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10 CFR 50.55a

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

Salem Generating Station, Unit 1
Renewed Facility Operating License No. DPR-70
NRC Docket No. 50-272

Subject: Request for Relief from Alloy 690 PWR Reactor Vessel Head Inspection Interval, Fourth (4^{th}) 10-Year Interval

In accordance with 10 CFR 50.55a(z)(1), "Codes and standards," PSEG Nuclear LLC (PSEG), hereby requests NRC approval of proposed Relief Request S1-I4R-191 for Salem Unit 1. PSEG is requesting relief from the reactor vessel closure head requirements of ASME Code Case N-729-4 and approval of an alternative to allow the use of the Code Case N-729-6, specifically the 20 year examination frequency for the Salem Unit 1 Alloys 690, 52, and 152 reactor vessel closure head penetrations.

PSEG requests approval of the proposed request by September 25, 2020 to support use of the proposed alternative during the Unit 1 refueling outage S1R27.

The proposed relief request is provided in Attachment 1.

There are no regulatory commitments contained in this letter.

If you have any questions or require additional information, please contact Lee Marabella at 856-339-1208.

Respectfully,

Paul R. Duke, Jr.
Manager – Licensing

Attachments:
1. 10 CFR 50.55a Relief Request S1-I4R-191

cc: Administrator, Region I, NRC
NRC Senior Resident Inspector, Salem
Project Manager, Salem, USNRC
Chief, NJBNE
Corporate Commitment Tracking Coordinator
Site Compliance Commitment Tracking Coordinator

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10 CFR 50.55a Request Number S1-I4R-191
Proposed Alternative In Accordance with 10 CFR 50.55a(z)(1)
—Alternative Provides Acceptable Level of Quality and Safety—

1. ASME Code Component(s) Affected

Code Class:	1
Examination Category:	Code Case N-729-4
Item Number:	B4.40
Description:	ASME Class 1 Pressurized Water Reactor (PWR) Reactor Vessel Upper Head (Closure Head) (RVCH) nozzles and partial-penetration welds fabricated with primary water stress corrosion cracking (PWSCC)-resistant materials.

2. Applicable Code Edition and Addenda

American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection and Testing of Components of Light-Water Cooled Plants," 2004 Edition with no Addenda. Examinations of the reactor vessel closure head (RVCH) penetrations are performed in accordance with 10CFR50.55a(g)(6)(ii)(D), which specifies the use of code case N-729-4, with conditions. Salem unit 1 Fourth (4^{th}) ISI interval began on May 20, 2011 and is scheduled to end on December 31, 2020.

3. Applicable Code Requirement

Code of Federal Regulations (CFR) 10CFR50.55a(g)(6)(ii)(D) delineates the Augmented ISI requirements for Reactor Vessel Head Inspections. 10CFR50.55a(g)(6)(ii)(D)(1), requires:

> "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 BPV Code Case N-729-4 paragraph 2410 specifies that the reactor vessel upper head penetrations shall be examined on a frequency in accordance with Table 1 of the code case. The basic inspection requirements of ASME Code Case N-729-4 for partial-penetration welded Alloy 690 head penetration nozzles are as follows:

> Item B4.30 – Direct visual examination (VE) of the entire outer bare metal surface of head for evidence of leakage every third refueling outage or five (5) calendar years, whichever is less.

> Item B4.40 – Volumetric and/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 have not previously been identified).

In Reference 3, the NRC authorized PSEG Nuclear (PSEG) a one-time extension of the Item B4.40 inspection interval to the refueling outage scheduled to begin in fall 2020.

4. Reason for Request

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) document Materials Reliability Program (MRP)-375 (Reference 2) was developed to support a technically based volumetric or surface reexamination interval using appropriate analytical tools. This technical basis demonstrates that the reexamination interval can be extended to at least a 20 year interval while maintaining an acceptable level of quality and safety. Additionally, the NRC proposed in Reference 4 to amend its regulations to incorporate by reference ASME BPV Code Case N–729–6 (Reference 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,’’ with conditions on its use. The examination frequency specified in Code Case N-729-6 for Item Number B4.40 is not to exceed two inspection intervals (nominally 20 calendar years).

Therefore, PSEG is requesting approval of this alternative to allow the use of the ISI examination interval specified in Code Case N-729-6 of 20 years for the Salem Unit 1 Alloys 690/52/152 reactor vessel closure head penetrations.

5. Proposed Alternative and Basis for Use

PSEG proposes to extend the inspection interval for Code Case N-729-4 item B4.40 from once each interval (nominally 10 calendar years) by 10 additional years for a total of 20 calendar years based on plant service experience and factor of improvement (FOI) studies using laboratory data (Reference 2).

In addition to maintaining an acceptable level of quality and safety with this proposed use of code case N-729-6 that specifies a 20 year volumetric examination frequency, Salem Unit 1 will continue to schedule one RVCH visual examination on an interval of every third refueling outage or 5 calendar years, whichever is less, in accordance with Item B4.30 of ASME Code Case N-729-4.

The original Salem Unit 1 RVCH was replaced with a new RVCH, using Alloy 690/52/152 materials, during the refueling outage that returned the unit to operation in November 2005. In accordance with Table 1 of ASME Code Case N-729-4 Item B4.40, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D)(*3*) and Reference 3, PSEG is required to perform a volumetric and/or surface examination of essentially 100% of the required volume or equivalent surfaces of the nozzle tube on a one-time 15 year frequency. Currently Salem has Item B4.40 examinations scheduled for fall 2020.

Materials Reliability Program MRP-375 (Reference 2) provides a technical justification to extend the volumetric/surface examination interval of the RVCH nozzle penetrations from 10 years to 20 years. The ASME Code Case Committee adopted the revised volumetric/surface examination of two inspection intervals (20 years) in ASME Code Cases N-729-5 and

N-729-6.

This request to utilize 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.

Basis for Use

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) was 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.

Evaluation of Existing Alloys 690/52/152 Data and Experience by MRP-375 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 6) for Alloy 600 material and in MRP-115 (Reference 7) 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 RPVH [Reactor Pressure Vessel Head], 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 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.
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>> 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.

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, Revision 1 (Reference 8) 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 Salem Unit 1 replacement head.

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 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 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 Salem Unit 1 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 RVCH operating temperature for Salem Unit 1 over the operating period from installation of the replacement head until the end of the requested volumetric or surface inspection period is 597.2°F (Reference 5). Thus, the Salem Unit 1 RVCH operating temperature 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 Salem Unit 1 RVCH was designed and fabricated using materials and techniques to reduce susceptibility to PWSCC and facilitate prompt detection of potential leakage by visual examination. The RVCH contains fifty-five (55) nozzle penetrations of which fifty-three (53) are used for control rod drive mechanisms (CRDMs), one (1) is used for reactor vessel level instrumentation (RVLIS), and one (1) is a small-diameter vent line penetration near the center of the RVCH. The replacement RVCH was manufactured by Framatome (AREVA) and placed in service in November 2005. The replacement RVCH was manufactured as a single forging, which eliminated several welds including the flange to head weld. The replacement RVCH is fabricated from SA-508, Grade 3, Class 1 low alloy steel and clad with an initial layer of 309 L stainless steel followed by subsequent layers of 308 L stainless steel. The nozzle housing penetrations on the replacement RVCH are fabricated from Inconel SB-167 (Alloy 690) UNS N06690 supplied by Valinox. The nozzle J-groove welds utilized ERNiCrFe-7 (UNS N06052) and/or ENiCrFe-7 (UNS W86152) weld materials.

A pre-service volumetric examination of the Salem Unit 1 replacement RVCH partial-penetration welded nozzles was performed prior to installation. The volumetric examinations included scanning the nozzles to the fullest extent possible, from the end of the nozzle to a minimum of 2 inches above the root of the J-groove weld on the uphill side. There were no recordable indications identified during the pre-service volumetric examinations of the nozzle tube in the area of the J-groove welds.

Bare metal visual examinations (VE) were performed on the Salem Unit 1 replacement RVCH in 2010 and 2013 in accordance with ASME Code Case N-729-1 and in 2017 in accordance with ASME Code Case N-729-4. The visual examinations were performed by VT-2 qualified examiners 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. During every refueling outage, a separate plant walkdown is performed to visually detect evidence of leakage of plant components, including leakage from the region of the RVCH. For surfaces not obscured by insulation, leakage is visually apparent due to streaking and the precipitation of white crystals of boric acid on the dark surface of reactor components.

Salem Unit 1 has an Alloy 600 management plan for managing PWSCC of reactor components and a boric acid corrosion control (BACC) program for minimizing the potential for consequential corrosion of reactor components. Under NEI 03-08 Materials Initiative of the Nuclear Energy Institute (NEI), the industry document EPRI MRP-126, *Generic Guidance for Alloy 600 Management*, requires every U.S. PWR to have an Alloy 600 management program. NRC Generic Letter 88-05 requires that utilities develop and implement programs to identify leaks and take corrective action to prevent recurrence. In addition, under NEI 03-08, the industry document WCAP-15988-NP Rev. 2, *Generic Guidance for an Effective Boric Acid Inspection Program for Pressurized Water Reactors*, requires that every U.S. PWR have a BACC program that addresses boric acid corrosion due to borated water from any plant system, including those outside of containment. Evidence of leakage during a plant walkdown could indicate the occurrence of PWSCC and is tracked by plant personnel in light of the industry guidance.

Minimum FOI Implied by Requested Inspection Period

PSEG has assessed the minimum Alloy 690/52/152 FOI that supports the requested Salem Unit 1 extension period for comparison with the laboratory crack growth rate data presented in MRP-375. An extension of the examination interval to 20 years would imply a factor of 10 for Alloys 690/52/152 relative to Alloys 600 and 182 for the proposed period between

volumetric or surface examinations for a head operated at a temperature of 600°F. To calculate the minimum implied FOI for the RVCH operating temperature of 597°F, the RIY parameter for the requested examination interval is compared with the ASME Code Case N-729-4 interval for Alloy 600 nozzles of RIY = 2.25.

The representative Salem Unit 1 RVCH operating temperature of 597°F corresponds to an RIY temperature adjustment factor of 0.932 (versus the reference temperature of 600°F) using the activation energy of 31 kcal/mol (130 kJ/mol) for crack growth from ASME Code Case N-729-4. As discussed previously, it is appropriate to apply this standard activation energy for modeling crack growth of Alloy 690/52/152 plant components. Conservatively assuming that the EFPYs of operation accumulated at Salem Unit 1 since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended interval of (0.932)(20) = 18.64. The FOI implied by this RIY value for Salem Unit 1 is (18.64)/(2.25) = 8.28.

Considering the statistical compilation of data provided in Figures 3-2, 3-4, and 3-6 of EPRI MRP-375, this factor of improvement is conservatively less than the FOI of 10 that bounds the crack growth rate data presented. Furthermore, as discussed in Section 2 and Section 3 of EPRI 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 the acceptability of the limited requested period of extension on a conservative basis.

<u>Duration of Proposed Alternative</u>

The proposed Alternative is requested for a 20-year period from the time the Salem Unit 1 replacement RVCH was placed in service.

	Placed In Service Date	Nominal End Of Alternative Date
Salem Unit 1 RVCH Head	November 6, 2005	November 6, 2025

7. <u>Precedents</u>

The NRC has authorized the 20-year examination frequency for volumetric examinations of heads with Alloy 690 nozzles at Prairie Island Units 1 & 2 as well as multiple other plants alternatives approved that extended the inspection interval from the currently required 10-year frequency specified in ASME Code Case N-729-1 and N-729-4 to greater than 15 year frequency.

Plant	Relief Request	NRC Safety Evaluation
Prairie Island Units 1 & 2	ML18065A583	ML19046A166
Diablo Canyon Unit 2	ML18213A375	ML19023A026
St. Lucie Unit 1	ML17045A357	ML17219A174
Arkansas Nuclear One Unit 1	ML16173A297	ML17018A283
Calvert Cliffs, Units 1 & 2	ML15201A067	ML15327A367

8. References

1. ASME Code Case N-729-6, “Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1,” Approved March 03, 2016.

2. Materials Reliability Program: MRP-375, Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (EPRI 3002002441), Attachment 1. [NRC ADAMS Accession No. ML14283A046]

3. NRC letter to PSEG, "Salem Nuclear Generating Station, Unit No. 1 - Relief from the Requirements of the ASME Code (CAC NO. MF6089)," December 24, 2015. [NRC ADAMS Accession No. ML15349A956]

4. Federal Register / Vol. 83, No. 218 / Friday, November 9, 2018 / Proposed Rules, "American Society of Mechanical Engineers 2015–2017 Code Editions Incorporation by Reference".

5. Salem Design Calculation: S-C-RC-MDC-1928 Revision 3 “Determination of Effective Degradation Years (EDY) for Salem Units 1 and 2”

6. EPRI Materials Reliability Program: “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, Final Report – November 2002 (Report No. 1006695)

7. EPRI 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, Final Report – November 2004 (Report No. 1006696)

8. EPRI Materials Reliability Program: “Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement (MRP-335, Revision 1),” EPRI, Palo Alto, CA, Final Report – January 2013 (Report No. 3002000073)