Entergy

Entergy Operations, Inc.
1340 Echelon Parkway
Jackson, MS 39213
Tel 601-368-5138

Ron Gaston
Director, Nuclear Licensing

10 CFR 50.55a(z)(2)

1CAN101905

October 31, 2019

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

SUBJECT: Request for Relief related to American Society of Mechanical Engineers (ASME) Code Case N-729-4 Supplemental Examination Requirements ANO1-ISI-033

Arkansas Nuclear One, Unit 1
NRC Docket No. 50-313
Renewed Facility Operating License No. DPR-51

Pursuant to 10 CFR 50.55a(z)(2), Entergy Operations, Inc. (Entergy) requests approval of the enclosed request for relief due to the hardship of performing supplemental examinations of the Arkansas Nuclear One, Unit 1 (ANO-1) reactor vessel closure head (RVCH) in accordance with Code Case N-729-4 as conditioned by 10 CFR 50.55a. The enclosure contains the affected components, applicable American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code Case requirements, basis for request, and a proposed inspection to be performed during the next refueling outage (1R29) currently scheduled during the first semester of 2021.

Entergy scheduled the ANO-1 RVCH visual examination (VE) in accordance with the requirements in Code Case N-729-4, Table-1, to be performed during the current refueling outage, 1R28. It was discovered that a previously identified Intermediate Cooling Water (ICW) system leak following refueling outage 1R27 had precipitated sodium molybdate onto the ANO-1 RVCH and was masking approximately 57% of the nozzles examination surface.

The relevant conditions identified in the "as-found" VE are not considered to be indicative of Reactor Coolant System (RCS) leakage based upon the analysis of the "as-left" VE examination, known ICW leak following 1R27, primary water stress corrosion cracking resistant material of the ANO-1 replacement RVCH, crack growth initiation studies of Alloy 690 material, and chemical analysis of the residue. However, it could not be absolutely refuted that the build-up of sodium molybdate was not masking relevant conditions indicative of RCS leakage from the Control Rod Drive Mechanism nozzles prior to cleaning the RVCH.

There is one new regulatory commitment that is included in Attachment 2 of the enclosure.

This request is similar to that submitted by the Palo Verde Nuclear Generating Station (Reference 1), as approved by the NRC via Reference 2.

Approval of the proposed relief is requested prior to initial entry into startup (Mode 2) following refueling outage 1R28 (currently scheduled to occur on or about November 15, 2019).

If there are any questions or if additional information is needed, please contact Tim Arnold, Manager, Regulatory Assurance, at (479) 858-7826.

Respectfully,

ORIGINAL SIGNED BY RON GASTON

Ron Gaston

RWG/dbb

Enclosure: Request for Relief – ANO1-ISI-033

Attachments to Enclosure:

1. As Found Obstructed Annulus and Reactor Vessel Head Sample Locations
2. List of Regulatory Commitments

REFERENCES:

1. Palo Verde Nuclear Generating Station Unit 1 Letter to the Nuclear Regulatory Commission (NRC), *Relief Request 57 - Request for Alternative to American Society of Mechanical Engineers Code Case N-729-4 for Replacement Reactor Vessel Closure Head Penetration Nozzles* (ML17299B333), dated October 26, 2017.
2. NRC letter to PVNGS, *Palo Verde Nuclear Generating Station, Unit 1 – Relief Request No. 57 to Approve Alternate Requirements for the Reactor Pressure Vessel Head Nozzles to Perform a Bare Metal Examination per ASME Code Case N-729-4* (EPID L-2017-LLR-0132) (ML18040A331), dated February 16, 2018.

cc: NRC Region IV Regional Administrator

NRC Senior Resident Inspector – Arkansas Nuclear One

NRC Project Manager – Arkansas Nuclear One

Enclosure to

1CAN101905

REQUEST FOR RELIEF
ANO1-ISI-033

REQUEST FOR RELIEF
ANO1-ISI-033

Components / Numbers:	Reactor Vessel Closure Head (RVCH) Penetration Nozzles
Code Classes:	American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Class 1
References:	ASME Section XI 2007 Edition through 2008 Addenda. ASME Section XI, Division 1, Code Case N-729-4, Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1
Examination Category:	Table IWB-2500, Category B-P Table 1 of ASME Code Case N-729-4
Item Number(s)	B15.10 B4.30
Description:	Pressure retaining components RVCH with nozzles and partial-penetration welds of pressurized water stress corrosion cracking (PWSCC)-resistant materials
Unit / Inspection Interval Applicability:	Arkansas Nuclear One, Unit 1 (ANO-1) / Fifth 10-Year Inservice Inspection (ISI) Interval (May 31, 2017 to May 30, 2027)

I. APPLICABLE REQUIREMENTS

The fifth 10-year ISI interval Code of Record for ANO-1 is the 2007 Edition through the 2008 Addenda of ASME Code, Section XI, *Rules for Inservice Inspection of Nuclear Power Plant Components.*

Examinations of the RVCH penetration nozzles are performed in accordance with ASME Code Case N-729-4, *Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles having Pressure-Retaining Partial-Penetration Welds*, Section XI, Division 1 (Reference 1), as conditioned by 10 CFR 50.55a(g)(6)(ii)(D).

10 CFR 50.55a(g)(6)(ii)(D) requires, in part, that licensees of pressurized water reactors shall augment the inservice inspection program with ASME Code Case N-729-4 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of this section.

10 CFR 50.55a(g)(6)(ii)(D)(1) requires:

(D) Augmented ISI requirements: Reactor vessel head inspections – (1) Implementation. Holders of operating licenses or combined licenses for pressurized-water reactors as of or after August 17, 2017 shall implement the requirements of ASME BPV Code Case N-729-4 instead of ASME BPV Code Case N-729-1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of this section, by the first refueling outage starting after August 17, 2017.

Table 1 of Code Case N-729-4 states that a bare metal visual examination is required every third refueling outage or 5 calendar years, whichever is less. Note 1 of Code Case N-729-4 states, regarding inservice visual examinations (VE):

(1) The VE shall consist of the following:

(a) A direct examination of the bare-metal surface of the entire outer surface of the head, including essentially 100% of the intersection of each nozzle with the head.

Paragraph 3141 of Code Case N-729-4 states, regarding inservice VEs:

(a) The VE required by -2500 and performed in accordance with IWA-2200 and the additional requirements of this case shall be evaluated by comparing the examination results with the acceptance standards specified in -3142.1.

(b) Acceptance of components for continued service shall be in accordance with -3142.

(c) Relevant conditions for the purpose of the VE shall include evidence of reactor coolant leakage, such as corrosion, boric acid deposits, and discoloration.

Paragraph 3142.1, *Acceptance by VE*, of Code Case N-729-4, states:

(a) A component whose VE confirms the absence of relevant conditions shall be acceptable for continued service.

(b) A component whose VE detects a relevant condition shall be unacceptable for continued service until the requirements of (1), (2), and (c) below are met.

1. Components with relevant conditions require further evaluation. This evaluation shall include determination of the source of the leakage and correction of the source of leakage in accordance with -3142.3.

2. All 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 surfaces shall be performed, prior to return to service, and again in the subsequent refueling outage. Any degradation detected shall be evaluated to determine if any corrosion has impacted the structural integrity of the component. Corrosion that has reduced component wall thickness below design limits shall be resolved through repair / replacement activity in accordance with IWA-4000.

(c) 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, *Acceptance of Supplemental Examination*, of Code Case N-729-4, 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, *Acceptance by Corrective Measures or Repair/ Replacement Activity*, of Code Case N-729-4, 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), *Supplemental Examinations*, of Code Case N-729-4 states:

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 Mandatory 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.

II. REQUEST FOR RELIEF

Entergy Operations, Inc. (Entergy) requests relief due to the hardship of performing supplemental examinations of the ANO-1 RVCH in accordance with Code Case N-729-4.

Entergy will perform a VE on the bare metal surface of the ANO-1 RVCH during refueling outage 1R29 in accordance with the latest revision of Code Case N-729 endorsed in 10 CFR 50.55a. The examination will be conducted in accordance with Paragraph 3140 and the results will be evaluated in accordance with Paragraph 3142. Subsequent VE examinations will be conducted in accordance with Table 1 Item No. B4.30.

Attachment 2 of this enclosure provides the list of regulatory commitments associated with this request for relief.

III. BASIS FOR RELIEF

ANO-1 RVCH

During refueling outage 1R19 in the fall of 2005, Entergy replaced the original RVCH which contained PWSCC susceptible Alloy 600 nozzles and filler materials with a replacement RVCH containing control rod drive mechanism (CRDM) nozzles fabricated from non-PWSCC susceptible material UNS N06690 and joined by filler materials UNS N06052 and UNS W86152.

The RVCH CRDM nozzle housings are fabricated from Inconel SB-167 (Alloy 690) UNS N06690 and the CRDM nozzle housing adapter flanges are fabricated from A182 F-304. Weld materials ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152) were used to construct the CRDM nozzles and to attach the nozzles to the RVCH. The CRDM nozzle housing materials and applicable weld materials were procured and certified to ASME Section III, 1989 Edition, no Addenda, but have been reconciled to ASME Section III, 1998 Edition, with Addenda through 2000, to eliminate the requirement to use Code Cases, as would be required through the use of the 1989 Edition. There are no similarities that indicate any specific concern for elevated PWSCC susceptibility of the RVCH nozzles at ANO in comparison to other RVCHs with Alloy 690 nozzles (Reference 2).

Alloy 690

MRP-375 Information Regarding the Structural Adequacy of the RRVCH Alloy 690 Nozzles

Evaluations were performed and documented in Materials Reliability Program (MRP)-375 (Reference 3) to demonstrate the acceptability of extending the inspection intervals for ASME Code Case N-729-1, Item B4.30 and B4.40 components. Based on plant service experience, factor of improvement (FOI) studies using laboratory data, deterministic study results, and probabilistic study results, MRP-375 supported extended volumetric and surface inspection intervals for Item B4.40. Although no proposed change to the direct VE inspection interval for Item B4.30 was proposed at this time, the evaluations performed in MRP-375 supported an extension of the interval for direct VEs. Specifically, an extension of the interval for direct VE to six calendar years is shown to result in an acceptably small effect on the nozzle ejection frequency. This information documents the structural suitability of the RVCH for extended periods of time.

Per MRP-375, much of the laboratory data indicated an FOI of 100 for Alloy 690/52/152 versus Alloy 600/182/82 (for equivalent temperature and stress conditions) in terms of crack growth rates (CGR). In addition, laboratory and plant data demonstrate an FOI in excess of 20 in terms of the time to PWSCC initiation. The reduced susceptibility to PWSCC initiation and growth supports elimination of all volumetric as well as visual examinations for evidence of leakage throughout the plant service period.

As documented in MRP-375, the resistance of Alloy 690 and corresponding weld metals Alloy 52 and Alloy 152 is demonstrated by the lack of PWSCC indications reported in these materials, in up to 24 consecutive years of service for thousands of Alloy 690 steam generator tubes, and more than 22 consecutive 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 on the RVCH 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 reactor vessel heads currently operating in the United States nuclear power plant fleet. This data supports at least a factor of 5-20 of improvement in time to detectable PWSCC flaw initiation when compared to service experience of Alloy 600 in similar applications worldwide.

Background

Per ASME Code Case N-729-4 (Reference 1), the ANO-1 RVCH is required to be visually examined every third refueling outage or every five years, whichever is less. The previous code required VEs were performed in spring 2010 (1R22) and spring 2015 (1R25) utilizing qualified and approved techniques, procedures, and personnel demonstrated to the satisfaction of the third party Authorized Nuclear Inservice Inspector as required by IWA-2110. These examinations did not reveal relevant conditions requiring evaluation to the acceptance criteria of Code Case N-729 Paragraph 3142.

On May 5, 2018, during plant heat-up following refueling outage 1R27, an Intermediate Cooling Water (ICW) leak occurred at CRDM 14, Nozzle 51, when Operations aligned ICW for plant operation. The source of the leakage was identified to be a cross-threaded mechanical fastener on the ICW tubing identified after CRDM maintenance. The leakage was stopped and corrective measures to replace the fastener were successful. The ICW leakage above the RVCH lasted for approximately nine days before being detected and repaired on May 14, 2018.

1R28 N-729-4 VE

During refueling outage1R28, following the placement of the RVCH on the head stand, Entergy was required to perform the scheduled VE of the bare metal surface on the RVCH and nozzle penetrations to meet the requirements of Code Case N-729-4 and 10 CFR 50.55a, as follows:

a. Item No. B4.30 – describes the head with nozzles and partial-penetration welds of PWSCC-resistant materials.

b. Item No. B4.30, Figure 1 – defines the examination area which includes essentially 100% of the intersection of each nozzle with the head.

In preparation for performing the VE, the inspection doors around the RVCH were opened revealing a white residue covering a large area on top of the head surface and around numerous CRDM nozzles. This observation was entered into the ANO Corrective Action Program, noting that the residue could mask actual leakage from a nozzle, which may not be detectable during the VE.

In order to ascertain the extent and confirm the source of the residue, a visual inspection by the ANO-1 Reactor Coolant System (RCS) Engineer was performed. This inspection was conducted in part from the top of the Service Structure. The conclusions of this inspection confirmed that the cross-threaded mechanical connection discovered and corrected during plant start-up from 1R27 at CRDM 14 was the sole source of ICW leakage resulting in the deposit of the white residue inside the Service Structure and present on the surface of the RVCH.

Entergy determined that due to the ICW leak, the white residue was most likely a form of sodium molybdate used at ANO-1 in the ICW system. The following actions were taken to substantiate the conclusion that the residue resulted from ICW leakage:

a. Samples were obtained from the upper head area in five locations away from the nozzles prior to completion of the bare metal visual (BMV) crawler inspection. Note that care was taken to not disturb the areas around the nozzles to preserve the as-found condition.

b. Due to accessibility and substance quantity limitations, samples were obtained from the upper head area around 12 nozzles following completion of the BMV crawler inspection (see Attachment 1 for sampling locations). Other accessible locations did not contain a sufficient quantity of material for chemical analysis.

All samples were sent to Southwest Research Institute (SWRI) for comprehensive analysis. Based on the preliminary results from SWRI, the amount of sodium, molybdate, and boron present in the RVCH samples is consistent with the ratios present in sodium molybdate rather than RCS boron. Specifically, if RCS fluid was leaking onto the RVCH, the amount of boron present in the sample results would be much higher. In addition, molybdate and sodium, at these levels, would not have been expected to be present. The x-ray diffraction analysis was consistent with a sodium molybdate substance versus boric acid.

Additional borescope inspections by the RCS Engineer of the CRDM flanges were also performed. The borescope was guided horizontally (with a guide tube) through the lanes between the CRDM flanges via the 28 two-inch inspection ports at the CRDM flange interface elevation. The inspections allowed a near 100% visual examination of the CRDM flanges which revealed no apparent RCS leakage from the CRDM flanges during Cycle 28. Significant ICW residue was present on many CRDM flanges in the northeast quadrant below CRDM-14, but there was no apparent indication indicative of boric acid accumulation at the CRDM flange to nozzle interface.

Utilizing Entergy NDE procedure CEP-NDE-0955, *Visual Examination of Bare Metal Surfaces*, a remote video crawler was deployed by Entergy's vendor General Electric Information Technology (GEIT) from October 11 through October 15, 2019 which recorded video data to be utilized in documenting and analyzing the VE results. After review of the video records, Entergy Level III qualified personnel determined that relevant conditions were present on approximately 57% of the RVCH nozzles due to the white substance on the RVCH. Qualified and approved procedures, techniques, and personnel were employed in performing the inspection.

After the collection of residue material for chemistry analysis, Entergy aggressively cleaned the RVCH to remove the residue in preparation of performing an as-left baseline VE. The focus of this examination was to inspect for corrosion, discoloration, wastage, or staining of the low-alloy steel head adjacent to the nozzle penetrations. The reactor head accessible surfaces and 360° around each penetration was re-inspected. There was no discoloration, corrosion, wastage, or staining identified that is believed to represent active nozzle leakage noted. No relevant conditions were found. There was some residual staining on the head and several nozzles that did not impact the VE. The results of this examination will serve as a baseline for the examination to be performed in 1R29.

Based on the information above, Entergy believes that all reasonable efforts have been taken to ensure the structural integrity of the RVCH and CRDM nozzle penetrations. However, Entergy is unable to demonstrate that the ICW leakage was not masking leakage from one of the obstructed nozzles. Although the samples that were analyzed, the examinations performed,

and the previously identified ICW leakage following 1R27 all suggest that the residue is a result of the ICW leak, it could not absolutely be refuted that the relevant conditions identified in the as-found examination were not masking relevant conditions indicative of nozzle leakage.

Once it is determined that the possibility of nozzle leakage exists, Code Case N-729-4, Paragraph 3142.2, requires the nozzles with relevant conditions to undergo supplemental examination. In order to perform the supplemental examinations in accordance with Code Case N-729-4, it will be necessary to mobilize qualified personnel to the site on an emergent basis. The time required for such mobilization is expected to be in the range of several weeks and is dependent on the availability of qualified personnel and equipment, the development of contracts, in-processing of personnel, and other factors, all of which would result in a significant extension of the current ANO-1 refueling outage with minimal, if any, nuclear safety benefit.

In addition, the supplemental examination requires access to the underside of the highly contaminated RVCH which would expose personnel to elevated dose rates. Because the inspection requires approximately 6-7 days to complete, the additional dose is estimated to be approximately 4.2 person-rem for the examinations. This does not account for setup and removal of equipment.

Access to the underside of the RVCH also requires the unit to remain with the Reactor Coolant System (RCS) open, delaying final closure and inspections of the RCS pressure boundary, reinstallation of missile shields and CRDM cooling apparatus, closeout of the Reactor Building, and testing of a number of systems and components required to be completed prior to restart of the unit.

Significantly extending the outage duration and increasing the personnel dose for performing supplemental examinations represents a hardship or unusual difficulty without a compensating increase in the level of quality and safety pursuant to 10CFR50.55a(z)(2).

IV. BASIS FOR HARDSHIP

Entergy has concluded that the subject supplemental examinations represents a hardship or unusual difficulty without a compensating increase in the level of quality and safety based on the information provided in Section III above.

Entergy will perform a BMV examination of the ANO-1 RVCH in the next refueling outage in accordance with the latest revision of Code Case N-729 endorsed in 10 CFR 50.55a. Entergy considers this to be a regulatory commitment (see Attachment 2 of this enclosure). The examination will be conducted in accordance with Paragraph 3140 and the results will be evaluated in accordance with Paragraph 3142. Subsequent VE examinations will be conducted in accordance with Table 1 Item No. B4.30.

V. PRECEDENT

This request is similar to that submitted by the Palo Verde Nuclear Generating Station (Reference 4), approved by the NRC via Reference 5.

The Palo Verde Nuclear Generating Station (PVNGS) Unit 1 RVCH was fabricated by Doosan Heavy Industries using Alloy 690 nozzle material produced by Doosan Heavy Industries per ASME SB-166. The nozzle J-groove welds were produced using Alloy 52 (ASME SFA-5.14 ERNiCrFe-7), Alloy 52 (ERNiCrFe-7A to ASME Code Case 2142-2 requirements), and Alloy 152 (ASME SFA-5.11 ENiCrFe-7) weld material. There are no similarities that indicate any specific concern for elevated PWSCC susceptibility of the RVCH nozzles at PVNGS in comparison to other RVCHs with Alloy 690 nozzles.

V. CONCLUSION

From 10 CFR 50.55a

(z) *Alternatives to codes and standards requirements*. Alternatives to the requirements of paragraphs (b) through (h) of this section or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. A proposed alternative must be submitted and authorized prior to implementation. The applicant or licensee must demonstrate that:

(1) *Acceptable level of quality and safety*. The proposed alternative would provide an acceptable level of quality and safety; or

(2) *Hardship without a compensating increase in quality and safety*. Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on foregoing discussion, Entergy has determined that the conditions of 10 CFR 50.55a(z)(2) are met in that performing the supplemental examinations subject to this relief request represents a hardship or unusual difficulty without a compensating increase in the level of quality and safety.

VI. REFERENCES:

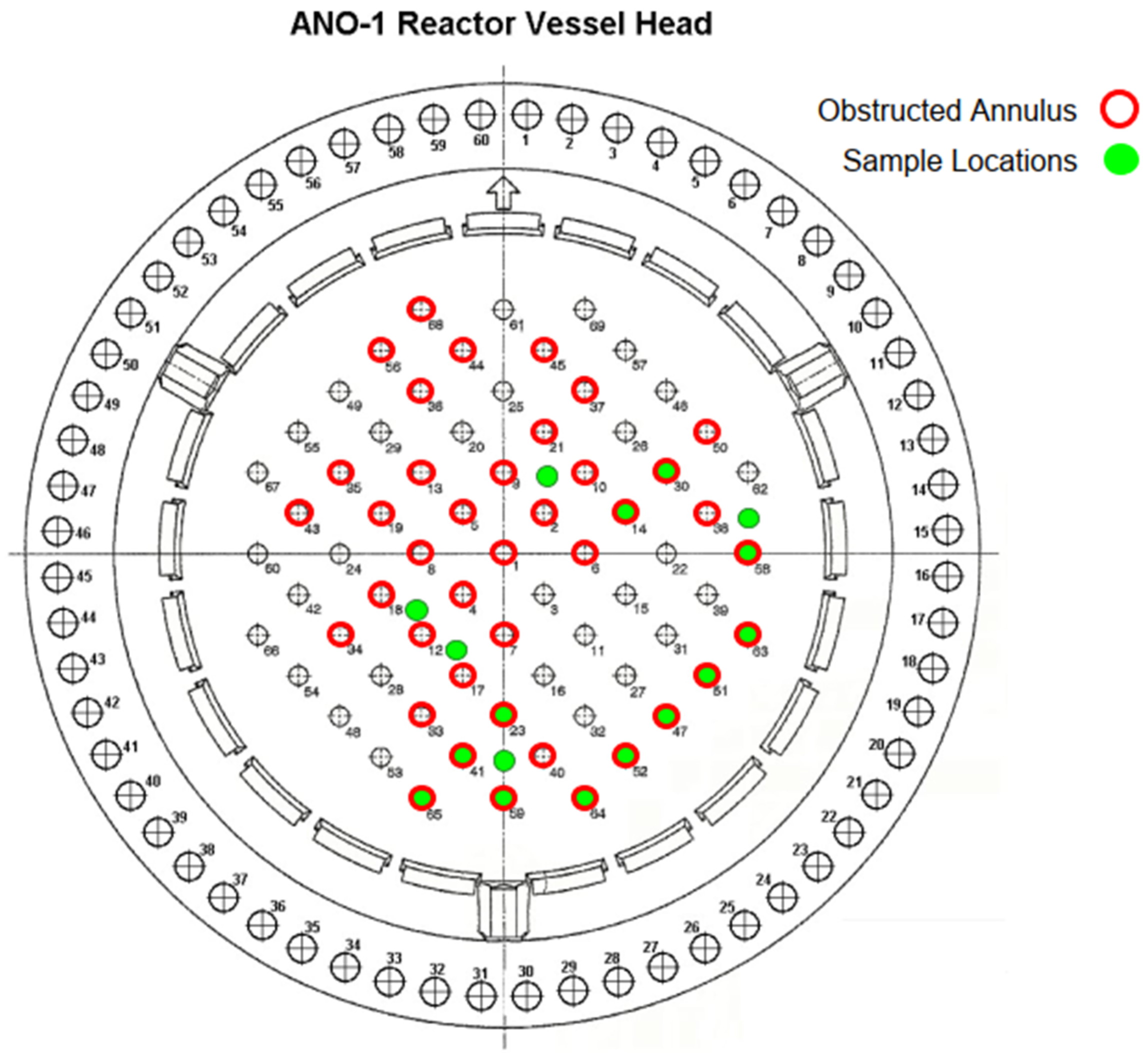
1. ASME BPV Code Case N-729-4, *Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1*, approval date June 22, 2012.
2. ER-ANO-2002-0638-000, *ANO-1 Reactor Vessel Closure Head (RVCH) Replacement*
3. Materials Reliability Program: *Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)*, EPRI, Palo Alto, CA: 2014.
4 Palo Verde Nuclear Generating Station (PVNGS) Unit 1 letter to the Nuclear Regulatory Commission (NRC), *Relief Request 57 - Request for Alternative to American Society of Mechanical Engineers Code Case N 729-4 for Replacement Reactor Vessel Closure Head Penetration Nozzles* (ML17299B333), dated October 26, 2017.
5. NRC letter to PVNGS, Unit 1, *Relief Request No. 57 to Approve Alternate Requirements for the Reactor Pressure Vessel Head Nozzles to Perform a Bare Metal Examination per ASME Code Case N-729-4 (EPID L-2017-LLR 0132)* (ML18040A331), dated February 16, 2018.

Enclosure, Attachment 1 to

1CAN101905

AS FOUND OBSTRUCTED ANNULUS AND REACTOR VESSEL HEAD SAMPLE LOCATIONS

As Found Obstructed Annulus and Reactor Vessel Head Sample Locations



Enclosure, Attachment 2 to

1CAN101905

LIST OF REGULATORY COMMITMENTS

LIST OF REGULATORY COMMITMENTS

The following table identifies those actions committed to by Entergy in this document. Any other statements in this submittal are provided for information purposes and are not considered to be regulatory commitments.

COMMITMENT	TYPE (check one)		SCHEDULED COMPLETION DATE
	ONE-TIME ACTION	CONTINUING COMPLIANCE	
Entergy will perform a BMV examination of the ANO-1 RVCH in the next refueling outage in accordance with the latest revision of Code Case N-729 endorsed in 10 CFR 50.55a.	✓		Prior to Completion of ANO-1 Refueling Outage 1R29