Nuclear Operating Company

South Texas Project Electric Generating Station P.O. Box 289 Wadsworth, Texas 77483

February 28, 2019
NOC-AE-19003620
10 CFR 50.55a

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

South Texas Project Unit 1 and 2
Docket No. STN 50-498 and STN 50-499
Proposed Alternative for Extension of Volumetric Examination Interval for Reactor Vessel Closure Head with Alloy 690 Nozzles in accordance with 10 CFR 50.55a(z)(1)
(Relief Request RR-ENG-3-23)

In accordance with the provisions of 10 CFR 50.55a(z)(1), STP Nuclear Operating Company (STPNOC) requests approval for South Texas Project (STP) Unit 1 and Unit 2 to allow the use of an extended ISI interval from 10 years to nominally 17 years for the South Texas Project (STP) Units 1 and 2 Alloys 690/52/152 reactor vessel closure head penetrations.

STPNOC requests NRC review and approval of this alternative request by September 1, 2019, to support the use of the proposed alternative.

There are no commitments in this letter.

If there are any questions, please contact Craig Younger at 361-972-8186.

Michael Page
Michael Page
General Manager, Engineering

rjg

Enclosure:

Proposed Alternative for Extension of Volumetric Examination Interval for Reactor Vessel Closure Head with Alloy 690 Nozzles in accordance with 10 CFR 50.55a(z)(1) (Relief Request RR-ENG-3-23)

cc:

(paper copy)

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
1600 East Lamar Boulevard
Arlington, TX 76011-4511

Lisa M. Regner
Senior Project Manager
U.S. Nuclear Regulatory Commission
One White Flint North (O9E01)
11555 Rockville Pike
Rockville, MD 20852

NRC Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 289, Mail Code: MN116
Wadsworth, TX 77483

Enclosure

Proposed Alternative for Extension of Volumetric Examination Interval for Reactor Vessel Closure Head with Alloy 690 Nozzles in accordance with 10 CFR 50.55a(z)(1)
(Relief Request RR-ENG-3-23)

**Proposed Alternative for Extension of Volumetric Examination Interval for Reactor Vessel Closure Head with Alloy 690 Nozzles in accordance with 10 CFR 50.55a(z)(1)
(Relief Request RR-ENG-3-23)**

A. ASME Code Component(s) Affected

Component: Reactor Vessel Closure Head (RVCH) Nozzles

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)

Examination Category: Class 1 Pressurized-Water Reactor (PWR) Reactor Vessel Upper Head

Item Number: B4.40 Nozzles and partial-penetration welds of PWSCC-resistant materials in head

Description:
Control Rod/ Drive Mechanism (CRDM) Nozzles (57 penetrations), SB-167 UNS N06690 (Alloy 690TT), 4-inch OD

Core Exit Thermal Couple (4 penetrations), SB-167 UNS N06690 (Alloy 690TT), 4-inch OD

RV Water Level Instrumentation System (RVWLIS) (2 penetrations), SB-167 UNS N06690 (Alloy 690TT), 4-inch OD

RVCH Vent Nozzle (1 penetration), SB-167 UNS N06690 (Alloy 690TT), 1-inch OD

Penetration to Weld Material: ERNiCrFe-7 (UNS N06052), "Filler Metal 52"

RVCH – SA-508 Grade 3 Class1 (one-piece forging)

B. Applicable ASME Code Edition and Addenda

ASME Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2004 Edition.

Table 1 - Applicable ASME Code Edition and Addenda

Unit	Interval	Edition	Start	End
Unit 1	3	2004	September 25, 2010	September 24, 2020
Unit 2	3	2004	October 19, 2010	October 18, 2020
Unit 1	4	TBD	September 25, 2020	August 20, 2027
Unit 2	4	TBD	October 19, 2020	December 15, 2028

C. Applicable ASME Code Requirement

The Code of Federal Regulations (CFR) 10 CFR 50.55a(g)(6)(ii)(D)(1), requires (in part):

"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."

ASME Code Case N-729-4 (Reference 1), section -2410 specifies the Inspection Program requirements for the reactor vessel upper head and its penetrations (nozzles and partial-penetration welds). The basic inspection requirements of Code Case N-729-4, section -2410 as contained within Table 1 for partial-penetration welded Alloy 690 head penetration nozzles are as follows:

- Item B4.40: Volumetric or surface examination of all nozzles, not to exceed one inspection interval (nominally 10 calendar years) provided that flaws attributed to primary water stress corrosion cracking (PWSCC) have not been identified.

- Item B4.30: Direct visual examination (VE) of the outer surface of the head for evidence of leakage every third refueling outage or 5 calendar years, whichever is less.

The Item B4.40 volumetric and/or surface re-examination interval of ASME Code Case N 729-4 is identical to that of Code Case N-729-1, which was mandated by NRC prior to August 17, 2017 by 10CFR50.55a(g)(6)(ii)(D). The previous NRC conditions on N-729-1 and the current NRC conditions on N-729-4 in 10CFR50.55a(g)(6)(ii)(D) do not affect the re-examination interval required for Item B4.40.

<u>D. Reason for Relief from Code Requirements</u>

Code Case N-729-4 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) requires volumetric and/or surface examination of the RVCH penetration nozzles and associated welds no later than nominally 10 calendar years after the head was placed into service.

This examination schedule was intended to be conservative and subject to reassessment once additional laboratory data and plant experience on the performance of Alloy 690 and Alloy 52/152 weld metals became available. Using plant and laboratory data that has since become available, Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) Report: "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)" (Reference 2) was developed to support a technically based volumetric or surface re-examination interval using appropriate analytical tools. This technical basis demonstrates that the re-examination interval can be extended to at least a 20-year interval while maintaining an acceptable level of quality and safety.

The South Texas Project (STP) Units 1 and 2 RVCH penetration nozzles and associated welds are made from Alloys 690/52/152. As discussed in EPRI MRP Report: "Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386)" (Reference 3), that compared to Alloys 600/82/182, these materials have much greater PWSCC resistance. Excellent operating experience with no observations of PWSCC in almost 30 years of service supports the superiority of Alloy 690 relative to Alloy 600 in PWR primary water environments, as does extensive laboratory testing. Research documented in EPRI MRP-386 further demonstrates the much greater resistance of these replacement alloys to PWSCC as compared to Alloys 600/82/182 for the conditions relevant to partial-penetration welded nozzles.

The technical bases of MRP-375 and MRP-386 together demonstrate that the reexamination interval can be extended to a 20-year interval while maintaining an acceptable level of quality and safety. Therefore, STP Nuclear Operating Company (STPNOC) is requesting approval of this alternative to allow the use of an extended ISI interval from 10 years to nominally 17 years for the South Texas Project (STP) Units 1 and 2 Alloys 690/52/152 reactor vessel closure head penetrations.

E. Proposed Alternative and Basis for Use:

Pursuant to 10 CFR 50.55a(z)(1), STPNOC is requesting relief from the examination frequency requirements of Code Case N 729-4, Item B4.40 for performing volumetric and/or surface exams of the STP RVCH penetrations. Specifically, this would allow volumetric or surface examinations currently scheduled for the Spring of 2020 for STP Unit 1 and the Fall of 2019 for STP Unit 2 to be extended to the Spring of 2026 for STP Unit 1 and the Spring of 2027 for STP Unit 2 (nominally, 17 calendar years). This request applies to the Item B4.40 inspection frequencies only. The alternative method proposed provides an acceptable level of quality and safety for the examination frequency of the RVCHs for the reasons specified herein.

Application of this alternative is requested for a nominal 17-year period from the time the replacement STP Unit 1 and Unit 2 RPV heads were placed in service, see Table 2 below.

Table 2 – Proposed Dates of Service

	Placed in Service Date	Nominal End of the Alternative Date
STP Unit 1 Replacement RPV Head	November 18, 2009	April 12, 2026
STP Unit 2 Replacement RPV Head	May 2, 2010	April 12, 2027

This request to use this alternative applies only to the inspection frequencies for volumetric/surface examinations of the RVCH as the inspection techniques or other requirements may change with later editions of ASME Section XI and 10 CFR 50.55a.

Evaluations were performed to demonstrate the resistance of Alloys 690/52/152 to PWSCC under an EPRI MRP initiative provided in MRP-375. This report combines an assessment of the test data and operating experience developed since the technical basis for the 10-year interval of ASME Code Case N-729 (Revisions 1 through 4) were developed in 2004 with deterministic and probabilistic evaluations to assess the improved PWSCC resistance of Alloys 690/52/152 relative to Alloys 600/82/182. Additional research was recently performed under an EPRI MRP initiative provided in MRP-386. This report compiled over 530 Alloy 690 Crack Growth Rate (CGR) data points and over 130 Alloy 52/152 CGR data points from seven research laboratories further supporting the improved PWSCC resistance of Alloys 690/52/152.

Operating experience to date for replaced and repaired components using Alloys 690/52/152 have shown a proven record of resistance to PWSCC determined through numerous examinations in over 24 years of application. This includes steam generator tubes, pressurizers, and RVCHs. In particular, the Alloys 690/52/152 operating experience includes inservice volumetric/surface examinations performed in accordance with ASME Code Case N-729 on replacement heads. Some of these examined heads had continuous full power operating temperatures that approached 613°F. However, none of these examinations revealed PWSCC cracking and these examination results further support the low likelihood or potential for the RVCH to experience PWSCC during the extension period.

The evaluation performed in MRP-375 considers a simple Factor of Improvement (FOI) approach applied in a conservative manner to model the increased resistance of Alloys 690/52/152 compared to Alloys 600 and 182 at equivalent temperature and stress conditions. FOIs were estimated for the material improvements of Alloys 690/52/152 using an extensive database of material test data. Results for both crack initiation and crack growth conclude that there was a substantially higher

resistance to PWSCC than for Alloy 600 base material and Alloy 82/182 weld materials. Figures 3-2, 3-4, and 3-6 of MRP-375 provide crack growth rate data for Alloys 690/52/152 materials and heat affected zones with curves plotting FOIs of 1, 5, 10, and 20 on a statistical basis reflecting the material variability exhibited in MRP-55 (Reference 4) for Alloy 600 material and in MRP-115 (Reference 5) for Alloys 82/182/132 weld material. An FOI of 20 bounds most of the data plotted and an FOI of 10 bounds essentially all of the crack growth rate data. Table 3-6 of MRP-375 provides a summary of FOIs determined on the basis of crack growth rate and crack initiation data. For crack initiation, FOIs reported, although significant, are conservatively small because crack initiation of Alloys 690/52/152 was not observed during testing; instead, the initiation time was assumed to be equivalent to the test duration.

Additional Evaluations Performed under MRP-375
MRP-375 applied the FOI results to perform a combination of deterministic and probabilistic evaluations to establish an appropriately conservative inspection interval for Alloy 690 RVCHs. The deterministic technical basis applies industry-standard crack growth calculation procedures to predict time to certain adverse conditions under various conservative assumptions. A probabilistic evaluation is then applied to make predictions for leakage and ejection risk, generally using best-estimate inputs and assumptions with uncertainties treated using statistical distributions.

The deterministic crack growth evaluation provides a precursor to the probabilistic evaluation to directly illustrate the relationship between the improved PWSCC growth resistance of Alloys 690/52/152 and the average time to certain adverse conditions. These evaluations apply conservative crack growth rate predictions and the assumption of an existing flaw, which is replaced with a PWSCC initiation model for probabilistic evaluation. The evaluations provide a reasonable lower bound on the average time to adverse conditions, from which a conservative inspection interval may be recommended. This evaluation draws from various EPRI MRP and industry documents that evaluate, for Alloys 600/82/182, the time from a detectable flaw being created to leakage occurring and from a leaking flaw to the time that net section collapse (nozzle ejection) would be predicted to occur. As stated in MRP-375:

> For different analyses and different crack types on an Alloy 690 RVCH, the conservative time between detectable flaw size and leakage varies between 23 and 77 EFPY [effective full power years] at 613°F (or between RIY [Re-Inspection Years] = 31 and 106). This result is supportive of an extension of the UT [ultrasonic] inspection interval to 20 calendar years. The RIY parameter adjusts the EFPYs of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth.
>
> The conservative time between evident leakage and risk of net section collapse varies between 121 and 320 EFPY at 613°F (i.e., between RIY = 167 and 441) for the Alloy 690 RPVH.

These results indicate that more than 20 years is required for leakage to occur and that more than 120 years would be required to reach the critical crack size subsequent to leakage.

The probabilistic model in MRP-375 was developed to predict PWSCC degradation and its associated risks in RVCHs. The model utilized in this probabilistic evaluation is modified from the model presented in Appendix B of MRP-335 (Reference 6) that evaluated surface stress improvement of RVCHs with Alloy 600 nozzles. The integrated probabilistic model in MRP-375 includes submodels for simulating component and crack stress conditions, PWSCC initiation, PWSCC growth, and flaw examination. The submodels for crack initiation and growth prediction for

Alloy 600 reactor pressure vessel head penetration nozzles in MRP-335, Revision 1 were adapted for RVCHs with Alloy 690 nozzles by applying FOIs to account for the superior PWSCC resistance of Alloys 690/52/152. The average leakage frequency and average ejection frequency were determined using the Monte Carlo simulation model with conservative FOI assumptions. The results show that, using only modest FOIs for Alloys 690/52/152, the potential for developing a safety significant flaw (risk of nozzle ejection) is acceptably small for a volumetric/surface examination period up to 40 years.

The evaluations performed in MRP-375 were prepared to bound all PWR replacement RVCH designs that are manufactured using Alloy 690 base material and Alloy 52/152 weld materials. The evaluations assume a bounding continuously operating RVCH temperature of 613°F and a relatively large number of RVCH penetrations (89). This number of penetrations bounds the number of penetrations found in the STP replacement heads.

Additional Evaluations Performed under MRP-386

MRP-386 summarizes years of laboratory testing by an international group of experts to quantify the PWSCC growth rates of Alloy 690 and its weld metals, Alloy 52/152, in simulated PWR primary water. Fracture mechanics-based tests were conducted under testing conditions designed to promote PWSCC in several product forms of wrought Alloy 690 and in several alloy variants of weld metal Alloy 52/152. For some Alloy 690 tests, laboratory-added plastic strain (i.e., "cold work") of up to 30% reduction in thickness was used to accelerate PWSCC growth rates. Variables known to affect PWSCC were assessed and included in the Crack Growth Rate (CGR) model and/or disposition equations, including: the mode I stress intensity factor, the test temperature, the yield strength of the material, the electrochemical potential in the test environment, and the orientation of the crack relative to the direction of added cold work. The data were vetted by an international expert panel and were then used to develop predictive models of the PWSCC growth rate in thick walled Alloy 690 (including the heat-affected zone) and its weld metals, Alloys 52 and 152, and variants of these alloys. The lower bound FOI for Alloy 690 compared to Alloy 600 is 25, while the more realistic and recommended FOI is 38. For Alloy 52/152 compared to Alloy 82/182, the lower bound FOI is 253, while the recommended FOI is 324.

RVCH Design and Operation

The analysis presented in MRP-375 was intended to cover all replacement heads in U.S. PWRs, including the STP RVCHs. The MRP-375 analyses assume a reactor vessel head operating temperature of 613°F to bound the known reactor vessel head temperatures of all U.S. PWRs currently operating. The average RVCH operating temperature for STP Unit 1 and Unit 2 over the operating period from installation of the replacement head until the end of the requested volumetric/surface inspection period is conservatively no more than 585°F based on several cycles of actual temperature data from the Reactor Vessel Water Level Instrument System upper head unheated junction thermocouples. Based on this the STP Unit 1 and Unit 2 RVCHs average operating temperature, which is the measure of the temperature relevant to potential PWSCC degradation, is bounded by the MRP-375 evaluation which assumes 613°F for its main deterministic and probabilistic calculations.

As stated in MRP-375 "...to further allow consistent interpretation, all results are adjusted to an operating temperature of 613°F (323°C) using the Arrhenius relationship with an activation energy of 130 kJ/mol. This operating temperature is believed to be an upper bound for operating Alloy 690 top heads in service today." Reduced operating temperature results in a significant improvement in both crack initiation and crack propagation. As stated in MRP-375 Case M2 – Reduced Operating Temperature:

Reducing the head temperature from 613°F to 600°F (323°C to 316°C) reflects that most Alloy 690 hot heads operate below 613°F (323°C), with a majority operating between 590°F and 600°F (310°C to 316°C). The reduced temperature decreases the thermally activated PWSCC flaw initiation and growth processes (i.e., through the Arrhenius relation in the model).

Reducing the head temperature leads to a more than tenfold reduction in AEF (Average Ejection Frequency). Similarly, the frequency of leakage is decreased to less than half its base case value.

The STP Unit 1 and Unit 2 RVCHs were designed and fabricated using materials and techniques to reduce susceptibility to PWSCC with enhanced access for visual inspection of RPV head penetrations.

Previous Examinations of the STP Units 1 and 2 Replacement RVCHs

Preservice volumetric examinations of the replacement RVCHs partial-penetration welded nozzles were performed prior to head installation at STP Units 1 and 2. There were no recordable indications identified during the preservice volumetric examinations of the nozzle tube in the area of the J-groove welds.

Bare metal visual examinations (VE) have been performed on the STP replacement RVCHs as follows:

STP Unit 1
Preservice VE November 18, 2009
Inservice VE June 1, 2014 and November 11, 2018

STP Unit 2
Preservice VE May 2, 2010
Inservice VE May 9, 2015

All VEs were performed in accordance with ASME Code Case N-729-1, Table 1, Item B4.30 except the VE performed in the Fall of 2018 on STP Unit 1 as a result of the 2017 NRC Rulemaking that required ASME Code Case N-729-4. These visual examinations were performed by VT-2 qualified examiners and the additional requirements of Note 2, Table 1 of both revisions of the ASME Code Case N-729. VE was performed on the outer surface of the RVCH including the annulus area of the penetration nozzles. These examinations did not reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage.

FOI Approach and Minimum FOI Implied by the Requested Inspection Period

ASME Code Case N-729-4 is based upon conclusions reached in MRP-117 (Reference 7) that a reexamination interval between volumetric/surface examinations of one 2 year operating cycle is acceptable for a head with Alloy 600 nozzles and operating at a temperature of 605°F. The inspection period for heads with Alloy 690 nozzles in ASME Code Case N-729-4 is a nominal 10 years, which represents a minimum implied FOI of five over Alloy 600.

Per the technical basis documents (MRP 117, MRP-110NP, and MRP-105NP) for ASME Code Case N-729-4 for heads with Alloy 600 nozzles [References 7, 8, and 9], the effect of differences in operating temperature on the required volumetric/surface reexamination interval for heads with Alloy

600 nozzles can be addressed on the basis of the Re-Inspection Years (RIY) parameter. The RIY parameter adjusts the EFPY of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth. For heads with Alloy 600 nozzles, ASME Code Case N-729-4 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) limits the interval between subsequent volumetric/surface inspections to RIY = 2.25. The RIY parameter, which is referenced to a head temperature of 600°F, limits the time available for potential crack growth between inspections.

The RIY parameter for heads with Alloy 600 nozzles is adjusted to the reference head temperature using activation energy of 31 kcal/mol (130 kJ/mol). Based on the available laboratory data, the same activation energy is applicable to model the temperature sensitivity of growth of a hypothetical PWSCC flaw in the Alloys 690/52/152 material of the replacement RVCHs. Key laboratory crack growth rate testing data for Alloy 690 wrought material investigating the effect of temperature are as follows:

1. Results from Argonne National Laboratory indicate that Alloy 690 with 0-26% cold work has an activation energy between 100 and 165 kJ/mol (24-39 kcal/mol) NUREG/CR-7137 (Reference 10) concludes that the activation energy for Alloy 690 is comparable to the standard value for Alloy 600 (130 kJ/mol).

2. Testing at Pacific Northwest National Laboratory found activation energy of approximately 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work (Reference 11).

These data show that it is reasonable to assume the same crack growth thermal activation energy as was determined for Alloys 600/82/182 (namely 130 kJ/mol (31 kcal/mol)) for modeling growth of hypothetical PWSCC flaws in Alloys 690/52/152 PWR plant components.

As discussed in the MRP-117 technical basis document for heads with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate reexamination interval to detect any PWSCC in a timely fashion. U.S. PWR inspection experience for heads with Alloy 600 nozzles has confirmed that the RIY = 2.25 interval results in a suitably conservative inspection program.

Minimum FOI Implied by Requested Inspection Period

STPNOC has assessed the minimum Alloys 690/52/152 FOI that supports the requested STP Unit 1 and Unit 2 extension period for comparison with the laboratory crack growth rate data presented in MRP-375 and MRP-386. To calculate the minimum implied FOI for the STP Unit 1 and Unit 2 RVCHs, an operating temperature of 585°F was conservatively assumed and the RIY parameter for the requested examination interval is compared with the ASME Code Case N-729-4 interval for Alloy 600 nozzles of RIY600 = 2.25.

The STP Unit 1 and Unit 2 RVCHs operating temperature of 585°F corresponds to an RIY temperature adjustment factor of 0.683 (versus the reference temperature of 600°F) using the activation energy of 31 kcal/mol (130 kJ/mol) for crack growth of ASME Code Case N-729-4. As discussed previously, it is appropriate to apply this standard activation energy for modeling crack growth of Alloys 690/52/152 plant components. Conservatively assuming that the EFPYs of operation accumulated for the STP Unit 1 and Unit 2 since the RVCHs replacement is equal to the calendar years since replacement, the RIY for the requested 17-year extended inspection period is:

$$RIY_{690} = (0.683 \text{ growth rate factor}) \times (17 \text{ calendar years extended interval}) = 11.61$$

The FOI implied by this RIY value for STP Unit 1 and Unit 2 is:

$$FOI = (11.61\ RIY_{690}) / (2.25\ RIY_{600}) = 5.2$$

Considering the statistical compilation of data provided in Figures 3-2, 3-4, and 3-6 of MRP-375, this factor of improvement is conservatively less than the FOI of 10 that bounds essentially all of the crack growth rate data presented in MRP-375 and less than one third the minimum FOI of 25 presented in Table 5-1 of MRP-386. Furthermore, as discussed in Sections 2 and 3 of MRP-375, PWR plant experience and laboratory testing have demonstrated a large improvement in resistance to PWSCC initiation of Alloys 690/52/152 in comparison to that for Alloys 600/82/182. Hence, the demonstrated improvements in PWSCC initiation and growth confirm on a conservative basis the acceptability of the requested period of extension.

Conclusion

In summary, the basis for extending the intervals from once each interval (nominally 10 calendar years) to nominally once every 17 calendar years is based on plant service experience, factor of improvement studies using laboratory initiation and growth data, deterministic modeling, and probabilistic study results. The results of the analysis show that the alternative proposed frequency results in a substantially reduced effect on nuclear safety when compared to a head with Alloy 600 nozzles and examined per the current requirements.

F. Duration of Proposed Alternative

This request is applicable to the STP Units 1 and 2 inservice inspection programs for the third and fourth 10-year inspection intervals because using the proposed examination frequency will require the examination to be performed in the fourth interval.

G. Precedents

There have been submittals from multiple plants to request an alternative from the frequency of ASME Code Case N-729-4 for volumetric or surface examinations of heads with Alloy 690 nozzles. Furthermore, the NRC has developed Safety Evaluations to permit the extension of the interval for volumetric or surface inspections per N-729-4. The first of these was Arkansas Nuclear One, Unit 1, and some subsequent requests including the associated Requests for Additional Information and Safety Evaluations are shown in the table below:

Plant	NRC ADAMS Accession No.				Status
	Relief Request	Request for Additional Information (RAI)	RAI Response	NRC Safety Evaluation	
Arkansas Nuclear One, Unit 1 (5.5-year Extension)	ML16173A297	None	None	ML17018A283	Accepted
Beaver Valley, Unit 1 (5-year Extension)	ML17044A440	None	None	ML17222A162	Accepted
Calvert Cliffs, Units 1 & 2 (6-year Extension)	ML15201A067	None	None	ML15327A367	Accepted
Comanche Peak, Unit 1 (5-year Extension)	ML15120A038	None	None	ML15259A004	Accepted
J.M. Farley, Unit 2 (5-year Extension)	ML15111A387	None	None	ML15104A192	Accepted
North Anna, Unit 2 (5-year Extension)	ML14283A044	None	None	ML15091A687	Accepted
Prairie Island, Units 1 and 2 (5-year Extension)	ML14258A124	ML15030A008	ML15036A252	ML15125A361	Accepted
Salem, Unit 1 (5-year Extension)	ML15098A426	None	None	ML15349A956	Accepted
St. Lucie, Unit 1 (5.5-year Extension)	ML17045A357	None	None	ML17219A174	Accepted
St. Lucie, Unit 2 (5.5-year Extension)	ML16076A431	None	None	ML16292A761	Accepted

H. Referencesk;

1. 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, Approved June 22, 2012.
2. Materials Reliability Program (MRP) Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375), EPRI, Palo Alto, CA: 2014, 3002002441.
3. Materials Reliability Program (MRP) Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386), EPRI, Palo Alto, CA: 2017, 3002010756.
4. Materials Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1, EPRI, Palo Alto, CA: 2002, 1006695.
5. Materials Reliability Program Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115), EPRI, Palo Alto, CA: 2004, 1006696.
6. Materials Reliability Program Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement (MRP-335) Revision 1, EPRI, Palo Alto, CA: 2013, 3002000073.
7. Materials Reliability Program Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants (MRP-117), EPRI, Palo Alto, CA: 2004, 1007830. [ML043570129]
8. Materials Reliability Program Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants (MRP-110NP), EPRI, Palo Alto, CA: 2004, 1009807-NP. [ML041680506]
9. Materials Reliability Program Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking (MRP-105NP), EPRI, Palo Alto, CA: 2004, 1007834-NP. [ML041680489]
10. U.S. NRC, Stress Corrosion Cracking in Nickel-Base Alloys 690 and 152 Weld in Simulated PWR Environment-2009, NUREG/CR-7137, ANL-10/36, June 2012, [ML12199A415]
11. Materials Reliability Program: Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking (MRP-237) Revision 2, EPRI, Palo Alto, CA: 2013, 3002000190.