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1CAN041402

April 28, 2014

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

SUBJECT: Request for Alternative from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1
Arkansas Nuclear One, Unit 1
Docket No. 50-313
License No. DPR-51

Dear Sir or Madam:

Pursuant to 10CFR50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) hereby requests NRC approval of the attached Inservice Inspection (ISI) Request for Alternative for Arkansas Nuclear One, Unit 1 (ANO-1). The request is associated with the volumetric/surface examination frequency requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Code Case N-729-1 as conditioned by 10CFR50.55a(g)(6)(ii)(D). Table 1, Item B4.40, of ASME Code Case N-729-1 requires that a volumetric/surface examination be performed within one inspection interval (nominally 10 calendar years) of its inservice date for a replacement reactor vessel closure head (RVCH). The ANO-1 replacement RVCH was placed in service in December 2005 and would nominally require volumetric/surface examination by December 2015. The next available ANO-1 refueling outage to comply with this examination under ASME Code Case N-729-1 will occur in January of 2015.

The Electric Power Research Institute (EPRI) recently published (February 2014) a technical report entitled “Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375).” This report provides justification for extending the volumetric/surface examination frequency from 10 years to 20 years. Entergy believes that the conclusions reached in this technical report are appropriate and applicable to establish an extended examination frequency for ANO-1. However, due to the expected time required for the NRC to review and accept the conclusions reached in this report, as well as the time to make appropriate ASME Code changes, Entergy is requesting a one-time deferral of the frequency requirements of Table 1 of ASME Code Case N-729-1, Item B4.40 for two (2) additional ANO-1 refueling cycles which corresponds to the refueling outage scheduled to commence in April of 2018. This is approximately 2.5 years beyond the nominal 10 years required by ASME Code Case N-729-1. The justification for this Alternative request is provided in the attachment to this letter. This Request for Alternative concludes that there is no significant likelihood for increased Primary Water Stress Corrosion Cracking and that this extension provides an acceptable level of quality and safety in accordance with 10CFR50.55a(a)(3)(i). This Request was discussed with members of the NRC Staff on March 18, 2014.

EPRI Technical Report MRP-375 is a non-proprietary document that is publically available through the EPRI Website.

This submittal contains no regulatory commitments.

The last refueling outage to comply with the one inspection interval requirement of ASME Code Case N-729-1 will commence in January 2015. In order to provide planning for this outage, Entergy requests approval of the proposed Request for Alternative by September 30, 2014.

If you have any questions or require additional information, please contact me.

Sincerely,

Original signed by Stephenie L. Pyle

SLP/sab

Attachment: Request for Alternative ANO1-ISI-024

cc: Mr. Marc L. Dapas
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Attachment to

1CAN041402

Request for Alternative
ANO1-ISI-024

REQUEST FOR ALTERNATIVE
ANO1-ISI-024

Inspection of Reactor Vessel Closure Head Nozzles
in Accordance with ASME Code Case N-729-1 as Conditioned by 10CFR50.55a

Components / Numbers:	Reactor Vessel Closure Head (RVCH) Penetration Nozzles 0-1 through 0-69
Code Classes:	American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), Class 1
Code References:	ASME Section XI, Division 1, Code Case N-729-1, as conditioned by 10CFR50.55a(g)(6)(ii)(D)
Examination Category:	Table 1 of ASME Code Case N-729-1, Item No., B4.40
Description:	Examination Categories for Class 1 Primary Water Reactor (PWR) Reactor Vessel Upper Head
Inspection Interval Applicability:	Arkansas Nuclear One, Unit 1 (ANO-1) / Fourth 10-Year Inservice Inspection Interval (ISI) Interval (May 31, 2008 through May 30, 2017)

I. CODE REQUIREMENTS

The Code of Federal Regulations 10CFR50.55a(g)(6)(ii)(D)(1), requires (in part):

> All licensees of pressurized water reactors shall augment their inservice inspection program with ASME Code Case N-729-1 subject to the conditions specified in paragraphs (g)(6)(ii)(D)(*2*) through (*6*) of this section. Licensees of existing operating reactors as of September 10, 2008 shall implement their augmented inservice inspection program by December 31, 2008.

10CFR50.55a(g)(6)(ii)(D)(3) conditions ASME Code Case N-729-1 by stating:

> Instead of the specified 'examination method' requirements for volumetric and surface examinations in Note 6 of Table 1 of Code Case N-729-1, the licensee shall perform volumetric and/or surface examination of essentially 100 percent of the required volume or equivalent surfaces of the nozzle tube, as identified by Figure 2 of ASME Code Case N-729-1. A demonstrated volumetric or surface leak path assessment, through all J-groove welds shall be performed. If a surface examination is being substituted for a volumetric examination on a portion of a penetration nozzle that is below the toe of the J-groove weld [Point E on Figure 2 of ASME Code Case N-729-1], the surface examination shall be of the inside and outside wetted surface of the penetration nozzle not examined volumetrically.

ASME Code Case N-729-1 (Reference 1) specifies that the reactor vessel upper head components shall be examined on a frequency in accordance with Table 1 of this code case.

II. REQUEST FOR ALTERNATIVE

Pursuant to 10CFR50.55a(a)(3)(i), Entergy Operations, Inc. (Entergy) requests an alternative from performing the required volumetric/surface examinations for the ANO-1 RVCH components identified above at the frequency prescribed in ASME Code, Section XI, Code Case N-729-1. Specifically, Entergy requests to extend the frequency of the volumetric/surface examination of the ANO-1 RVCH of Table 1, Item B4.40 of ASME Code Case N-729-1 for approximately 2.5 years beyond the one inspection interval (nominally 10 calendar years) from installation of the ANO-1 replacement RVCH. This request would extend the volumetric/surface examination to the 27th refueling outage which is scheduled to commence in April 2018. .

III. BASIS FOR ALTERNATIVE

The original ANO-1 RVCH, which was manufactured with Alloys 600/82/182 materials, was replaced with a new RVCH using Alloys 690/52/152 material during the refueling outage that returned to operation in December 2005. In accordance with Table 1 of ASME Code Case N-729-1, Item B4.40, as conditioned by 10CFR50.55a(g)(6)(ii)(D)(3), Entergy will be required to perform a volumetric and/or surface examination of essentially 100% of the RVCH the end of 2015.

The basis for the inspection frequency for ASME Code Case N-729-1 comes, in part, from the analysis performed in Electric Power Research Institute (EPRI) Materials Reliability Program (MRP)-111 (Reference 2) which was summarized in the safety assessment for RVCHs in EPRI MRP-110 (Reference 3). The material improvement factor for Primary Water Stress Corrosion Cracking (PWSCC) of Alloys 690/52/152 materials over that of mill annealed Alloys 600/82/182 was shown by these reports to be in the order of 26 or greater.

<u>Additional Evaluations Performed under EPRI MRP-375</u>

Further evaluations were performed to demonstrate the resistance of Alloys 690/52/152 to PWSCC under a recent EPRI MRP initiative provided in EPRI MRP-375 (Reference 4). This report presents both deterministic and probabilistic evaluations that assess the improved PWSCC resistance of Alloys 690/52/152.

Operating experience to date for replacement and repaired components using Alloys 690/52/152 has shown a proven record of resistance to PWSCC during numerous examinations in the 20+ years of its application. This includes steam generators, pressurizers, and RVCHs. In particular, Alloys 690/52/152 operating experience includes inservice volumetric/surface examinations performed on nine of the 40 plant replacement RVCHs in the US in accordance with ASME Code Case N-729-1. Some of these heads had continuous full power operating temperatures that may approach 613°F. None of these examinations had revealed PWSCC cracking.

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 Alloy 690 compared to Alloy 600 at equivalent temperature and stress conditions. Even though base metal and welding variability of test data exist (i.e. heat affected zones, weld dilution zones, etc.), relative, but conservative, FOIs were estimated for the material improvements of Alloys 690/52/152 materials using an extensive database of test data. Results for both crack initiation and crack growth conclude a higher resistance to PWSCC for Alloy 690 base material and Alloy 52/152 weld materials. EPRI MRP-375, Figures 3-2, 3-4, and 3-6 provide crack growth data for Alloy 690/52/152 materials and heat affected zones with represented curves plotting FOIs of 1, 5, 10, and 20. An FOI of 20 bounds most of the data plotted, however, an FOI of 10 or less bounds all of the data[1].

EPRI MRP-375 then performed a combination of deterministic and probabilistic evaluations for establishing a reasonable 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 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 time to adverse conditions, from which a *conservative* inspection interval may be recommended. This evaluation draws from various EPRI MRP and industry documents. The results show a substantial improvement of Alloys 690/52/152 over that of Alloys 600/82/182 for the time between a detectable flaw being created and the time to leakage and between a leakage flaw to the time that net section collapse (nozzle ejection) would be predicted to occur.

The probabilistic model in EPRI 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 EPRI MRP-335, Rev. 1 (Reference 5) that evaluated Alloy 600 RVCHs for surface stress improvement. The integrated probabilistic model in EPRI 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 (RPVHPNs) in MRP-335, Rev. 1 were adapted for Alloy 690 RVCHs by applying FOIs to account for superior PWSCC resistance. The probabilistic calculations are based on a Monte Carlo simulation model including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing. The average leakage frequency and average ejection frequency were determined using conservative FOI assumptions. The results show that using only modest FOIs for Alloys 690/52/152 RVCHs, the potential for developing a safety significant flaw (risk of nozzle ejection) is acceptably small for a volumetric/surface examination period of 20 years.

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[1] As discussed in Section 3.3 of MRP-375, the laboratory crack growth rate data compiled in MRP-375 represent the values reported by individual researchers, without any adjustment by the authors of EPRI MRP-375 other than for temperature and stress intensity factor. The data presented in Figures 3-2, 3-4, and 3-6 of EPRI MRP-375 represent essentially the entire set of data points reported by the various laboratories. No screening process was applied to the data on the basis of test characteristics such as minimum required crack extension or minimum required engagement to intergranular cracking. Instead, an inclusive process was applied to conservatively assess the factors of improvement apparent in the data for specimens with less than 10% added cold work.

The evaluations performed in EPRI 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).

While Entergy is not requesting NRC review and approval of EPRI MRP-375 to approve this Request for Alternative, the insights gained in this technical report help substantiate the limited extension duration being requested for ANO-1 of approximately 2.5 years beyond the 10 year examination frequency established in ASME Code Case N-729-1. In particular, the tabulation of crack growth rate data for Alloys 690/52/152 (Section 3 of EPRI MRP-375) and review of inspection experience for Alloys 690/52/152 plant components (Section 2 of EPRI MRP-375) are sufficient to demonstrate the acceptability of the limited extension duration being requested. This request is not dependent on the more detailed probabilistic calculations presented in Section 4 of EPRI MRP-375.

ANO-1 Replacement RVCH Design and Operation

The analysis performed by EPRI MRP-375 bounds the design and operation of the ANO-1 Replacement RVCH. The RVCH contains sixty-nine (69) flanged nozzle housing penetrations of which sixty-eight (68) are used for control rod drive mechanisms (CRDMs) and the remaining nozzle housing penetration (center of RVCH) is used for supporting the reactor vessel inadequate core cooling monitoring and display instruments. The Replacement RVCH was manufactured by Framatome (AREVA) and placed in service in December 2005. The replacement RVCH was manufactured as a single forging which eliminated the center disc and flange circumferential weld in the original ANO-1 RVCH. The replacement RVCH is fabricated from SA-508, Class 3 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. The nozzle J-groove welds utilized ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152) weld materials.

A preservice volumetric examination of the ANO-1 Replacement RVCH J-groove welded CRDM nozzles and a preservice liquid penetrant examination of the outer periphery CRDM nozzle to flange dissimilar metal welds was performed by AREVA in 2004. 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 volumetric examinations. The 24 outer periphery nozzle dissimilar metal welds examined by liquid penetrant also had no recordable indications.

Similarly, the NRC concluded that the ANO-1 Replacement RVCH met its design requirements as documented in an Inspection Report dated February 13, 2006 (Reference 6) using Inspection Procedure 71007, “Reactor Vessel Head Replacement Inspection”. The inspectors reviewed numerous design and manufacturing documents including the certified material test reports, heat treatment records, welding processes, as well as the preservice volumetric examinations. No findings of significance were identified regarding the ANO-1 Replacement RVCH.

The EPRI MRP-375 analyses assume a reactor vessel head operating temperature of 613ºF. This assumed head temperature is based on the reporting from certain Babcock & Wilcox (B&W) designed plants as having an operating head temperature that approach 613ºF as discussed in EPRI MRP letter 2011-034 (Reference 7). Entergy was notified by AREVA NP in 2010 that actual RVCH temperatures are in the range of 5ºF to 10ºF above reactor coolant system (RCS) hot leg temperature due to a portion of the RCS fluid experiencing non-homogeneous mixing prior to reaching the RVCH. The design and operating hot leg temperature for ANO-1 is 602ºF which would represent average RVCH temperatures that may be upwards of 612ºF. Based on this, the ANO-1 RVCH average operating temperature (which is the measure of temperature relevant to potential PWSCC degradation) is bounded by the EPRI MRP-375 evaluation results.

Entergy has also assessed the representative Alloy 690/52/152 FOI for the requested ANO-1 extension period based on the full set of laboratory crack growth rate data. ASME Code Case N-729-1 is based upon conclusions reached that a head with Alloy 600 nozzles and operating at a temperature of 605ºF is safe to operate up to 2 years (one 24 month operating cycle) between volumetric/surface examinations. The same period for Alloy 690 RVCHs is 10 years which represents a factor of 5 over Alloy 600. A simple extension of that improvement factor to 12.5 years would be a factor of 6.25 for the period between volumetric/surface examinations for ANO-1. However, the RVCH operating temperature assumed in the technical basis for heads with Alloy 600 nozzles (References 3, 8, and 9) for ASME Code Case N-729-1 was 605ºF, compared to an assumed operating temperature of 613°F for ANO-1. Code Case N-729-1 addresses the effect of differences in operating temperature on the required volumetric/surface reexamination interval for heads with Alloy 600 nozzles on the basis of the Reinspection 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-729-1 as conditioned by 10CFR50.55a 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. As discussed in the technical basis documents 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. There have been no reports of nozzle leakage or of safety-significant circumferential cracking for times subsequent to the time that the Alloy 600 nozzles in a head were first examined by non-visual inservice non-destructive examination (References 7 and 10).

The representative ANO-1 RVCH operating temperatures of 613ºF would result in an RIY temperature adjustment factor of 1.379 (versus the reference temperature of 600ºF) using the activation energy of 31 kcal/mol for crack growth of ASME Code Case N-729-1[2]. Conservatively assuming that the EFPYs of operation accumulated at ANO-1 since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended period at ANO-1 would be (1.379)(12.5) = 17.24. The FOI implied by this RIY value for ANO-1 is (17.24)/(2.25) = 7.7. 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 years that bounds the crack growth rate data presented. Furthermore, as discussed in Sections 2 and 3 of EPRI

[2] Laboratory PWSCC crack growth rate testing for Alloy 690 wrought material by multiple investigators (References 11, 12, and 13) has shown thermal activation energy values comparable to the standard activation energy applied to model growth of Alloys 600/82/182 (31 kcal/mol or 130 kJ/mol). Thus, it is appropriate to apply this standard activation energy for modeling crack growth of Alloy 690/52/152 plant components.

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 limited requested period of extension.

A bare metal visual examination was performed in 2010 on the ANO-1 replacement RVCH in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. This visual examination was performed by VT-2 qualified examiners on the outer surface of the RVCH including the annulus area of the penetration nozzles. This examination did not reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage. This examination will be performed again in the upcoming 25th refueling outage scheduled to commence in January 2015.

In addition, based on communications with Duke Energy for Oconee Units 1, 2, and 3 and Exelon Generation for Three Mile Island (TMI) Unit 1, these units received head replacements in the 2003 to 2004 timeframe. The replacement RVCHs for the Oconee units were manufactured by B&W Canada and the TMI RVCH was manufactured by AREVA (similar to ANO-1). These four units have received volumetric head examination in accordance with ASME Code Case N-729-1. These examinations did not reveal any recordable indications. Being B&W plant designs, these units would have similar head configurations and design operating conditions to that of ANO-1. Entergy believes that these examination results additionally support the low likelihood of the potential to experience PWSCC for the ANO-1 RVCH for the extension period.

IV. PROPOSED ALTERNATIVE EXAMINATIONS

No alternative examination processes are proposed to those required by ASME Code Case N-729-1, as conditioned by 10CFR50.55a(g)(6)(ii)(D). The visual (VT-2) examinations and acceptance criteria as required by Item B4.30 of Table 1 of ASME Code Case N-729-1 are not affected by this request and will continue to be performed on a frequency not to exceed every 5 calendar years.

V. DURATION OF PROPOSED ALTERNATIVE REQUEST

The proposed Alternative is requested for the duration up to and including the 27th ANO-1 refueling outage that is schedule to commence in April 2018 and which will occur in the fifth ten-year ISI inspection interval.

VI. PRECEDENT

The purpose of EPRI MRP-375 is to support obtaining approval to implement the proposed alternative inspection regime, either through relief of current NRC requirements or revision of the ASME Code inspection regime followed by NRC acceptance. No ASME Code alternative requests using the application of EPRI MRP-375 are known to have been submitted at this time. However, Entergy is aware of other licensees who intend to request an alternative using the insights gained from EPRI MRP-375.

VII. CONCLUSION

10CFR50.55a(g)(6)(i) states: “The Commission will evaluate determinations under paragraph (g)(5) of this section that Code requirements are impractical. The Commission may grant such relief and may impose such alternative requirements as it determines is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that could result if the requirements were imposed on the facility”.

10CFR50.55a(a)(3) states: “Proposed alternatives to the requirements of paragraphs (c), (d), (e), (f), (g), and (h) of this section, or portions thereof, may be used when authorized by the Director, Office of Nuclear Reactor Regulation, or Director, Office of New Reactors, as appropriate. Any proposed alternatives must be submitted and authorized prior to implementation. The applicant or licensee shall demonstrate that: (i) The proposed alternatives would provide an acceptable level of quality and safety; or (ii) Compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.”

Entergy believes that the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the ANO-1 replacement RVCH provide for a clearly 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 remote. This is further supported by visual examination of the ANO-1 RVCH in 2010 and the volumetric examinations performed by other B&W designed plants during their nominal 10-year examination under similar operating conditions which did not reveal PWSCC.

The 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 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 clearly supports the limited requested period of extension. Therefore, the ANO-1 RVCH FOI corresponding to the requested period of extension to perform a volumetric/ surface examination provides an acceptable level of quality and safety in accordance with 10CFR50.55a(a)(3)(i).

VIII. REFERENCES

1. ASME Code Case N-729-1, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds, Section XI, Division 1," Approved March 28, 2006.

2. EPRI MRP-111, “Resistance to Primary Water Stress Corrosion Cracking of Alloys 690, 52, and 152 in Pressurized Water Reactors,” Report No. 1009801, March 2004 (ML041680546).

3. EPRI MRP-110, “Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants,” Report No. 1009807, April 2004 (ML041680506).

4. EPRI MRP-375, “Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles”, Report No. 3002002441, February 2014 (publically available at www.epri.com)

5. EPRI MRP-335 (Rev. 1), "Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement,” Report No. 3002000073, January 2013. Transmitted from EPRI to NRC under letter dated May 1, 2013 (ML13126A009)

6. NRC Inspection Report for Arkansas Nuclear One - NRC Integrated Inspection Report 05000313/2005010 dated February 13, 2006 (ML060460164)

7. EPRI MRP Letter 2011-034, “T_{cold} RV Closure Head Nozzle Inspection Impact Assessment,” dated December 21, 2011 (ML12009A042)

8. EPRI MRP-117, “Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants,” Report No. 1007830, December 2004 (ML043570129).

9. EPRI MRP-105, “Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking,” Report No. 1007834, April 2004 (ML041680489).

10. 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.

11. 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, published June 2012 (ML12199A415).

12. EPRI MRP-237 (Rev. 2), “Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking: Summary of Findings Between 2008 and 2012 from Completed and Ongoing Test Programs,” Report No. 3002000190, April 2013 (publically available at www.epri.com)

13. 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.