VIRGINIA ELECTRIC AND POWER COMPANY
RICHMOND, VIRGINIA 23261

March 19, 2020

U.S. Nuclear Regulatory Commission		Serial No.:	20-029
Attention: Document Control Desk		NRA/ENC:	R0
Washington, DC 20555		Docket Nos.:	50-338
		License Nos.:	NPF-4

VIRGINIA ELECTRIC AND POWER COMPANY (DOMINION ENERGY VIRGINIA)
NORTH ANNA POWER STATION UNIT 1
ASME SECTION XI INSERVICE INSPECTION PROGRAM
PROPOSED INSERVICE INSPECTION ALTERNATIVE N1-I5-NDE-004

Pursuant to 10 CFR 50.55a(z)(1), Virginia Electric and Power Company (Dominion Energy Virginia) requests approval of a proposed alternative to the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (BPV) Code, Section XI, reactor vessel head examination frequency requirements for North Anna Power Station Unit 1 (NAPS1). Specifically, Dominion Energy Virginia requests the inspection interval defined in 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, be extended from 10 years to 20 years. The request applies to volumetric/surface examinations of Alloy 690/52/152 reactor vessel upper head penetrations at NAPS1.

The technical basis for the proposed alternative is provided in the enclosure. This alternative provides an acceptable level or quality and safety in accordance with 10 CFR 50.55a(z)(1).

Dominion Energy Virginia requests NRC approval of this proposed alternative by March 1, 2021 to support the next scheduled NAPS1 refueling outage.

If you have any questions or require additional information, please contact Erica Combs at (804)-273-3386.

Sincerely,

Mark D. Sartain
Vice President – Nuclear Engineering and Fleet Support

Enclosure:

Relief Request N1-I5-NDE-004, Request for Alternative from Reactor Pressure Vessel Closure Head Examination Frequency Requirements

Commitments made in this letter: None

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Enclosure

Relief Request N1-I5-NDE-004

Request for Alternative from Reactor Pressure Vessel Closure Head Examination Frequency Requirements

Virginia Electric and Power Company
(Dominion Energy Virginia)
North Anna Power Station Unit 1

North Anna Power Station Unit 1
10 CFR 50.55a Request
Relief Request N1-I5-NDE-004

Proposed Alternative in Accordance with 10 CFR 50.55a(z)(1)
--Alternative Provides Acceptable Level of Quality and Safety--

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

The affected components are ASME Class 1 Pressurized Water Reactor (PWR) Reactor Pressure Vessel (RPV) Upper Head (Closure Head) nozzles and partial-penetration welds fabricated with Primary Water Stress Corrosion Cracking (PWSCC)-resistant materials. North Anna Power Station Unit 1 (NAPS1) penetration tubes and vent pipe are fabricated from Alloy 690 with Alloy 52/152 attachment welds.

2. Applicable Code Edition and Addenda

The applicable Code for the NAPS1 fifth 10-year inservice inspection (ISI) interval and ISI Program is the ASME Boiler and Pressure Vessel (BPV) Code Section XI, 2013 Edition with no Addenda [Reference 1]. The NAPS1 fifth interval started May 1, 2019 and ends April 30, 2029.

Note that a later applicable code edition will be adopted for the sixth ISI interval, and that this alternative applies to the frequency of the next RPV Closure Head (RPVCH) Exam.

3. Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(D) requires licensees of existing, operating PWRs to implement the requirements of 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, by the first refueling outage starting after August 17, 2017. Code Case N-729-4, Inspection Item B4.40 for ASME Class 1 RPVCH nozzles and partial-penetration welds fabricated with PWSCC-resistant materials requires volumetric and/or surface examination of essentially 100% of the required volume or equivalent surface of the nozzle tube

each inspection interval (nominally 10 calendar years). A demonstrated volumetric or surface leak assessment through all J-groove welds is required.

4. Reason for Request

Treatment of Alloy 690 RPVCHs in Code Case N-729-4 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 become available. Using plant and laboratory data, Electric Power Research Institute (EPRI) document *Materials Reliability Program: Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP–375)* [Reference 1] was developed to support a technically based volumetric / surface reexamination interval using appropriate analytical tools. MRP-375 demonstrates that the re-examination interval can be extended to the requested interval length while maintaining an acceptable level of quality and safety. Therefore, Dominion is requesting approval of this alternative to allow the use of the ISI interval extension for the affected NAPS1 components. ASME Code Case N-729-6, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds," approved March 3, 2016, requires Item No. B4.40 to be performed at a frequency not to exceed two inspection intervals (nominally 20 years) for all nozzles in the RPVCH having PWSCC-resistant materials.

The NAPS1 RPVCH penetration nozzles and associated welds are made from Alloys 690/52/152. As discussed in EPRI document *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)* [Reference 14], these materials have much greater PWSCC resistance compared to Alloys 600/82/182. Nearly 30 years of operating experience with no observations of PWSCC 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 re-examination interval for Alloy 690/52/152 components can be extended to a 20-year interval while maintaining an acceptable level of quality and safety. Therefore,

Dominion is requesting approval of this alternative to allow extension of the ISI interval from 10 years to 20 years for NAPS1 Alloy 690/52/152 RPVCH penetrations.

5. Proposed Alternative and Basis for Use

Dominion is requesting extension of the requirements of Code Case N-729-4, Inspection Item B4.40 for performing volumetric/surface exams of the NAPS1 RPVCH. Specifically, this would allow volumetric/surface examinations currently scheduled for the spring of 2021 (baseline exams performed in the spring of 2012) to be moved to the fall of 2031. This request applies to the inspection frequencies and not the inspection techniques, as the inspection techniques or other requirements may change with later editions of ASME Section XI and 10 CFR 50.55a.

Basis for Use

The basis for the inspection frequency for ASME Code Case N-729-4 comes, in part, from the analysis of laboratory and plant data presented in report MRP-111 [Reference 2], which was summarized in the safety assessment for RPVCHs in MRP-110 [Reference 3]. The material improvement factor for PWSCC of Alloy 690/52/152 materials over that of mill-annealed Alloys 600 and 182 was shown by this report to be on the order of 26 or greater.

Further evaluations were performed to demonstrate the acceptability of extending the inspection intervals for Code Case N-729-4, Inspection Item B4.40 components and documented in MRP-375. In summary, the basis for extending the intervals to once each interval (nominally 10 calendar years) to once every second interval (nominally 20 calendar years) is based on plant service experience, factor of improvement studies using laboratory data, deterministic study results, and probabilistic study results.

Per MRP-375, much of the laboratory data indicated a factor of improvement of 100 in terms of crack growth rates (CGRs) for Alloys 690/52/152 versus Alloys 600/182/82 (for equivalent temperature and stress conditions). In addition, laboratory and plant data demonstrate a factor of improvement in excess of 20 in terms of the time to PWSCC initiation. This reduced susceptibility to PWSCC initiation and growth supports elimination of all volumetric exams throughout the plant service period based on conservatively smaller factors of improvement confirmed in the research. Deterministic calculations demonstrate that the proposed alternative volumetric re-examination schedule is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size necessary to

produce a nozzle ejection. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring. Probabilistic calculations based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing show a substantially reduced effect on nuclear safety compared to a head with Alloy 600 nozzles examined per current requirements.

Service Experience

As documented in MRP-375, the resistance of Alloy 690 and corresponding weld metals Alloy 52 and 152 is demonstrated by the lack of any PWSCC indications reported in these materials in up to 24 calendar years of service for thousands of Alloy 690 steam generator tubes and more than 20 calendar years of service for thick-wall and thin-wall Alloy 690 applications. This operating experience includes service at pressurizer and hot leg temperatures and includes Alloy 690 wrought base metal and Alloy 52/152 weld metal. This experience also includes inservice volumetric/surface examinations performed in accordance with ASME Code Case N-729-1/N-729-4 on 13 of the 40 replacement RPVCHs currently operating in the U.S. fleet. This data supports a factor of improvement of at least 5 to 20 in time to detectable PWSCC when compared to the service experience of Alloy 600 in similar applications.

Factor of Improvement (FOI) Determination

Per the technical basis documents for ASME Code Case N-729-4 for heads with Alloy 600 nozzles [References 3 and 7], the effect of differences in operating temperature on the required volume/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 effective full power years (EFPYs) 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)(2), 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 an activation energy of 130 kJ/mol (31 kcal/mol) [Reference

8]. 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 Alloy 690/52/152 material of the replacement RPVCH. Key results from laboratory crack growth rate testing for Alloy 690 wrought material investigating the effect of temperature are as follows:

(1) Results from Argonne National Laboratory (ANL) indicate that the Alloy 690 with 0-26% cold work has an activation energy between 100 and 165 kJ/mol (24-39 kcal/mol) [Reference 9]. NUREG/CR-7137 [Reference 9] 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 (PNNL) found an activation energy of about 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work [Reference 10].

(3) Additional PNNL testing determined an activation energy of 123 kJ/mol (29.4 kcal/mol) for Alloy 690 with 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 Alloy 690/52/152 PWR plant components.

As discussed in the MRP-117 technical basis document [Reference 7] for heads with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate re-examination 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. There have been no reports of nozzle leakage or of safety-significant circumferential cracking during the time periods subsequent to when the Alloy 600 nozzles in a head were first examined by non-visual inservice non-destructive examination [References 12 and 13]. For a replacement RPVCH with Alloy 690/52/152 material that operates at 600°F, the implied FOI needed to support 20-year inspections is equal to 20 divided by 2.25, which is approximately 9, where 2.25 is RIY.

Factors of Improvement (FOI) for Crack Initiation

Alloy 690 is highly resistant to PWSCC due to its approximate 30% chromium content. Per MRP-115 [Reference 4], it was noted that Alloy 82 CGR is a factor of 2.6 slower than Alloy 182. There is no strong evidence for a difference in Alloy 52 and 152 CGRs. Therefore, data used to develop FOI for Alloy 52/152 were

referenced against the base case Alloy 182, as Alloy 182 is more susceptible to initiation and growth when compared to Alloy 82. A simple FOI approach was applied in a conservative manner in MRP-375 using multiple data. As discussed in MRP-375, laboratory and plant data demonstrate a FOI in excess of 20 in terms of the time to PWSCC initiation.

Factors of Improvement (FOI) for Crack Growth

MRP-375 also assessed laboratory PWSCC crack growth rate data for the purpose of assessing FOI values for growth. Data analyzed to develop a conservative FOI include laboratory specimens with substantial levels of cold work. Similar processing, fabrication, and welding practices apply to the original (Alloy 600) and replacement (Alloy 690) components. It is important to note that much of the data used to support Alloy 690 CGRs were produced using materials with significant amounts of cold work, which tends to increase the CGR. MRP-375 considered the most current worldwide set of available PWSCC CGR data for Alloy 690/52/152 materials.

Figure 3-2 of MRP-375 compares data from Alloy 690 specimens with less than 10% cold work and the statistical distribution from MRP-55 [Reference 5] describing the material variability in CGR for Alloy 600. Most of the laboratory comparisons were bounded by a factor of improvement of 20, and all were bounded by a factor of 10. Most data support a FOI of much larger than 20. This is similar for testing of the Alloy 690 Heat Affected Zone (HAZ) as shown in Figure 3-4 of MRP-375 and for the Alloy 52/152 weld metal as shown in Figure 3-6 of MRP-375. Based on the data, it is conservative to assume a FOI of between 10 and 20 for CGRs.

Deterministic Modeling

A deterministic crack growth evaluation is commonly applied to assess PWSCC risks for specific components and operating conditions. The deterministic evaluation is intended to demonstrate the time from an assumed initial flaw to some adverse condition.

Deterministic crack modeling results were presented in MRP-375 for previous references in which both growth of part-depth surface flaws and through-wall circumferential flaws were evaluated and normalized to an adjusted growth at 613°F to bound the PWR fleet. The time for through-wall crack growth in Alloy 600 nozzle tube material, when adjusted to a bounding temperature of 613°F, ranged between 1.9 and 3.8 EFPY. Assuming a growth FOI of 10 to 20 as previously established for Alloy 690/52/152 materials, the median time for through-wall growth was 37.3 EFPYs.

In a similar manner, crack growth results for through-wall circumferential flaws were tabulated and adjusted to a temperature of 613°F. Applying a growth FOI of 20 resulted in a median time of 176 EFPYs for growth of a through-wall circumferential flaw to 300 degrees of circumferential extent. The results of the generic evaluation are summarized in Table 4-1 of MRP-375. All cases were bounding and support an inspection interval greater than is being proposed. It is important to note that NAPS1 RPV head temperatures are 600 °F and within the bounds of the assumptions.

Note that for a head with Alloy 600 nozzles and Alloy 82/182 attachment welds operating at a temperature of 600°F, the re-inspection year [normalized to a reference temperature of 1059.67°R (588.71°K)] (RIY) = 2.25 constraint on the volumetric/surface reexamination interval of ASME Code Case N-729-4 corresponds to an interval of 2.25 EFPYs. Thus, a nominal interval of 20 calendar years for the NAPS replacement heads implies a FOI of less than 9 [Reference Attachment 2] versus the standard interval for heads with Alloy 600 nozzles. It is emphasized that the FOI of 9 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 EPRI MRP-375. Given the lack of PWSCC detected to date in any PWR plant applications of Alloys 690/52/152, the simple FOI assessment supports the requested period of extension.

Deterministic calculations performed in MRP-375 demonstrate that the alternative volumetric/surface re-examination interval is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size necessary to produce a nozzle ejection. The deterministic calculations also demonstrate that any base metal PWSCC would likely be detected prior to a through-wall flaw occurring.

Probability of Cracking or Through-Wall Leaks

Probabilistic calculations are based on a Monte Carlo simulation model of the PWSCC process, including PWSCC initiation, PWSCC crack growth, and flaw detection via ultrasonic testing and visual examinations for leakage. The basic structure of the probabilistic model is similar to that used in the MRP-105 technical basis report [Reference 6] for inspection requirements for heads with Alloy 600 nozzles. However, the current approach includes more detailed modeling of surface flaws (including multiple flaw initiation for each nozzle on base metal and weld surfaces), and the initiation module has been calibrated to consider the latest set of experience for U.S. heads. The outputs of the probabilistic model are leakage frequency (i.e., frequency of through-wall cracking) and nozzle ejection frequency.

Assuming conservatively small factors of improvement for the crack growth rate for the replacement nickel-base alloys (with no credit for improved resistance to initiation), the probabilistic results with the alternative inspection regime show:

1. An effect on nuclear safety substantially within the acceptance criterion applied in the MRP-117 [Reference 7] technical basis for Alloy 600 heads, and
2. A substantially reduced effect on nuclear safety compared to that for a head with Alloy 600 nozzles examined per current requirements.

Furthermore, the results confirm a low probability of leakage if some modest credit is taken for improved resistance to PWSCC initiation compared to that for Alloys 600 and 182.

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 HAZ) and its weld metals, Alloy 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.

Conclusion

In summary, the basis for extending the examination frequency from once each interval (nominally 10 calendar years) to once in 20 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 proposed alternative Alloy 690 examination frequency results in a substantially reduced effect on nuclear safety when compared to a RPVCH with alloy 600 nozzles examined per the current requirements. The proposed 20-year examination frequency will continue to provide reasonable assurance of structural integrity.

Dominion has replaced five RPVCHs with Alloy 690 and Alloy 52/152 weld metals. Three of these RPVCHs are of similar design, were fabricated by Framatome, and contain Alloy 690 Control Rod Drive Mechanism (CRDM) tubing from the same supplier. These three Dominion replacement RPVCHs were installed at NAPS1, NAPS2, and Surry Power Station Unit 1 (SPS1). The other two replacement RPVCHs, fabricated by Mitsubishi Heavy Industries, are located at Surry Power Station Unit 2 (SPS2) and Millstone Power Station Unit 2 (MPS2). Additional assurance of structural integrity is provided by successful completion of volumetric inspections for replacement RPVCHs in accordance with the requirements of Code Case N-729-1 at NAPS1, NAPS2, SPS1, SPS2 and MPS2 between 2007 and 2012. The proposed revised interval continues to provide reasonable assurance of structural integrity and thus an acceptable level of quality and safety.

The visual examinations and acceptance criteria as required by Item B4.30 of Table 1 of ASME Code Case N-729-4 are not affected by this request and will continue to be performed on a frequency of every third refueling outage or 5 calendar years, whichever is less. Furthermore, the VT-2 examination of the reactor vessel head will continue each refueling outage per the requirements of Section XI, Item No B15.10. As discussed in Section 5.2.3 of MRP-375, the visual examination requirement of the outer surface of the head for evidence of leakage supplements the volumetric/surface examination requirement and conservatively addresses the potential concern for boric acid corrosion of the low-alloy steel head due to PWSCC leakage.

For the reasons note above, it is requested that the NRC authorize this proposed alternative in accordance with 10 CFR 50.55a(z)(1) as the alternative provides an acceptable level of quality and safety.

6. <u>Duration of Proposed Alternative</u>

The provisions of this alternative are applicable to the fifth 10-year inservice inspection interval for NAPS1 which commenced on May 1, 2019 and ends on April 30, 2029, until the welds are examined during the sixth 10-year inservice inspection interval commencing May 1, 2029 and ending on April 30, 2039.

7. Precedents

NRC has previously approved submittals from multiple plants that requested an alternative from the frequency of ASME Code Case N-729-1 and N-729-4 for volumetric/surface examinations of heads with Alloy 690 nozzles. Examples of these plant requests and NRC approvals are provided in Table 1.

Table 1. NRC approved requests for relief from requirements of ASME Code Case N-729-1 and N-729-4

Plant	NRC ADAMS Accession No.				Status
	Relief Request	Request for Additional Information (RAI)	RAI Response	NRC Safety Evaluation	
Arkansas Nuclear One, Unit 1	ML14118A477	ML14258A020	ML14275A460	ML14330A207	Approved
Arkansas Nuclear One, Unit 1	ML16173A297	None	None	ML17018A283	Approved
Beaver Valley, Unit 1	ML14290A140	None	None	ML14363A409	Approved
Beaver Valley, Unit 1	ML17044A440	None	None	ML17222A162	Approved
Calvert Cliffs, Unit 1 & 2	ML15201A067	None	None	ML15327A367	Approved
Comanche Peak, Unit 1	ML15120A038	None	None	ML15259A004	Approved
D.C. Cook, Units 1 & 2	ML15023A038	None	None	ML15156A906	Approved
D.C. Cook, Units 1 & 2	ML18075A329	None	None	ML18103A059	Approved
J.M. Farley, Unit 2	ML15111A387	None	None	ML15104A192	Approved
North Anna, Unit 2	ML14283A044	None	None	ML15091A687	Approved
Prairie Island, Units 1 and 2	ML14258A124	ML15030A008	ML15036A252	ML15125A361	Approved
Palo Verde, Units 1, 2, and 3	ML17101A678	None	None	ML17306B432	Approved
H.B. Robinson, Unit 2	ML17269A016	None	None	ML18163A412	Approved
Salem, Unit 1	ML15098A426	None	None	ML15349A956	Approved
St. Lucie, Unit 1	ML17045A357	None	None	ML17219A174	Approved
St. Lucie, Unit 2	ML16076A431	None	None	ML16292A761	Approved

8. References

1. *Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)*, (EPRI 3002002441).

2. *Material Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52 and 152 in Pressurized Water Reactors (MRP-111)*, EPRI, Palo Alto, CA U.S. Department of Energy, Washington, DC: 2004. 1009801.

3. *Materials Reliability Program: Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants (MRP-110NP)*, EPRI, Palo Alto, CA: 2004, 1009807-NP.

4. *Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloys 82, 182 and 132 Welds (MRP-115)*, EPRI, Palo Alto, CA: 2004. 1006696.

5. *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: 2002. 1006695.

6. *Materials Reliability Program: Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking (MRP-105)*, EPRI, Palo Alto, CA: 2004. 1007834.

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.

8. 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.

9. U.S. NRC, *Stress Corrosion Cracking in Nickel-Base Alloys 690 and 152 Weld in Similated PWR Environment – 2009*, NUREG/CR-7137, ANL-10/36, published June 2012. [NRC ADAMS Accession No. ML12199A415].

10. *Materials Reliability Program: Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking* (MRP-237, Rev. 2); Summary of Findings Between 2008 and 2012 from Completed and Ongoing Test Programs, EPRI, Palo Alto, CA; 2013. 3002000190. [Freely available at www.epri.com].

11. M.B. Toloczko, M.J. Olszta, and S.M. Bruemmer, "One Dimensional Cold Rolling Effects on Stress Corrosion Crack Growth in Alloy 690 Tubing and Plate Materials," *15th International Conference on Environmental Degradation of*

Materials in Nuclear Power Systems – Water Reactors, TMS (The Minerals, Metals & Materials Society), 2011.

12. EPRI MRP Letter 2011-034, "T_{cold} RV Closure Head Nozzle Inspection Impact Assessment," dated December 21, 2011. [NRC ADAMS Accession No. ML12009A042].

13. G. White, V. Moroney, and C. Harrington, "PWR Reactor Vessel Top Head Alloy 600 CRDM Nozzle Inspection Experience," presented at *EPRI International BWR and PWR Material Reliability Conference*, National Harbor, Maryland, July 19, 2012.

14. *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.