44, 45, and 46. Boric acid deposits were attributed to a maintenance-induced RCS spill on April 15, 2011. An approximate 5 gallon spill of refueling boron concentration water flowed down to the RVH through the head insulation. The report concludes that the lack of any visible corrosion products in the boric acid deposits also was in line with the deposits resulting from the recent leakage. The licensee stated that the deposits found on the RVH nozzle annuli were in the bottom half of the annuli, which gave further evidence to support that the spillage came from on top of the RVH. The licensee did not report any pressure boundary leakage on the RVH.

For future bare-metal VE, the licensee stated that it will follow ASME Code Case N-729-1, paragraph -3141, which specifies acceptance criteria and evaluation of relevant indications, and paragraph -3142.1(b)(2), which specifies how the boric acid residue shall be removed and subsequent examinations.

To evaluate the relevant condition, the licensee stated that it will follow ASME Code Case N-729-1, paragraph -3142.1, which specifies that a relevant condition shall be unacceptable for continued service until the requirements of paragraphs -3142.1(b)(1), (b)(2), and (c) are met, which require specific evaluation of relevant indications.

In terms of inspection frequency, the licensee stated that it will follow ASME Code Case N-729-1, Table 1, to perform VE of RVH nozzles and J-groove welds every third refueling outage or 5 calendar years, whichever is less. The licensee will also perform volumetric or surface examination of all RVH nozzles with J-groove welds not to exceed one inspection interval (nominally 10 calendar years).

<u>Leakage Monitoring</u>

In its letter dated May 13, 2015, the licensee stated that the RCS leakage limits are based on Technical Specification section 2.1.4, which specifies 1 gallon per minute (gpm) for unidentified leakage, 10 gpm for identified leakage, and 150 gallons per day primary-to-secondary leakage through any one steam generator.

The licensee stated that these leakage limits by themselves cannot physically identify the difference between a leak at a nozzle penetration or weld versus a leak at any other location in the chemical and volume control system (CVCS) or RCS system, especially given the low

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1 "Pre-Service Inspection Guidance for New Reactor Pressure Vessel Heads," Letter from L. N. Hartz (Chair, MRP Senior Representatives; Dominion Generation) to MRP Utility Members, MRP 2003-013, dated June 26, 2003.

operational levels of past cycles. However, the licensee stated that it can apply analysis and inspection at these limits to positively identify the likelihood of a leak on the RVH and the necessity of performing an inspection of the RVH nozzles. The licensee noted that analysis can be quickly applied to known points of possible leakage within the monitored systems to eliminate possible locations and identify the possibility of an RVH leak. The licensee explained that during normal and the low action level, it can monitor parameters in addition to the leak rate such as containment sump level and temperature, spent regenerent tank levels, auxiliary building sump level, volume control tank level, pressurizer quench tank level and temperature, reactor coolant drain tank level, safety injection tank levels, pressurizer pressure or level, RCS temperature, primary-to-secondary leak rate chemistry, radiation monitors, CEDM seal temperatures, and charging pump packing leakage. The licensee noted that there is no airborne radiation monitoring equipment in the vicinity of the RVH; inlets to these process monitors are located further towards the outer edge of containment.

The licensee stated that the RCS Unknown Leakage action levels and their required actions are as follows.

Action Level 1 is at a single 7-day rolling average for the RCS unknown leak rate, which exceeds 0.1 gpm, or nine consecutive daily values greater than the baseline leakage. This is often entered into based on higher standard deviations in testing for a given period, or sudden degradations in equipment that cause a high leak rate before repairs can be enacted, such as charging pump packing failure or valve leak by. Required actions are to document the event in a corrective report, confirm the indications and check for equipment manipulations that may render the data invalid, then evaluate the trends of monitored parameters and look for abnormal trends to correct.

Action Level 2 is at either two consecutive daily values of unidentified RCS leakage greater than 0.15 gpm, or two of three consecutive daily unidentified RCS leak rate values greater than the baseline plus two standard deviation values. Required actions are to perform a confirmatory leak test and if confirmed then perform all Action Level 1 required actions. If evidence suggests the leakage is inside containment, visual inspections of accessible locations is required. Samples of various drains and other locations are to be taken for analysis. If the leakage cannot be located but evidence still supports a location inside containment, then remote inspections of inaccessible locations and placement of local filters to obtain airborne radiation data is to be recommended.

Action Level 3 is at one daily RCS unidentified leakage value greater than 0.3 gpm, or greater than the baseline plus 3 standard deviation values. This action level is almost never entered due to its high statistical significance. It was not entered during Cycle 27. All Action Level 1 and 2 actions are required prior to Action Level 3 is entered, and more frequent samples. If data suggests the increased leakage is inside the containment, the placement of air filters to check local airborne radiation levels and remote inspections of inaccessible locations is required. Action Level 3 would be the level at which detailed inspections of the RVH will be required if the source of the leakage cannot be identified elsewhere. If remote inspections still do not identify leakage, the licensee will schedule a down power to allow for detailed examinations of inaccessible locations.

The licensee stated that it will continue to monitor 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 the licensee's plant procedure, 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," March 2003 (ADAMS Accession No. ML041190170).

The licensee noted that 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 CVCS valves. The licensee repaired these valves during refueling outage 26 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.

Structural Analysis

The licensee evaluated the potential of nozzle ejection using Table 3-1 of EPRI Materials Reliability Program report MRP-395, "Materials Reliability Program: Reevaluation of Technical Basis for Inspection of Alloy 600 PWR Reactor Vessel Top Head Nozzles (MRP-395). EPRI, Palo Alto, CA: 2014, 3002003099." This table shows the results of several representative calculations for the time for growth of an initial 30 degree through-wall circumferential flaw to reach a circumferential extent of 300 degrees. These calculations are for Alloy 600 nozzles, and thus they take no credit for the slower crack growth rate of Alloy 690/52/152 materials which are used in the replacement RVH. In addition, these calculation results reflect application of a factor of 2 on the PWSCC crack growth rate to account for the potential effect of chemical concentration on the nozzle outside diameter. The licensee stated that because the calculation cases in Table 3-1 of MRP-395 are representative of top head nozzle geometries in the U.S. PWR fleet, the minimum calculated growth time from this table is expected to be bounding for the FCS top head nozzle geometries. For a head temperature of 605 °F, the lower bound time for circumferential crack growth in MRP-395 is 7.4 EFPY. At the RVH temperature for FCS of 588 °F, the crack growth time is 11.4 EFPY for a 30-degree circumference flaw to reach to 300 degree circumference flaw. The total operating time for the replacement RVH from the time of replacement in 2006 projected until the time of the next refueling outage scheduled to begin the fall 2016 is 6.56 EFPY. The licensee stated that a period of 11.4 EFPY is greater than 6.56 EFPY; therefore, it is not likely that nozzle ejection will occur between now and the next refueling outage.

Repairs

The licensee stated that it will repair RC-100, ICI Port #44 flange, RC-10-03, RC-10-12, RC-10-14, RC-10-24, RC-10-27, RC-10-35, and RC-10-39 during the current refueling outage. The licensee will also perform a surveillance test based on its plant Procedure OP-ST-RC-3007, "Periodic Reactor Coolant System Integrity Test," which specifies inspection areas inside the containment including the ICI ports, CEDM housings, and RC-100. This surveillance test will verify the correction of the identified leaks and confirm the absence of new leaks.

Material Selection

The licensee stated that the resistance of Alloy 690/52/152 material 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 volumetric or surface examinations performed in accordance with ASME Code Case N-729-1 on some replacement RVH currently operating in the U.S. fleet. The licensee stated that the 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.

Commitments and Compensatory Measures

In its letter dated May 16, 2015, the licensee made the following regulatory commitments:

(a) Use administrative controls such that at an unidentified leak rate increase of greater than 0.1 gpm above a stable baseline, actions will be taken to identify the source of the leakage. If the source is not identified within 24 hours, actions will be taken to shut the plant down during cycle 28.

(b) Perform a bare-metal visual inspection on the upper RVH in accordance with the criteria contained in ASME Code Case N-729-1 on the first Cold Shutdown of greater than 72 hours that occurs after at least 4 months of operation during cycle 28.

(c) Conduct UT/surface examinations of all RVH nozzles at the next refueling outage.

In addition, in its letter dated May 13, 2015, in response to NRC staff's question number 13, the licensee stated that it will perform a bare-metal visual inspection in the next refueling cycle FCR28. The licensee further stated that FCS is currently scheduled to perform a volumetric examination in the next refueling outage in accordance with the ASME Code. The licensee noted that it will perform a cause analysis to develop corrective actions as a result of the FCR27 inspection conducted in May 2015. Subsequently, the licensee will implement the corrective actions to ensure compliance with all code requirements.

3.6 Duration of Proposed Alternative

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

4.0 NRC STAFF EVALUATION

The NRC staff had concerns regarding the method the licensee used to remove the boric acid residue from the RVH. The licensee did not initially follow the requirements of ASME Code

Case N-729-1, as conditioned, to investigate all relevant conditions prior to performing aggressive head cleaning techniques. As such, the licensee may have unnecessarily removed the residue on the RVH nozzles during its cleaning, and thereby may have destroyed evidence to determine the source of the leakage. When NRC staff questions arose as to whether the residue came from degraded nozzles rather than from leaking mechanical connections above the RVH, the licensee was unable to fully address these issues. The NRC staff recognizes the licensee's intent to clean the RVH. However, the NRC staff notes that the current regulatory requirements of ASME Code Case N-729-1, as conditioned, prescribe the steps necessary to address relevant conditions before aggressive cleaning begins.

The NRC staff notes that the purpose of the bare-metal visual inspection requirement of ASME Code Case N-729-1, as conditioned, is to identify any indications of leakage or corrosion on the RVH surface. Operational experience has shown that indications of leakage from the CEDM or other nozzles and associated welds can be very small residue of boric acid. As such, when leakage from sources above the RVH fall down and cover the nozzle to RVH interface, using the proper inspection and cleaning steps is essential in order to verify that all of the deposit came from above the RVH. In cases where this inspection and cleaning cannot make this justification, ASME Code Case N-729-1, as conditioned, requires a volumetric and surface inspection of each particular penetration nozzle and weld for which a relevant condition could not be resolved.

The NRC staff concludes that the licensee's final inspection results show that all RVH nozzles have a relevant condition and is, therefore, acceptable. This is a conservative approach as opposed to some nozzles showed no relevant condition because a nozzle with a relevant condition requires supplement examinations per ASME Code Case N-729-1.

However, the licensee was not prepared to perform the supplemental examinations remotely during the refueling outage in May 2015. Additionally, the remote inspection methods were not qualified for the ICI nozzles and not developed for the vent line or J-groove welds. As such, only manual inspection techniques were available to complete a large portion of the supplemental examinations. These manual inspections would have been required in a high radiation dose area. The licensee estimated that the supplemental examinations would require approximately 4 to 6 rem in additional radiological dose during the current refueling outage, an increase of nearly 10 percent in radiological dose for this one activity. The NRC staff concludes that the radiological dose is a hardship. As such, the NRC staff concludes that a plant -specific stress analysis to demonstrate reasonable assurance of RVH nozzle structural integrity and compensatory measures could be used to provide a basis for inspection relief because the inspection requirements of ASME Code Case N-729-1 do present a hardship. Therefore, the NRC staff reviewed the licensee's proposed alternative under the requirements of 10 CFR 50.55a(z)(2), which states that,

> Compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

The regulations in 10 CFR 50.55a(g)(6)(ii)(D) requires the implementation of ASME Code Case N-729-1 with certain conditions to ensure reasonable assurance of the structural integrity and detect any leakage as soon as reasonably possible for all of the RVHs of U.S. PWRs. The

NRC staff notes that PWSCC, the active degradation mechanism, can challenge the structural integrity of the RVH in two ways. First, a PWSCC crack could grow circumferentially through the susceptible nozzle material to an extent, approximately 300 degrees, to cause a nozzle to be ejected. Second, leakage could cause significant boric acid corrosion of the low alloy steel head material causing a potential rupture in the RVH itself.

In order to address the issue of nozzle ejection, the licensee projected that a postulated through wall circumferential flaw with an initial length of 30 degrees in circumference would grow to 300 degrees in circumference in 11.4 EFPY. The licensee conservatively did not take credit for the less susceptible nozzle material in the replacement RVH. The NRC staff performed an independent calculation and verified that 11.4 EFPY is acceptable. Since the licensee will have only operated the plant, with the replacement RVH, for 10 years (~6.6 EFPY) by the time of the next refueling outage in the fall of 2016, the NRC staff concludes that the likelihood of RVH nozzle ejection prior to the 2016 refueling outage is unlikely. Based on the above, the NRC staff concludes that the licensee has provided sufficient basis to provide reasonable assurance of the structural integrity of the RVH in regards to nozzle ejection until the fall 2016 refueling outage, FCR28.

In order to address the issue of boric acid corrosion, the licensee could not rule out the possibility of leakage from a RVH nozzle or weld, and therefore the potential of boric acid corrosion to begin over the next operating cycle. However, the licensee did note that no significant corrosion was identified of the cleaned RVH during the spring 2015 refueling outage, and that leakage rates of greater than 0.1 gpm would be required before boric acid corrosion of the upper head would challenge the structural integrity of the RVH. In order to address the leakage rate issue, the licensee included a requirement to shut down the plant if a leakage source could not be identified after 24 hours of unidentified reactor coolant leakage rates increasing to 0.1 gpm above the baseline. Additionally, the licensee was able to provide sufficient information to demonstrate that the leakage on the RVH, in all the areas identified, could have come from sources above the head. Hence, the licensee determined that none of the relevant conditions of RVH leakage came from inside of the reactor vessel. Also, the licensee's chemical analysis showed that each deposit could have come from locations other than the inside of the reactor vessel. The NRC staff reviewed each of the licensee's actions, and concludes, in combination, they provide reasonable assurance of the structural integrity of the RVH for any potential leakage that could cause significant boric acid corrosion until the fall 2016 refueling outage, FCR28.

As a compensating measure, the licensee stated in RR-14 that if an outage occurred after at least 4 months of operation that lasts greater than 72 hours under cold shutdown conditions, the licensee would perform a bare-metal VE of the upper RVH during fuel cycle 28. The licensee will also be performing a bare-metal VE and ultrasonic/surface examinations of all RVH nozzles in accordance with ASME Code Case N-729-1, as conditioned, during the fall 2016 refueling outage, FCR28. The licensee will implement a stringent leak rate criteria and action plan during fuel cycle 28. Given these steps, the NRC staff concludes that the licensee has provided an inspection plan that will identify leakage as soon as reasonably available, and therefore meets the intent of the original regulation for leakage detection.

In summary, the NRC staff concludes that:

1. The licensee has demonstrated that the nozzle ejection and RVH degradation are not likely in the next fuel cycle.

2. The licensee's bare-metal VEs did not identify any areas of significant corrosion.

3. The licensee has demonstrated that there was an alternate possible source other than nozzle leakage of the relevant condition for each of the nozzles for which relief is requested.

4. The licensee's chemistry analysis provided additional supporting information for leakage sources other than possible nozzle leakage.

5. During fuel cycle 28, the licensee will use administrative controls such that at an unidentified leak rate increase of greater than 0.1 gpm above stable baseline, actions will be taken to identify the source of leakage. If the source is not identified within 24 hours, actions will be taken to shut down the plant.

6. During fuel cycle 28, the licensee will perform a bare-metal VE of all reactor vessel head nozzles in accordance with ASME Code Case N-729-1, as conditioned, on the first cold shutdown of greater than 72 hours that occurs after at least 4 months of operation.

7. During the fall 2016 refueling outage (FCR28), the licensee will perform a bare-metal VE and ultrasonic/surface examinations of all RVH nozzles in accordance with ASME Code Case N-729-1, as conditioned.

Based on the above plant-specific analysis and compensating measures, the NRC staff has determined that the licensee's proposed alternative provides reasonable assurance that the structural integrity of the RVH and attached nozzles will be maintained until the next refueling outage, which is scheduled for fall 2016. The NRC staff concludes that given the potential significant radiological dose associated with performing required inspections during the 2015 refueling outage, compliance with the requirements of ASME Code Case N-729-1, as conditioned, would result in hardship without a compensating increase in the level of quality and safety.

5.0 CONCLUSION

The NRC staff concludes that RR-14 will provide reasonable assurance of the structural integrity of the RVH and attached nozzles and that complying with the specified inspection in accordance with ASME Code Case N-729-1, as conditioned, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2) and is in compliance with the requirements of ASME Code Case N-729-1, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). Therefore, the NRC authorizes the use of RR-14 at FCS until the end of operating cycle 28, which is currently scheduled for fall 2016, or until a degraded RVH nozzle is detected.

All other requirements of ASME Code, Section XI, and 10 CFR 50.55a(g)(6)(ii)(D) for which relief was not specifically requested and authorized by the NRC staff remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: J. Tsao, NRR/DE/EPNB

Date: August 21, 2015

The NRC staff provided verbal authorization for RR-14 during a teleconference with your staff on May 17, 2015. If you have any questions, please contact Fred Lyon at 301-415-2296 or via e-mail at Fred.Lyon@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-285

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