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March 6, 2018

L-PI-18-007
10 CFR 50.55a(z)(1)

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

Prairie Island Nuclear Generating Plant, Units 1 and 2
Docket Nos. 50-282 and 50-306
Renewed Facility Operating License Nos. DPR-42 and DPR-60

10 CFR 50.55a Requests Nos. 1-RR-5-9 and 2-RR-5-9 Associated with the Fifth Ten-Year Interval for the Inservice Inspection Program

References: 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”, dated June 22, 2012.

Pursuant to 10 CFR 50.55a(z)(1), Northern States Power Company, a Minnesota corporation, doing business as Xcel Energy (hereafter "NSPM"), requests U.S. Nuclear Regulatory Commission (NRC) approval of an alternative to the frequency specified for performance of volumetric/surface examinations of Reactor Vessel Closure Head components in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Code Case N-729-4 (Reference 1). A 10 CFR 50.55a request to apply an alternative for the Fifth Ten-Year Interval of the Prairie Island Nuclear Generating Plant (PINGP) Inservice Inspection (ISI) Program applying to both Units 1 and 2 (1-RR-5-9 and 2-RR-5-9, respectively) is provided within the enclosure to this letter.

NSPM requests approval of these 10 CFR 50.55a requests by March 30, 2019.

Summary of Commitments

This letter makes no new commitments and no revisions to existing commitments.

If there are any questions or if additional information is needed, please contact Mr. Richard Loeffler at 612-342-8981.

Scott Sharp
Site Vice President, Prairie Island Nuclear Generating Plant
Northern States Power Company – Minnesota

Enclosure

cc: Administrator, Region III, USNRC
Project Manager, Prairie Island, USNRC
Resident Inspector, Prairie Island, USNRC

10 CFR 50.55a Request 1-RR-5-9, Revision 0 (PINGP Unit 1)
10 CFR 50.55a Request 2-RR-5-9, Revision 0 (PINGP Unit 2)
Proposed Alternative for Examination of Reactor Vessel Upper Head Closure Head

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)

Provides an Acceptable Level of Quality and Safety

1. American Society of Mechanical Engineers (ASME) Code Component(s) Affected

Component:	Reactor Vessel Closure Head (RVCH) Nozzles
Code Class:	1
Reference:	Code Case N-729-4 (Reference 1)
Examination Category:	Class 1 Pressurized Water Reactor (PWR) Vessel Upper Head
Item Number:	B4.40
Description:	Nozzles and partial-penetration welds of PWSCC[1]–resistant materials in head
Component Number:	157-051 and 257-051

2. Applicable Code Edition and Addenda

The applicable ASME Boiler and Pressure Vessel (BPV) Code edition and addenda for the Fifth Ten-Year Interval beginning December 21, 2014, is Section XI, “Rules for Inservice Inspection of Nuclear Power Plant components,” 2007 Edition through the 2008 Addenda. Note, the upper head for the Prairie Island Nuclear Generating Plant (PINGP) Unit 2 reactor pressure vessel (RPV) was replaced in May 2005 and the upper head for the PINGP Unit 1 RPV was replaced in May 2006.

3. Applicable Code Requirement

The Code of Federal Regulations 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.

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1. Primary water stress corrosion cracking

10 CFR 50.55a(g)(6)(ii)(D)(*2*) conditions the ASME Code Case N-729-4 by stating:

> Appendix I of ASME BPV Code Case N–729–4 shall not be implemented without prior NRC approval.

10 CFR 50.55a(g)(6)(ii)(D)(*3*) conditions ASME Code Case N-729-4 bare metal visual examination frequency and is not relevant to this request.

10 CFR 50.55a(g)(6)(ii)(D)(*4*) conditions ASME Code Case N-729-4 by stating:

> In addition to the requirements of Paragraph–3132.1(b) of ASME BPV Code Case N-729-4, a component whose surface examination detects rounded indications greater than allowed in Paragraph NB–5352 in size on the partial-penetration or associated fillet weld shall be classified as having an unacceptable indication and corrected in accordance with the provisions of [P]aragraph–3132.2 of ASME BPV Code Case N-729–4.

ASME Code Case N-729-4 specifies that the RPV upper head components shall be examined on a frequency in accordance with Table 1 of this 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:

- Volumetric or surface examination of all nozzles every inspection interval (nominally 10 calendar years) provided that flaws attributed to PWSCC have not previously been identified in the head; and

- Direct visual examination of the outer surface of the head for evidence of leakage every third refueling outage or 5 calendar years, whichever is less.

4. <u>Reason for Request</u>

ASME Code Case N-729-4 with the conditions of 10 CFR 50.55a(g)(6)(ii)(D) requires volumetric/surface examination of the RPV upper head nozzles and welds at a once per nominal 10 calendar year interval after the head was placed into service.

The PINGP RVCH penetration nozzles and associated welds are made from Alloys 690/52/152. As discussed in Electric Power Research Institute (EPRI) Materials Reliability Program (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 2), 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.

As stated in EPRI MRP report: “Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)” (Reference 3):

> … current inspection regime was established in 2004 as a conservative approach and was intended to be subject to reassessment upon the availability of additional laboratory data and plant experience on the performance of Alloy 690 and Alloy 52/152. Since that time, plant experience and laboratory testing have continued to demonstrate the much greater resistance of these replacement alloys to PWSCC compared to that for Alloys 600/82/182 for the material conditions relevant to partial-penetration welded nozzles. Although laboratory research is ongoing to investigate and understand the times to crack initiation and the crack growth rates for these materials under various conditions, there are now sufficient data available to develop an improved technical basis for inspection of these components.

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, NSPM is requesting approval of this alternative to allow the use of a 20-year interval for the affected PINGP Unit 1 and Unit 2 components.

5. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), Northern States Power Company – Minnesota (NSPM) requests an alternative to performing the required volumetric/surface examinations for the RVCH components identified previously at the frequency prescribed in ASME Code Case N-729-4. Specifically, NSPM requests to extend the frequency of the volumetric/surface examination of the RVCH specified in Table 1, Item B4.40 of ASME Code Case N-729-4 for a nominal 10 year period beyond the one inspection interval (nominally 10 calendar years) from installation of the PINGP Unit 1 and Unit 2 replacement RVCHs. 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 20-year period from the time the replacement PINGP Unit 1 and Unit 2 RPV heads[2] were placed in service.

	Placed in Service Date	Nominal End of the Alternative Date
PINGP Unit 1 Replacement RPV Head	June 6, 2006	June 6, 2026
PINGP Unit 2 Replacement RPV Head	June 10, 2005	June 10, 2025

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.

2. Pre-service inspection of the PINGP Unit 1 RVCH was performed in 2005 and the head was replaced in May 2006. Pre-service inspection of the PINGP Unit 2 RVCH was performed in 2004 and the head was replaced in May 2005.

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 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][3] = 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.

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 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 PINGP 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 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

3. 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.

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 PINGP 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 PINGP Unit 1 and Unit 2 over the operating period from installation of the replacement head to until the end of the requested volumetric/surface inspection period is conservatively no more than 595°F based on hot leg temperatures recorded over a year of operation. Based on this the PINGP 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 PINGP Unit 1 and Unit 2 RVCHs were designed and fabricated using materials and techniques to reduce susceptibility to PWSCC and enhanced access doors for inspection of RPV head penetrations.

The following table summarizes the design attributes of the PINGP Unit 1 and 2 RVCHs.

Feature	Description
Reactor Vessel (RV) Head Material	SA-508, Grade 3, Class 1 with a targeted chromium content of 0.15% or less, and sulfur content limited to 0.015% maximum. Cladding is generally 309L or 308L, 0.125 inch minimum and 0.2 inch nominal thickness.
RV Head Assembly Upgrade Package	Enhanced access doors for inspection of RV Head penetrations.
Coatings	The RVCH is not painted to ensure full compliance with the NRC bare metal inspection criteria for RV head inspections.
RV Head 4 Inch Outside Diameter Penetration Material	SB-167 UNS N06690 (Inconel Alloy 690)
Penetration to Head Weld Material, Type, Interference Fit	Alloy 52 ERNiCrFe-7 (UNS N06052) and Alloy 152 ERNiCrFe-7 (UNS W86152), Narrow J-Groove, 0.001-0.002 inch interference fit.
Full Length Control Rod Drive Mechanism Penetrations	29
Spare Penetrations	4
Thermocouple Penetrations	3 core exit thermocouple nozzle assemblies.
Reactor Coolant Gas Vent System Penetration Pipe Nozzle	1 inch Schedule 160 SB-167 UNS N06690 (Inconel Alloy 690) with Alloy 52/152 ERNiCrFe-7 Narrow J-Groove Weld.
Reactor Vessel Level Indication System Penetration Pipe Nozzle	1 inch Schedule 160 SB-167 UNS N06690 (Inconel Alloy 690) with Alloy 52/152 ERNiCrFe-7 Narrow J-Groove Weld.

Note that the probabilistic analysis in MRP-375 was performed assuming a head with 89 partial-penetration welded nozzles. This number bounds the number present in the PINGP replacement heads (38 nozzles). The number of penetrations included in the probabilistic model is not a key variable, and the assumed number of penetrations results in a small change in results relative to other sensitivity cases. Thus, the probabilistic calculations of MRP-375 cover all U.S. replacement RVCHs regardless of the precise number of penetrations.

The RVCHs were buttered with Alloy 690 in the area of the penetration. A partial penetration J-groove weld using Alloy 52/52M (ERNiCrFe-7 UNS N06052) filler metal was used between the Alloy 690 penetration and the head on the inside of the RVCH. Two modifications were introduced in the weld to reduce residual stress: A narrow gap J-groove weld edge preparation was used to reduce the volume of weld metal deposited and automatic welding processes were used. Manual welding of the first and reinforcement layer was permitted; all remaining layers were deposited using an automatic welding process.

Based on the above, the PINGP RVCHs are expected to have as good, if not better, resistance to PWSCC as compared to the MRP-375 base case.

A bare metal visual examination was performed of the PINGP Unit 1 replacement RVCH in 2014 and the PINGP Unit 2 RVCH in 2017, in accordance with ASME Code Case N-729-1 and Code Case N-729-4, respectively. These visual examinations were performed by VT-2 qualified examiners on the outer surface of the RVCHs 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. Continued visual inspection in accordance with ASME Code Case N-729-4 will provide prompt detection of any potential leakage.

As an alternative to the requirements of ASME Code Case N-729-4 with conditions of 10 CFR 50.55a for volumetric/surface examination of the reactor vessel closure heads on a nominal 10-year interval, NSPM requests approval to perform volumetric examination of the RVCHs to the requirements of ASME Code Case N-729-4 with conditions of 10 CFR 50.55a on a nominal 20-year interval.

FOI Approach and Minimum FOI Implied by the Requested Inspection Period

ASME Code Case N-729 is based upon conclusions reached in MRP-117 (Reference 7) that a reexamination interval between volumetric/surface examinations of one 24-month 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.

<u>FOI Approach</u>

Per the technical basis documents for ASME Code Case N-729 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 easily 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 130 kJ/mol (31 kcal/mol) (Reference **Error! Bookmark not defined.**). 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 (Reference 7) 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.

<u>Minimum FOI Implied by Requested Inspection Period</u>

NSPM has assessed the minimum Alloys 690/52/152 FOI that supports the requested PINGP Unit 1 and Unit 2 extension period for comparison with the laboratory crack growth rate data presented in MRP-375 and MRP-386. An extension of the examination interval to 20-years would imply a factor of 20/2 or 10 for Alloys 690/52/152 relative to Alloys 600 and 182 for the proposed period between volumetric/surface examinations for a head operated at a temperature of 600°F. To calculate the minimum implied FOI for the PINGP Unit 1 and Unit 2 RVCHs, an operating temperature of 595°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 RIY_{600} = 2.25.

The PINGP Unit 1 and Unit 2 RVCHs operating temperature of 595°F corresponds to an RIY temperature adjustment factor of 0.882 (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 PINGP Unit 1 and Unit 2 since the RVCHs replacement is equal to the calendar years since replacement the RIY for the requested 20-year extended inspection period is:

RIY_{690} = (0.882 growth rate factor) x (20 calendar years extended interval) = 17.64

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

FOI = (17.64 RIY_{690}) / (2.25 RIY_{600}) = 7.8

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.

Conclusions

The proposed alternative provides an acceptable level of quality and safety for structural integrity as the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the PINGP Unit 1 and Unit 2 replacement RVCHs provide a superior reactor coolant system pressure boundary where the potential for PWSCC has been shown by analysis and by years of positive industry experience to be minute. The minimum FOI implied by the requested extension period represents a level of reduction in PWSCC crack growth rate versus that for Alloys 600/82/182 that is completely bounded on a statistical basis by the laboratory data compiled in MRP-375. Given the lack of PWSCC detected to date in any PWR plant applications of Alloys 690/52/152, the simple FOI assessment clearly supports the requested period of extension.

NSPM proposes to perform volumetric/surface examinations of the PINGP Unit 1 and Unit 2 reactor pressure vessel heads in accordance with ASME Code Case N-729-4 with conditions of 10 CFR 50.55a on a nominal 20-year interval as an alternative to the nominal 10-year interval required by the ASME Code Case N-729-4.

6. Duration of Proposed Alternative

Application of this alternative is requested for a nominal 20-year period from the time the replacement PINGP Unit 1 and Unit 2 RPV heads were placed in service.

	Placed in Service Date	Nominal End of the Alternative Date
PINGP Unit 1 Replacement RPV Head	June 6, 2006	June 6, 2026
PINGP Unit 2 Replacement RPV Head	June 10, 2005	June 10, 2025

7. Precedents

There have been many submittals from multiple plants requesting an alternative from the nominal 10-year interval of ASME Code Case N-729-1 (and now ASME Code Case N-729-4) for volumetric/surface examinations of RVCHs with Alloy 690 nozzles. A selection of some of the plants is shown below. Alternative intervals greater than 15 years have previously been granted in order to align with scheduled refueling outages. The alternatives approved for three sites (Items 2, 3 and 4) extended the inspection interval from an initial approved alternative to a total interval of up to 15.5 years. The approved alternative for Calvert Cliff Units 1 and 2 (Item 1) permitted an inspection interval not to exceed 16 years in order to align with scheduled refueling outages.

1. Letter from NRC to Exelon Nuclear, “Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2, Relief Request for Extension of Volumetric Examination Interval for Reactor Vessel Heads with Alloy 690 Nozzles (CAC Nos. MF5829 and MF5830)”, dated December 7, 2015 (ADAMS Accession No. ML15327A367)

2. Letter from NRC to Entergy Operations, Inc., “Arkansas Nuclear One, Unit 1, Request for Alternative ANO1-ISI-026 from Volumetric/Surface Examination Frequency Requirements of American Society of Mechanical Engineers Code Case N-729-1 (CAC No. MF8007)”, dated February 13, 2017 (ADAMS Accession No. ML17018A283)

3. Letter from NRC to FirstEnergy Nuclear Operating Company, “Beaver Valley Power Station, Unit No. 1 – Issuance of Relief Request 1-TYP-4-RV-05, Revision 0, From Certain ASME Code Reactor Pressure Vessel Penetration Examination Frequency Requirements (CAC No. MF9283)”, dated September 8, 2017 (ADAMS Accession No. ML17222A162)

4. Letter from NRC to Florida Power & Light Company, "St. Lucie Plant, Unit No. 1 – Relief from the Requirements of the ASME Code Regarding Relief Request 12 for the Fourth 10-Year Inservice Inspection Interval (CAC No. MF9273)", dated August 31, 2017 (ADAMS Accession No. ML17219A174)

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8. <u>References</u>

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. Electric Power Research Institute (EPRI) Materials Reliability Program: "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, Final Report – December 2017 (Report No. 3002010756)

3. EPRI 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, Final Report – February 2014 (Report No. 3002002441)

4. 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)

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