Exelon Generation.

200 Exelon Way
Kennett Square, PA 19348

www.exeloncorp.com

10 CFR 50.55a

July 13, 2015

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

Calvert Cliffs Nuclear Power Plant, Units 1 and 2
Facility Operating License Nos. DPR-53 and DPR-69
NRC Docket Nos. 50-317 and 50-318

Subject: Relief Request for Extension of Volumetric Examination Interval for Reactor Vessel Heads with Alloy 690 Nozzles

Reference: 1) Letter from J. Barstow (Exelon Generation Company, LLC) to U.S. Nuclear Regulatory Commission, "Relief Request for Extension of Volumetric Examination Interval for Reactor Vessel Heads with Alloy 690 Nozzles," dated January 8, 2015

In the Reference 1 letter, Exelon Generation Company, LLC (Exelon) requested relief from the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." This relief requests an extension of the volumetric examination interval for Calvert Cliffs Nuclear Power Plant (CCNPP), Units 1 and 2. Based on further discussions with the U.S. Nuclear Regulatory Commission staff on July 9, 2015, this relief request is being revised to limit the relief to 16 years. Attached is a revised relief request with this revision identified with revision bars.

There are no regulatory commitments in this letter.

If you have any questions concerning this letter, please contact Tom Loomis at (610) 765-5510.

Respectfully,

James Barstow

James Barstow
Director - Licensing & Regulatory Affairs
Exelon Generation Company, LLC

A047
NRR

Attachments:
1) Relief Request ISI-024
2) "TECHNICAL NOTE - Assessment of Laboratory PWSCC Crack Growth Rate Data Compiled for Alloys 690, 52, and 152 with Regard to Factors of Improvement (FOI) versus Alloys 600 and 182," TN-3688-00-01, Revision 0, December 2014.
3) Calculation No. C-3688-00-01, "Minimum Factor of Improvement Implied by Extension of CCNPP RVCH UT Inspection Interval," Revision 0, December 1, 2014.

cc: Regional Administrator, Region I, USNRC
USNRC Senior Resident Inspector, CCNPP
Project Manager [CCNPP] USNRC
S. T. Gray, State of Maryland

ATTACHMENT 1

Relief Request ISI-024

Request for Relief for Extension of Volumetric Examination Interval for Reactor Vessel Heads with Alloy 690 Nozzles in accordance with 10 CFR 50.55a(z)(2)

1.0 ASME CODE COMPONENT(S) AFFECTED:

Component Numbers:	Calvert Cliffs Nuclear Power Plant (CCNPP), Unit 1 and Unit 2 Reactor Vessel Heads (Component No. 729-1-4)
Code Class:	Class 1
Examination Category:	ASME Code Case N-729-1
Code Item:	B4.40
Component Identification:	All reactor vessel closure head nozzles and welds
Drawing Numbers:	Various

2.0 APPLICABLE CODE EDITION AND ADDENDA:

The fourth Inservice Inspection (ISI) interval code of record for CCNPP, Unit 1 and Unit 2 is the 2004 Edition without Addenda of the ASME Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components." Examinations of the Reactor Vessel Closure Head (RVCH) penetrations are performed in accordance with 10 CFR 50.55a(g)(6)(ii)(D), which specifies the use of Code Case N-729-1, with conditions.

3.0 APPLICABLE CODE REQUIREMENT:

10 CFR 50.55a(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."

ASME Code Case N-729-1, "-2410," specifies that the reactor vessel upper head penetrations (nozzles and partial-penetration welds) shall be examined on a frequency in accordance with Table 1 of this code case. The basic inspection requirements of Code Case N-729-1, as amended by 10 CFR 50.55a, for partial-penetration welded Alloy 690 head penetration nozzles are as follows:

- Volumetric or surface examination of all nozzles every ASME Section XI 10-year ISI interval (provided that flaws attributed to Primary Water Stress Corrosion Cracking (PWSCC) have not been identified).

- Direct Visual Examination (VE) of the outer surface of the head for evidence of leakage every third refueling outage or 5 calendar years, whichever is less.

4.0 REASON FOR REQUEST:

Code Case N-729-1 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. Performance of these examinations results in hardships that are not compensated for by increases in safety or quality for the following reasons:

- These examinations require personnel accessing the underside of the head, which is highly contaminated. The dose for performance of the inspection is estimated to be approximately 3.0 Rem for each inspection.
- Work performed within containment inherently includes some industrial safety risks associated with lifting of heavy loads and working within confined spaces. Eliminating inspections reduces risk of industrial injury.
- Additionally, these inspection activities have the potential to impact the outage critical path should difficulties be encountered.

Treatment of Alloy 690 RVCH nozzles and associated welds in Code Case N-729-1 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) Report MRP-375 was developed to support a technically based volumetric or surface re-examination 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.

5.0 PROPOSED ALTERNATIVE AND BASIS FOR USE:

Exelon is requesting relief from the exam frequency requirements of Code Case N-729-1 [1], Item B4.40 for performing volumetric and/or surface exams of the CCNPP, Units 1 and 2 RVCHs. The Unit 1 reactor vessel head was replaced in the spring 1RFO18 refueling outage which ended April 12, 2006. The Unit 2 reactor vessel head was replaced in the spring 2RFO17 which ended April 2, 2007. Specifically, this would allow volumetric or surface examinations currently scheduled for the spring of 2016 and the spring of 2017 to be extended to 2022 for Unit 1 and 2023 for Unit 2 (not to exceed 16 calendar years in order to align with scheduled refueling outages). This request applies to the Item B4.40 inspection frequencies only.

As discussed in the original ASME technical basis document [4], the inspection frequency of ASME Code Case N-729-1 [1] for heads with Alloy 690 nozzles and Alloy 52/152 attachment welds is based, in part, on the analysis of laboratory and plant data presented in report MRP-111 [2], which was summarized in the safety assessment for RVCHs in MRP-110 [3]. The material improvement factor for PWSCC of Alloy 690

materials over that of mill-annealed Alloy 600 material was shown by this report to be on the order of 26 or greater. The 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 [4].

Further evaluations were performed to demonstrate the acceptability of extending the inspection intervals for Code Case N-729-1 [1], Item B4.40 components and documented in MRP-375 [5]. In summary, the basis for extending the intervals from once each interval (nominally 10 calendar years) to once every 16 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 for Alloys 690/52/152 versus Alloys 600/182/82 (for equivalent temperature and stress conditions) in terms of Crack Growth Rates (CGRs). 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. However, since work is still ongoing to determine the performance of Alloys 690/52/152 metals, the determination of the proposed inspection interval is based on conservatively smaller factors of improvement.

Deterministic calculations demonstrate that the alternative volumetric reexamination schedule is sufficient to detect any PWSCC before it could develop into a safety significant circumferential flaw that approaches the large size (i.e., more than 300°) 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 22 calendar years of service for thick-wall and thin-wall Alloy 690 applications. This excellent 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 includes ISI volumetric or surface examinations performed in accordance with ASME Code Case N-729-1 on 13 of the 41 replacement RPVCHs currently operating in the U.S. fleet. This data supports a factor of improvement in time of at least 5 to 20 to detectable PWSCC when compared to service experience of Alloy 600 in similar applications.

One of the replacement heads that was volumetrically examined in accordance with N-729-1 was R.E. Ginna Nuclear Power Plant. That head was fabricated by the same manufacturer (Babcock & Wilcox Canada) using Alloy 690 nozzle material produced by

the same material supplier (Valinox) per the same nozzle material specifications (SB-167 and ASME Boiler and Pressure Vessel Code, Section III, 1995 Edition with 1996 Addenda). All the control rod/element drive mechanism (CRDM/CEDM) nozzles in the replacement heads at CCNPP, Units 1 and 2 and Ginna were thermally treated and electropolished. Both Ginna and CCNPP are owned by Exelon. As stated above, none of the prior examinations of replacement RVCHs with Alloy 690 nozzles has revealed any indications of PWSCC or service-induced cracking.

Factors of Improvement (FOI) for Crack Initiation

Alloy 690 is highly resistant to PWSCC due to its approximate 30% chromium content. Per MRP-115 [6], it was noted that Alloy 82 CGR is 2.6 times slower than Alloy 182. There is no strong evidence for a difference in Alloy 52 and 152 CGRs. Therefore data used to develop factors of improvement 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 factor of improvement approach was applied in a conservative manner in MRP-375 using multiple data. As discussed in MRP-375, laboratory and plant data demonstrate a factor of improvement in excess of 20 in terms of the time to PWSCC initiation. Conservatively, credit was not taken for the improved resistance of Alloys 690/52/152 to PWSCC initiation in the MRP-375 analyses.

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 factor of improvement include laboratory specimens with substantial levels of cold work. It is important to note that much of the data used to support Alloy 690 CGRs was produced using materials with significant amounts of cold work, which tends to increase the CGR. Similar processing, fabrication, and welding practices apply to the original (Alloy 600) and replacement (Alloy 690) components. MRP-375 considered the most current worldwide set of available PWSCC CGR data for Alloys 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 [7] 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 (relative to the distribution from MRP-55) and for the Alloy 52/152 weld metal (relative to the distribution from MRP-115 [6]) 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.

Note that for a head with Alloy 600 nozzles and Alloy 82/182 attachment welds operating at a temperature of 605°F, the reinspection years (RIY) = 2.25 constraint on the volumetric or surface reexamination interval of ASME Code Case N-729-1 correspond to an interval of 2.0 Effective Full Power Years (EFPYs). Thus, a nominal interval of 20 calendar years for the Calvert Cliffs replacement heads implies a FOI of 7.6 versus the standard interval for heads with Alloy 600 nozzles. It is emphasized that the FOI of 7.6 (reference Attachment 3) 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 by the laboratory data compiled in EPRI MRP-375 when material variability is accounted for. 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.

Attachment 2 provides further support for the requested alternative inspection interval based on the available laboratory PWSCC crack growth rate data and the FOI approach. The attachment provides responses to the requests for additional information that NRC has transmitted to other licensees in the context of similar relief requests (see Section 7.0 Precedent). Attachment 2 describes the materials tested for data points within a factor of 7.6 below the MRP-55 [7] and MRP-115 [6] crack growth rate curves for the 75th percentile of material variability. Attachment 2 also compares these test materials to the specific nozzle and weld materials used in the CCNPP replacement heads. It is concluded that the available crack growth rate data do not indicate any susceptibility concerns specific to the nozzle or weld materials of the CCNPP replacement heads.

Design Features Further Increasing the Resistance of the Calvert Cliffs Replacement Heads to PWSCC

Methods were used to reduce residual stress caused by J-groove welding by applying a narrow gap for weld preparation (lower weld volume [8]) and employing automated welding methods (eliminating stress raising features of manual welding, such as frequent stops and starts). The straightening operations involved in manufacture of the thermally treated Alloy 690TT nozzles were controlled in a manner that minimized cold work during straightening and thereby minimized the as-fabricated yield strength, resulting in lower weld residual stresses [9]. The nozzle inner surface and the wetted outer surface of the nozzle and J-groove weld were electropolished to remove the surface cold worked layer associated with machining and grinding of these areas, further reducing the welding induced residual stresses in these areas ([10] and [11]).

These methods substantially reduce PWSCC susceptibility beyond that assumed in the generic MRP-375 study, resulting in additional assurance that the CCNPP, Units 1 and 2 heads can be operated for 20 years prior to their next volumetric and/or surface examination with an acceptable level of quality and safety. Performance of the volumetric and/or surface examination after 10 years would constitute a hardship without a compensating increase in level of quality or safety.

Previous Examinations of the Calvert Cliffs Replacement Heads

A preservice volumetric examination of the replacement RVCH partial-penetration welded nozzles was performed prior to head installation at both CCNPP, Unit 1 and Unit 2. There were no recordable indications identified during the preservice volumetric examinations of the nozzle tube in the area of the J-groove welds.[1]

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[1] Furthermore, the NRC concluded that the Calvert Cliffs replacement RVCH met its design requirements as documented in an Inspection Report dated May 5, 2006 for Unit 1 [12] and in an Inspection Report dated May 7, 2007 for Unit 2 [13] 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 replacement RVCH.

A bare metal VE was performed of the CCNPP, Unit 1 replacement RVCH in 2014 in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. The most recent VE of the CCNPP, Unit 2 replacement RVCH was in 2011. These 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.

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 of 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 EFPYs. Assuming a growth FOI of 10 to 20 as previously established for Alloys 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 the operating temperature of the CCNPP RVCH is 594°F and well within the bounds of the assumptions.

Deterministic calculations performed in MRP-375 demonstrate that the alternative volumetric 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 [14] technical basis report for inspection requirements for heads with Alloy 600 nozzles, but the current approach includes more detailed modeling of flaw initiation and growth (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. Even 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 [15] 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 modest credit is taken for improved resistance to PWSCC initiation compared to that for Alloys 600 and 182.

Conclusion

In summary, the basis for extending the intervals from once each interval (nominally 10 calendar years) to once every 16 calendar years is based on plant service experience, factor of improvement studies using laboratory initiation and growth data, deterministic modeling, and probabilistic study results. The results of the analysis show that the alternative proposed frequency results in a substantially reduced effect on nuclear safety when compared to a head with Alloy 600 nozzles and examined per the current requirements. 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 by the laboratory data compiled in MRP-375 when accounting for heat-to-heat variability of Alloy 600 and weld-to-weld variability of Alloy 82/182/132. The proposed revised interval will continue to provide reasonable assurance of structural integrity.

Additional assurance of structural integrity is provided by the design features of the CCNPP replacement heads such as the narrow groove weld joint preparation design and by electropolishing the wetted surfaces of the penetration prior to operation. Furthermore, the visual 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 of every third refueling outage or 5 calendar years, whichever is less. 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 and/or surface examination requirement and conservatively addresses the potential concern for boric acid corrosion of the low-alloy steel head due to PWSCC leakage.

The requested period of extension to perform a volumetric and/or surface examination provides an acceptable level of quality and safety. Furthermore, the interval for volumetric and/or surface examination of N-729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) constitutes a hardship without a compensating increase in level of quality or safety in accordance with 10 CFR 50.55a(z)(2).

6.0 DURATION OF PROPOSED ALTERNATIVE:

The proposed Alternative is requested for the remainder of the fourth and fifth ISI intervals because utilizing the proposed examination frequency will require the examination to be performed in the fifth interval.

7.0 PRECEDENT:

There have been submittals from multiple plants to request an alternative from the frequency of ASME Code Case N-729-1 for volumetric or surface examinations of heads with Alloy 690 nozzles. The first of these was Arkansas Nuclear One, Unit 1, and some subsequent requests including the associated status at the time of submittal of the original revision of this request are shown below:

Plant	NRC ADAMS Accession No.			Status
	Relief Request	Request for Additional Information (RAI)	RAI Response	
Arkansas Nuclear One, Unit 1	ML14118A477	ML14258A020	ML14275A460	Under NRC Review
Beaver Valley, Unit 1	ML14290A140			Under NRC Review
J.M. Farley, Unit 2	ML14280A260			Under NRC Review
North Anna, Unit 2	ML14283A044			Under NRC Review
Prairie Island, Units 1 and 2	ML14258A124			Under NRC Review
H.B. Robinson, Unit 2	ML14251A014	ML14294A587	ML14325A693	Under NRC Review
St. Lucie, Unit 1	ML14206A939	ML14251A222	ML14273A011	Under NRC Review

8.0 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. Materials 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. [freely available at www.epri.com; NRC ADAMS Accession No. ML041680546]
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. [ML041680506]
4. ASME Section XI, Code Case N-729, Technical Basis Document, dated September 14, 2004.
5. 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: 2014. 3002002441. [freely available at www.epri.com]
6. Materials Reliability Program Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115), EPRI, Palo Alto, CA: 2004. 1006696. [freely available at www.epri.com]

7. Materials Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Materials (MRP-55) Revision 1, EPRI, Palo Alto, CA: 2002. 1006695. [freely available at www.epri.com]
8. Proceedings: 1997 EPRI Workshop on PWSCC of Alloy 600 in PWRs - Part 1, EPRI, Palo Alto, CA. 1997. TR-109138-P1.
9. PWSCC of Alloy 600 Materials in PWR Primary System Penetrations, EPRI, Palo Alto, CA. 1994. TR-103696.
10. P. Berge, et al., PWSCC- Effects of Initial Surface Preparation, Proceedings: 1997 EPRI Workshop on PWSCC of Alloy 600 in PWRs - Part 2, EPRI, Palo Alto, CA. 1997. TR-109138-P2.
11. S. Le Hong, Influence of Surface Condition on Primary Water Stress Corrosion Cracking Initiation of Alloy 600, Corrosion, v57n4p323-333. April 2001.
12. Letter from U.S. NRC to J.A. Spina, Calvert Cliffs Nuclear Power Plant – NRC Integrated Inspection Report 05000317/2006002 AND 05000318/2006002, dated May 5, 2006. [ML061250346]
13. Letter from U.S. NRC to J.A. Spina, Calvert Cliffs Nuclear Power Plant – NRC Integrated Inspection Report 05000317/2007002 AND 05000318/2007002, dated May 7, 2007. [ML071270730]
14. Materials Reliability Program: Probabilistic Fracture Mechanics Analysis of PWR Reactor Pressure Vessel Top Head Nozzle Cracking (MRP-105 NP), EPRI, Palo Alto, CA: 2004. 1007834. [ML041680489]
15. Materials Reliability Program: Inspection Plan for Reactor Vessel Closure Head Penetrations in U.S. PWR Plants (MRP-117), EPRI, Palo Alto, CA: 2004. 1007830. [freely available at www.epri.com; NRC ADAMS Accession No. ML043570129]

ATTACHMENT 2

"TECHNICAL NOTE - Assessment of Laboratory PWSCC Crack Growth Rate Data Compiled for Alloys 690, 52, and 152 with Regard to Factors of Improvement (FOI) versus Alloys 600 and 182," TN-3688-00-01, Revision 0, December 2014.

Dominion Engineering, Inc.

TECHNICAL NOTE

Assessment of Laboratory PWSCC Crack Growth Rate Data Compiled for Alloys 690, 52, and 152 with Regard to Factors of Improvement (FOI) versus Alloys 600 and 182

TN-3688-00-01
Revision 0

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Principal Investigators
K. Fuhr
G. White

Prepared for
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

12100 Sunrise Valley Drive, Suite 220 ■ Reston, VA 20191 ■ PH 703.657.7300 ■ FX 703.657.7301

RECORD OF REVISIONS

Rev.	Description	Prepared by Date	Checked by Date	Reviewed by Date	Approved by Date
0	Original Issue	K. J. Fuhr 12/1/2014 K. J. Fuhr Associate Engineer	P. P. Loo 12/1/2014 P. P. Loo Engineering Technician	G. A. White 12/1/2014 G. A. White Principal Engineer	G. A. White 12/1/2014 G. A. White Principal Engineer

The last revision number to reflect any changes for each section of the technical note is shown in the Table of Contents. The last revision numbers to reflect any changes for tables and figures are shown in the List of Tables and the List of Figures. Changes made in the latest revision, except for Rev. 0 and revisions which change the technical note in its entirety, are indicated by a double line in the right hand margin as shown here.

CONTENTS

LIST OF FIGURES

ACRONYMS

ANL	Argonne National Laboratory
ASME	American Society of Mechanical Engineers
AWS	American Welding Society
BWC	Babcock & Wilcox Canada
CCNPP	Calvert Cliffs Nuclear Power Plant [Unit 1 & Unit 2]
CEDM	Control Element Drive Mechanism
CGR	Crack Growth Rate
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas
CRDM	Control Rod Drive Mechanism
CT	Compact Tension
DEI	Dominion Engineering, Inc.
EPRI	Electric Power Research Institute
FOI	Factor of Improvement
GE-GRC	General Electric Global Research Center
GTAW	Gas Tungsten Arc Welding
HAZ	Heat Affected Zone
ICI	In-Core Instrumentation
K	Stress Intensity Factor
MRP	Materials Reliability Program
NRC	Nuclear Regulatory Commission
PNNL	Pacific Northwest National Laboratory
PPU	Partial Periodic Unloading
PWR	Pressurized Water Reactor
PWSCC	Primary Water Stress Corrosion Cracking
RIY	Re-Inspection Year
RV	Reactor Vessel
RVCH	Reactor Pressure Closure Head
UNS	Unified Numbering System

1 INTRODUCTION

The purpose of this DEI technical note is to examine laboratory crack growth rate (CGR) data for primary water stress corrosion cracking (PWSCC) compiled for Alloys 690, 52, and 152 to identify test specimens within a factor of 7.6 improvement below the MRP-55 and MRP-115 deterministic CGR for Alloys 600 and 182. Furthermore, those test specimens are compared to the specific nozzle and weld material used in the current Calvert Cliffs Nuclear Power Plant Unit 1 and Unit 2 (CCNPP) replacement reactor vessel closure heads (RVCHs).

As discussed in Section 3 of Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) report MRP-375 [1], a conservative approach was taken in MRP-375 to develop the factor of improvement (FOI) values describing the PWSCC crack growth rates applicable to Alloy 690 reactor vessel (RV) top head penetration nozzles. The crack growth rate data points presented in Figures 3-1, 3-3, and 3-5 of MRP-375 represent the values reported by individual researchers, without any adjustment by the authors of MRP-375 other than to normalize for the effect of temperature. The data in these figures represent essentially all of the Alloys 690, 52, and 152 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 extent of transition along the crack front 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 percent added cold work.

The approach was conservative in that no effort was made to screen out data points reflecting tests that are not applicable to plant conditions. Instead the data were treated on a statistical basis in Figures 3-2, 3-4, and 3-6 of MRP-375,* and compared to the crack growth rate variability due to material variability for Alloy 600 in MRP-55 [2] and Alloy 182 in MRP-115 [3]. A comparison between the cumulative distributions of the crack growth rates for Alloys 690/52/152 and Alloys 600/82/182 treats the full variability in both original and replacement alloys, rather than comparing the variability of the replacement alloy against a conservative mean (75th percentile) growth rate for the original alloys. By considering the cumulative distributions, a fuller perspective of the improved resistance of Alloys 690/52/152 emerges where over 70% of

* Figures 3-2, 3-4, and 3-6 of MRP-375 show cumulative distribution functions of the variability in crack growth rate normalized for temperature and crack loading (i.e., stress intensity factor). Each ordinate value in the plots shows the fraction of data falling below the corresponding crack growth rate. Thus the cumulative distribution function has the benefit of illustrating the variability in crack growth rate for a standard set of conditions.

the data in each of Figures 3-2, 3-4, and 3-6 of MRP-375 indicate a factor of improvement beyond 20 and all of the data* correspond to a factor of improvement of 12 or greater. As described below, nearly all of the data points for the conditions directly relevant to plant conditions (e.g., constant load conditions) fall a factor of 7.6 times below the deterministic MRP-55 and MRP-115 equations.

The deterministic MRP-55 and MRP-115 crack growth rate equations were developed not to describe bounding crack growth rate behavior but rather reflect 75th percentile values of the variability in crack growth rate due to material variability. Twenty-five percent of the material heats (MRP-55) and test welds (MRP-115) assessed in these reports on average showed crack growth rates exceeding the deterministic equation values. Thus, the appropriate FOI comparisons are made on a statistical basis (e.g., Figures 3-2, 3-4, and 3-6 of MRP-375). Comparing the crack growth rate for Alloys 690/52/152 versus the deterministic crack growth rate lines in Figures 3-1, 3-3, and 3-5 of MRP-375 represents an unnecessary compounding of conservatisms. None of the data presented lies within a statistical FOI of 7.6 below the MRP-55 and MRP-115 distributions of material variability. The technical basis for the inspection requirements for heads with Alloy 600 nozzles ([4], [5], [6]) are based on the full range of crack growth rate behavior, including heat-to-heat (weld-to-weld) and within-heat (within-weld) material variability factors. Thus, the Re-Inspection Year (RIY) = 2.25 inspection interval developed for heads with Alloy 600 nozzles reflects the possibility of crack growth rates being many times higher than the deterministic 75th percentile values per MRP-55 and MRP-115. Nevertheless, as described below, nearly all of the data points for the conditions directly relevant to plant conditions (e.g., constant load conditions) fall below a line a factor of 7.6 times below the deterministic MRP-55 and MRP-115 equations.

2 DISCUSSION OF DATA POINTS FROM MRP-375 [1]

2.1 *Data Points Above a Hypothetical 7.6 Factor of Improvement Line in Figure 3-1, 3-3, and 3-5 of MRP-375*

- *Figure 3-1 of MRP-375.* Figure 3-1 shows the complete set of data points compiled by the PWSCC Expert Panel organized by EPRI at the time MRP-375 was completed for Alloy 690 specimens with less than 10% added cold work. The following points are within a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600:

* Excluding data points reflecting fatigue pre-cracking conditions that are not relevant to PWSCC.

- There are 10 points within a factor of 7.6 below the MRP-55 75th percentile curve, out of a total of 75 points shown in Figure 3-1 of MRP-375.
- These data represent test segments from four distinct Alloy 690 Compact Tension (CT) specimens that were tested by Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT) and one that was tested by Argonne National Laboratory (ANL). All but three of the data points are for partial periodic unloading (PPU) conditions, which are discussed below.
- Two of the points tested by CIEMAT are from specimen 9ARB1, comprised of Alloy 690 plate material, loaded to 37 $MPa(m)^{0.5}$, and tested at 340°C and 15 cc H_2/kg H_2O [7]. Both of these data are for the first half of segments that exhibited a crack growth rate that was an order of magnitude lower in the second half of the segment. A plot of crack growth rate versus crack-tip stress intensity factor (K) for the Alloy 690 data from MRP-375 for plate material tested by CIEMAT is provided here as Figure 1. These two points have minimal implications for the requested inspection interval extension for several reasons:
  - As illustrated in Figure 1 and subsequent figures using open symbols, one of the two points was generated under partial periodic unloading conditions. As discussed below in Section 2.2, PPU conditions may result in accelerated crack growth rates that are not directly representative of plant conditions, especially for the case of alloys with relatively high resistance to environmental cracking like Alloy 690. The other data point obtained under constant load/K conditions is only slightly above the line representing a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600.
  - U.S. pressurized water reactors (PWRs) operate with a dissolved hydrogen concentration per EPRI guidelines in the range of 25-50 cc/kg for Mode 1 operation. Testing at 15 cc/kg results in accelerated crack growth rates versus that for normal primary water due to the proximity of the Ni-NiO equilibrium line [3].
  - Specimens fabricated from Alloy 690 plate material are not as relevant to plant RV top head penetration nozzles as specimens fabricated from control rod drive mechanism (CRDM) / control element drive mechanism (CEDM) nozzle material. CRDM and CEDM nozzles in U.S. PWRs are fabricated from extruded pipe or bar stock material. Note that term CRDM nozzle is used henceforth to refer to both CRDM and CEDM nozzles (CEDM is the terminology used by plants designed by Combustion Engineering).
  - The wide variability in crack growth rate within even the same testing segment indicates that significant experimental variability exists. Thus, there is a substantial possibility that a limited number of elevated growth rate data points do not reflect the true characteristic behavior of the material tested.
- The remaining seven CIEMAT points are from specimens comprised of Valinox material heat WP787 CRDM nozzle material that was cold worked by a 20% tensile elongation (9.1% thickness reduction) and tested at 325°C and 35 cc H_2/kg H_2O [8]. One of the data is from specimen 9T2 (loaded to roughly 36 $MPa(m)^{0.5}$), and the remaining six data are from specimens 9T5 or 9T6 (loaded to roughly 27 $MPa(m)^{0.5}$). The results for 9T2 are contained in Reference [8]; the final data for 9T5 and 9T6 are

contained in EPRI MRP-340, but have not been openly published. As discussed later in Section 2.4, the addition of cold work may result in a material that is substantially more susceptible than the as-received material. The extent of transition along the crack front to intergranular cracking for these data was extremely low (≤ 5%). A plot of crack growth rate versus K for the Alloy 690 data from MRP-375 for heat WP787 is provided here as Figure 2. As in Figure 1, there is significant growth rate variability within the data for the same heat of material. The median for the CIEMAT specimens is more than a factor of 12 below the MRP-55 curve. Four of the five points are for PPU testing; this method may accelerate growth beyond what would be expected for in-service components, as discussed later. Additionally, the Pacific Northwest National Laboratory (PNNL) data indicate that the specific laboratory that produces the data can significantly influence the reported growth rate, such that there is a substantial possibility that a small number of reported data points with relatively high crack growth rates from a single laboratory are not characteristic of the true susceptibility of a specific heat of Alloy 690 material.

– The ANL datum is for CT specimen C690-LR-2, comprised of Valinox heat number WP142 CRDM nozzle material that was not cold worked and was tested at roughly 21 $MPa(m)^{0.5}$, 320°C, and 23 cc H_2/kg H_2O [9]. The intergranular engagement for this specimen was extremely low (almost entirely transgranular). A plot of crack growth rate versus K for the Alloy 690 data from MRP-375 for heat WP142 is provided here as Figure 3. As in Figure 2, PNNL data indicate that the specific laboratory that produces the data can significantly influence the reported growth rate.

- *Figure 3-3 of MRP-375.* Figure 3-3 shows the complete set of data points compiled by the PWSCC Expert Panel organized by EPRI at the time MRP-375 was completed for Alloy 690 heat affected zone (HAZ) specimens. The following points are within a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600:
  – There are five points within a factor of 7.6 below the MRP-55 75th percentile curve, out of a total of 34 points shown in Figure 3-3 of MRP-375. All five of the data points are from PPU testing. All of the data appear to have had very little to no intergranular engagement.
  – Three of the points are from ANL testing of specimens comprised of Valinox CRDM nozzle material heat WP142 and Alloy 152 filler (Special Metals heat WC43E9), tested under PPU at 320°C and 23 cc H_2/kg H_2O [10]. Two of the points are from specimen CF690-CR-3 (loaded to roughly 28 $MPa(m)^{0.5}$) [10], and the other point is from specimen CF690-CR-4 (loaded to roughly 22 $MPa(m)^{0.5}$) [11]. A plot of crack growth rate versus K for all the Alloy 690 HAZ data from MRP-375 for heat WP142 is provided here as Figure 4. As discussed below, the PPU conditions may result in accelerated crack growth relative to plant conditions, and the three points within a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600 are only slightly within the factor of 7.6.
  – The remaining two points are from CIEMAT testing of specimens 19ARH1 and 19ARH2, comprised of welded Alloy 690 plate material, tested at 340°C and 15 cc H_2/kg H_2O, and loaded to roughly 37 $MPa(m)^{0.5}$ [7]. A plot of crack growth rate versus K for the Alloy 690 HAZ data from MRP-375 for plate material tested by CIEMAT is shown in Figure 5. As discussed later, the orders of magnitude difference between these two PPU points and the constant load testing for this HAZ is indicative

of the substantial accelerating effect that PPU testing can have beyond what would be expected in service environments.

- *Figure 3-5 of MRP-375.* Figure 3-5 shows the complete set of data points compiled by the PWSCC Expert Panel organized by EPRI at the time MRP-375 was completed for Alloy 52 and 152 weld metal specimens. The following points are within a factor of 7.6 below the MRP-115 deterministic crack growth rate for Alloy 182:
  - There are 11 points within a factor of 7.6 below the MRP-115 75th percentile curve, out of a total of 212 points shown in Figure 3-5 of MRP-375. Three of these points are not relevant to PWR conditions and should not be considered further, as discussed in the following bullets. The eight relevant data are from ANL testing, and six of these are for PPU testing.
    - One of these points is from PNNL testing of the dilution zone of a dissimilar metal weld between 152M (Special Metals heat WC83F8) and carbon steel, tested at 360°C and 25 cc H_2/kg H_2O [12]. This material condition is not applicable to the wetted surfaces of CRDM nozzle J-groove welds because the dilution zone where Alloy 52/152 contacts the low-alloy steel RV head is below the stainless steel cladding. A plot of crack growth rate versus K for the Alloy 152 data from MRP-375 for heat WC83F8 is provided here as Figure 6.
    - Two of the remaining points, including the point closest to the MRP-115 curve, are for environmental fatigue pre-cracking test segments [13]. (Similarly, two of the data points more than a factor of 7.6 below the MRP-115 curve are for environmental fatigue pre-cracking test segments [13].) The status of these four data points, which are shown in black in Figure 7, as being fatigue pre-cracking test segments irrelevant to PWSCC conditions was clarified subsequent to publication of MRP-375.
  - The remaining eight data points represent four specimens from Alloy 152 weld material (Special Metals heat WC04F6) that were tested by ANL at 320°C and 23 cc H_2/kg H_2O ([14] and [9]). The specimens were predominantly loaded to between 27 and 32 MPa(m)$^{0.5}$, but one datum is from testing at 46 MPa(m)$^{0.5}$. These Alloy 152 specimens came from welded plate material. A plot of crack growth rate versus K for the Alloy 152 data from MRP-375 for heat WC04F6 is provided here as Figure 7. Six of these eight points were for PPU conditions. Figure 7 shows a very large variability in the crack growth rate reported by different laboratories for this heat of Alloy 152 weld material. Roughly one third the ANL data, all of the General Electric Global Research Center (GE-GRC) data, and all the PNNL data for this heat are for specimens from a single weld made by ANL [15], illustrating the role of experimental variability. A small number of elevated data points for a weld produced by a single laboratory may not be representative of the true material susceptibility.

2.2 Data Most Directly Applicable to Plant Conditions

As described above, Section 3 of MRP-375 took an inclusive approach to statistical assessment of the compiled data. A conservative approach was applied in which both constant load data and data under PPU conditions were plotted together. In addition, weld data reflecting various levels

of weld dilution adjacent to lower chromium materials was included in the data for Alloys 52/152. An assessment of the crack growth rate data points most applicable to plant conditions is presented in Figure 8 through Figure 13. The assessment shows very few points located within a factor of 7.6 below the deterministic MRP-55 and MRP-115 lines, with all such points only slightly above the line representing a factor of 7.6:

- Figure 8 for Alloy 690 with Added Cold Work Less than 10%.
  - Only three points are within a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600.
  - Figure 9 shows that the data are bounded by a FOI of more than 12 relative to Alloy 600 data on a statistical basis.
- Figure 10 for Alloy 690 HAZ.
  - None of the 24 points are within a factor of 7.6 below the MRP-55 deterministic crack growth rate for Alloy 600.
  - Figure 11 shows that the data are bounded by a FOI of more than 12 relative to Alloy 600 data on a statistical basis.
- Figure 12 for Alloys 52/152.
  - Only two of 83 points are within a factor of 7.6 below the MRP-115 deterministic crack growth rate for Alloy 182.
  - Figure 13 shows that the data are bounded by a FOI of more than 12 relative to Alloy 182 data on a statistical basis.

As discussed above, the technical basis for heads with Alloy 600 nozzles assumes the substantial possibility of crack growth rates substantially greater than that predicted by the deterministic equations of MRP-55 and MRP-115. The MRP-55 and MRP-115 deterministic crack growth rate equations are not bounding equations, but rather reflect the 75th percentile of material variability. Thus, the perspective provided in Figure 9, Figure 11, and Figure 13 is most relevant to drawing conclusions regarding FOI values applicable to inspection intervals for heads with nozzles fabricated using Alloy 690, 52, and 152 materials.

The data presented in Figure 8 through Figure 13 were included on the basis of the following considerations:

- As demonstrated and discussed in MRP-115, certain PPU conditions will act to accelerate the crack growth rate. PPU conditions, which include a periodic partial reduction in load, are often used in testing to transition from initial fatigue conditions toward constant load conditions with the crack in a state most representative of stress corrosion cracks if they had initiated in plant components over long periods of time. The periodic load reductions and accompanying load increases may rupture localized crack ligaments along the crack front, facilitating transition of the crack to an intergranular morphology. In MRP-115, data with hold times less than 1 hour were screened out of the database for Alloys 82/182/132.

The greater resistance of Alloys 690/52/152 to cracking is expected to result in a greater sensitivity of the crack growth rate to partial periodic unloading conditions. Figure 1 and Figure 5 show that there is an apparent significant bias for the data for Alloy 690 in which the data for partial periodic unloading conditions are substantially higher than for constant load conditions. Thus, the data presented in Figure 8 through Figure 13 have been restricted to the constant load (or constant K) conditions that are most relevant to plant conditions for growth of stress corrosion cracks.

- The Alloy 52/152 weld metal data shown in Figure 3-5 and Figure 3-6 of MRP-375 include data reflecting a range of weld dilution levels. The data presented in Figure 12 and Figure 13 exclude the weld dilution data points. The weld dilution data are not reflective of the full chromium content of Alloy 52/152 weld metal.
- The data presented in Figure 12 and Figure 13 exclude a small number of data points that reflect cracking at the fusion line with carbon or low-alloy steel material. Some of these data reflect cracking in the adjacent carbon or low-alloy steel material that was not post-weld heat treated as would be the case in plant applications.
- The data presented in Figure 12 and Figure 13 eliminate the few data points that in fact reflect fatigue pre-cracking rather than stress corrosion cracking. The status of these data points was clarified subsequent to publication of MRP-375.
- As noted above, of the data most relevant to plant conditions, a total of only five data points are located within a factor of 7.6 below the standard deterministic crack growth rate lines. The three remaining points in Figure 8 that lie within a factor of 7.6 below the deterministic MRP-55 line represent only a small fraction of the data and are only slightly within the factor of 7.6. These three points represent two specimens from Alloy 690 material heat WP787 and one from Alloy 690 plate material that was tested by CIEMAT. As discussed above, most of the data points for heat WP787 are far below a line that is a factor of 7.6 below the deterministic MRP-55 line, meaning that the two points for heat WP787 above this line are not representative of the susceptibility for this heat. The single data point produced by CIEMAT above a line that is a factor of 7.6 below the deterministic MRP-55 line is for a material form (i.e., plate material) not directly representative of reactor vessel head nozzle material, and was produced for a hydrogen concentration outside of the normal range for PWR operation. The two remaining points in Figure 12 that lie within a factor of 7.6 below the deterministic MRP-115 line represent only a small fraction of the data and are only slightly within the factor of 7.6. These two points represent specimens from two Alloy 152 test welds produced using filler material heat WC04F6 that were tested by ANL. The other data points produced by ANL for these two test welds are substantially lower.

The limited number of remaining points in Figure 8 and Figure 12 that lie within a factor of 7.6 below the deterministic MRP-55 and MRP-115 lines represent the upper end of material and/or experimental variability. Figure 9, Figure 11, and Figure 13 consider the variability in crack growth rate among different heats/welds of Alloys 600/82/182 and compare this against the full variability of the Alloy 690/52/152 data most applicable to plant conditions. The lack of *any* points within a factor of 12 when accounting for variability in Alloy 600/82/182 crack growth rates supports a reexamination interval longer than the requested interval corresponding to a FOI

of 7.6. The volumetric or surface inspection interval for heads with Alloy 600 nozzles reflects consideration of crack growth rates on a statistical basis, with crack growth rates often higher than that given by the deterministic equations of MRP-55 and MRP-115.

2.3 Data Specific to Argonne National Laboratory (ANL) and Pacific Northwest National Laboratory (PNNL)

An assessment of the crack growth rate data points particular to data generated by ANL and PNNL, including data with up to 20% cold work, was performed because the U.S. Nuclear Regulatory Commission (NRC) is most familiar with the testing performed at these two national laboratories. The large majority of the data from ANL and PNNL in MRP-375 [1] are openly published in NRC contractor reports, conference papers, and Research Collaboration Meeting presentations (e.g., Reference [10] through [15]). Only 2 of the total of 94 constant load (or constant K) data points from ANL and PNNL are within a factor of 7.6 below the deterministic MRP-55 and MRP-115 lines.

2.4 Data for Alloy 690 Wrought Material Including Added Cold Work up to 20% for CRDM Nozzle and Bar Material Product Forms

An assessment of the crack growth rate data points for Alloy 690 CRDM nozzle and bar material product forms for cold work levels up to 20% indicates that only two of the 33 points between 10% and 20% cold work are within a factor of 7.6 below the deterministic MRP-55 curve. An equivalent assessment for Alloy 52/152 material for the purpose of including the limited number (i.e., five) of weld metal data points generated for added cold work conditions indicates that none are within a factor of 7.6 below the MRP-115 curves. Added cold work for weld metals is not directly relevant to plant material conditions.

For Alloy 690 control rod drive mechanism (CRDM) / control element drive mechanism (CEDM) nozzles and other RV head penetration nozzles, the effective cold-work level in the bulk Alloy 690 base metal is expected to be no greater than roughly 10%. This is based on fabrication practices specific to replacement heads, i.e., material processing and subsequent nozzle installation via welding [16]. Furthermore, the crack growth rate data presented for Alloy 600 in MRP-55 do not include cases of added cold work. Comparing cold worked Alloy 690 data against non-cold worked Alloy 600 data results in a conservatism in the factor of improvement for Alloy 690 material as the cold worked material condition for Alloy 600 would be expected to result in a somewhat increased deterministic crack growth rate for Alloy 600, and thus a greater apparent factor of improvement. Nevertheless, this assessment is included in this document to

consider the effect of higher levels of cold work. These data show the potential for modestly higher crack growth rates for such elevated cold work levels for the material product forms most relevant to RV top head nozzles.

Based on the discussion above and the presentation of data in Figure 1 through Figure 13, it is shown that the factor of improvement for the replacement head materials is conservatively greater than that needed to justify the requested inspection deferral at Calvert Cliffs, Unit 1 and Unit 2.

3 SIMILARITIES BETWEEN THE CRACK GROWTH RATE TESTING SPECIMENS AND THE CCNPP RVCH PENETRATION NOZZLE MATERIAL

3.1 *Fabrication Details for the Calvert Cliffs Replacement Reactor Vessel Closure Heads and Penetration Nozzles*

The Calvert Cliffs RVCHs were designed and fabricated using materials and techniques to reduce susceptibility to PWSCC and facilitate prompt detection of potential leakage by visual examination. Each RVCH contains seventy (70) nozzle penetrations of which sixty-one (61) are used for control element drive mechanisms (CEDM), eight (8) are used for in-core instrumentation (ICI), and one is a small-diameter vent line penetration near the center of the RVCH [17]. Both replacement RVCHs were manufactured by Babcock & Wilcox Canada (BWC) in Cambridge, Ontario, Canada, and BWC also was the welding organization [18]. The Alloy 690 nozzle material used in the heads was supplied by Valinox Nucleaire [19]. The replacement heads were manufactured as two piece welded forgings and were procured to ASME Section III, 1995 Edition with 1996 Addenda [17]. Each replacement RVCH is fabricated from SA-508, Grade 3, Class 1 low-alloy steel and clad with an initial layer of 309L stainless steel followed by subsequent layers of 308L stainless steel [17]. The CEDM and ICI nozzles on each replacement RVCH are fabricated from thermally treated SB-167 (Alloy 690TT) UNS N06690 tubing, and the vent pipe on each head was also made from SB-167 (Alloy 690) piping [17]. The penetration nozzle material was procured to ASME Section III, 1995 Edition with 1996 Addenda and Code Case N-747-2 [17]. The nozzle J-groove welds utilized Unified Numbering System (UNS) N06052 / American Welding Society (AWS) ERNiCrFe-7 (Alloy 52 – Gas Tungsten Arc Welding (GTAW)) [17]. The replacement RVCH at Unit 1 was placed into service in April 2006 [20], and the replacement RVCH at Unit 2 was placed into service in April 2007 [21].

3.2 Similarities Between Laboratory Specimen Material and CCNPP CEDM Nozzle Material

Any similarities between (a) the data points above a hypothetical 7.6 factor of improvement line in Figure 3-1, 3-3, and 3-5 of MRP-375 and (b) the associated nozzles and weld material used in the current reactor pressure vessel upper heads at CCNPP are as follows:

- *Figure 3-1 of MRP-375* [1]. The Alloy 690TT nozzle material used in the heads at CCNPP was supplied by Valinox Nucleaire. Seven of the Alloy 690 data points above a crack growth rate 7.6 times lower than the MRP-55 [2] deterministic crack growth rate in Figure 3-1 of MRP-375 were produced for specimens of Alloy 690 CEDM nozzle material that was supplied by Valinox Nucleaire. However, for the reasons explained below (e.g., the variability among data from different laboratories, the variability among data for a single heat and laboratory, and the use of PPU for four of these five data), this similarity in no way indicates any specific concern for elevated PWSCC susceptibility of the head nozzles at CCNPP in comparison to other heads with Alloy 690 nozzles. Furthermore, it is noted that the thermal treatment and electroplating of the CEDM nozzle material at Calvert Cliffs serve to improve the material resistance to PWSCC. The 28 data points for thermally treated material in Figure 3-1 of MRP-375 are more than a factor of 150 below the deterministic 75th percentile curve of MRP-55.
- *Figure 3-3 of MRP-375* [1]. Three of the Alloy 690 HAZ data points above a crack growth rate 7.6 times lower than the MRP-55 deterministic crack growth rate in Figure 3-3 of MRP-375 were also produced for specimens of Alloy 690 CRDM nozzle material that was supplied by Valinox Nucleaire. However, for the reasons explained below, this similarity in no way indicates any specific concern for elevated PWSCC susceptibility of the head nozzles at CCNPP in comparison to other heads with Alloy 690 nozzles. It is noted that the welding process used to produce the HAZ in the test specimens is not specific to any particular categories of replacement heads.
- *Figure 3-5 of MRP-375* [1]. There are no relevant similarities between (a) the Alloy 52 and 152 data points above a crack growth rate 7.6 times lower than the MRP-115 [3] Alloy 182 deterministic crack growth rate in Figure 3-5 of MRP-375 and (b) the Alloy 52/152 weld material used in the RV top head at CCNPP. The variability among test welds with respect to PWSCC crack growth susceptibility reflects a combination of how the weld was made (welding procedure, weld design, degree of constraint, etc.) and perhaps the material variability in the weld consumable (e.g., composition). The test welds used to produce the specimens that showed crack growth rates within a factor of 7.6 below the MRP-115 crack growth rate are not identified with any particular fabricator of replacement RV heads. Furthermore, the weld specimens used in the crack growth rate testing were machined from test welds in flat plates, not from actual J-groove welds. Thus, the test weld specimens should not be associated with particular fabrication categories of replacement heads.

3.3 Implications of the Similarities for the CCNPP Relief Request

The material and welding similarities in no way indicate any specific concern for elevated PWSCC susceptibility of the head nozzles at CCNPP in comparison to other heads with Alloy

690 nozzles. It is emphasized that a small number of data points showing relatively high crack growth rates cannot readily be concluded to be characteristic of the true material behavior expected in the field. This conclusion is made considering the following:

- The data points showing the highest crack growth rates for the tested Valinox material reflect partial periodic unloading conditions. As discussed above, such conditions tend to result in accelerated crack growth rates that are not representative of plant conditions.
- Most of the crack growth rate data for heats that had points within a factor of 7.6 below the MRP-55 deterministic curve or MRP-115 deterministic curve were substantially lower. Thus, the best-estimate behavior for every heat or test weld of material presented in Figures 3-1, 3-3, and 3-5 of MRP-375 reflects a factor of improvement substantially greater than 7.6. In addition, other factors being equal, one would expect a greater range of crack growth rates for a material heat for which a greater number of data points was produced. Some of the scatter likely reflects experimental uncertainty as opposed to true material variability. Experimental uncertainty is more of a factor for the data for Alloys 690/52/152 than for Alloys 600/82/182/132 considering the greater testing challenges associated with the more resistant replacement alloys.
- In some cases, different laboratories have reported large differences in crack growth rate for the same material heat or test weld. This behavior is illustrated in Figure 7 for the Alloy 152 heat WC04F6 and Figure 3 for the Alloy 690 heat WP142. Thus, individual data points showing relatively high crack growth rates might not reflect the true susceptibility of particular categories of nozzle or weld material. Consistent data from multiple laboratories may be needed before one can conclude that a particular category of nozzle or weld material has an elevated susceptibility to PWSCC growth.
- Some type of PWSCC initiation is necessary to produce a flaw that may grow via PWSCC. Laboratory and plant experience show that Alloys 690/52/152 are substantially more resistant to PWSCC initiation than Alloys 600/82/182 [1]. PWSCC has not been shown to be an active degradation mode for Alloys 690/52/152 components after use in PWR environments for over 25 years.
- The crack growth rate data compiled in MRP-375 [1] for Alloys 52 and 152 reflect the composition variants applicable to PWR plant applications. Data are included for the following variants: Alloy 52 (UNS N06052 / AWS ERNiCrFe-7), Alloy 52M (UNS N06054 / AWS ERNiCrFe-7A), Alloy 52MSS (UNS N06055 / AWS ERNiCrFe-13), Alloy 52i (AWS ERNiCrFe-15), Alloy 152 (UNS W86152 / AWS ENiCrFe-7), and Alloy 152M (UNS W86152 / AWS ENiCrFe-7). Considering the overall set of available crack growth rate data for the various variants of Alloy 52 and 152, there is no basis for concluding that the variant used at CCNPP (Alloy 52) is of specific concern for relatively high crack growth rates.

In addition, it should be recognized that PWSCC of Alloy 690 RV head penetration nozzles or their Alloy 52/152 attachment welds is not an active degradation mode. Thus, it is premature to single out individual materials or fabrication categories of heads with Alloy 690 nozzles for additional scrutiny on the basis of subsets of laboratory crack growth rate data. In the case of heads with Alloy 600 nozzles, for which PWSCC is an active degradation mode, materials and

fabrication categories of heads with relatively high incidence of PWSCC are inspected in accordance with the same requirements as other heads.

Based on the additional information and discussion provided above, it is concluded that the available crack growth rate data do not indicate any susceptibility concerns specific to the nozzle or weld materials specific to the CCNPP replacement heads.

Data from Individual Heats

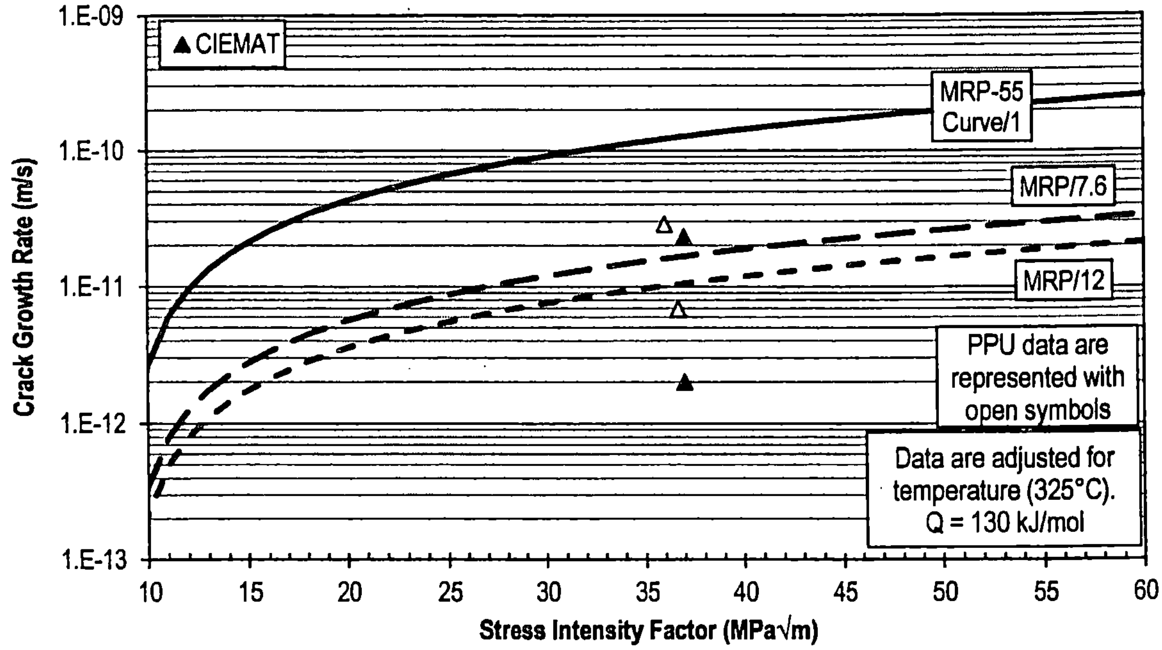

Figure 1. **Plot of Crack Growth Rate (*da/dt*) versus Stress Intensity Factor (K_I) for Alloy 690 Data from Plate Material Tested by CIEMAT**

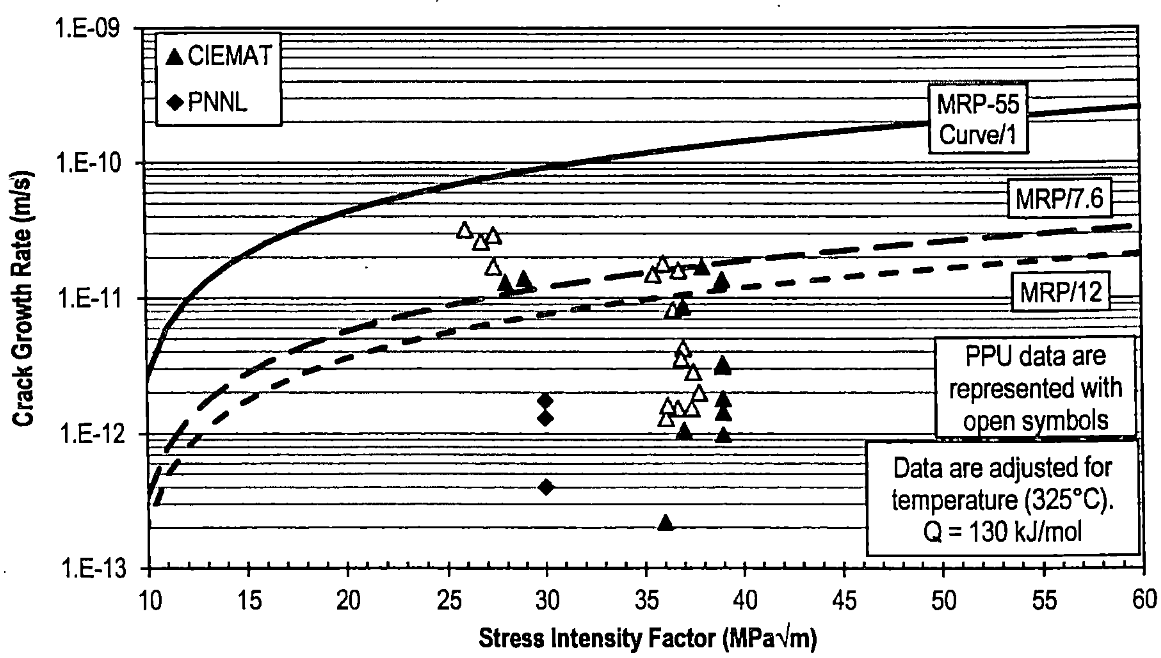

Figure 2. **Plot of *da/dt* versus K_I for Alloy 690 Data from Heat WP787**

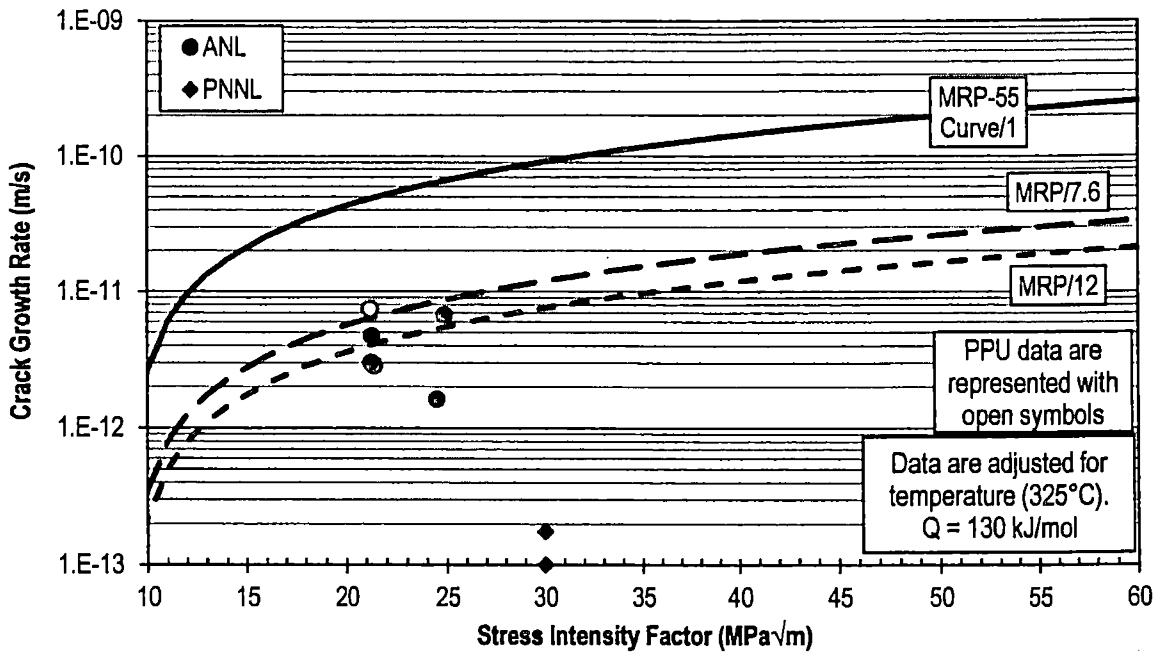

Figure 3. Plot of *da/dt* versus K_I for Alloy 690 Data from Heat WP142

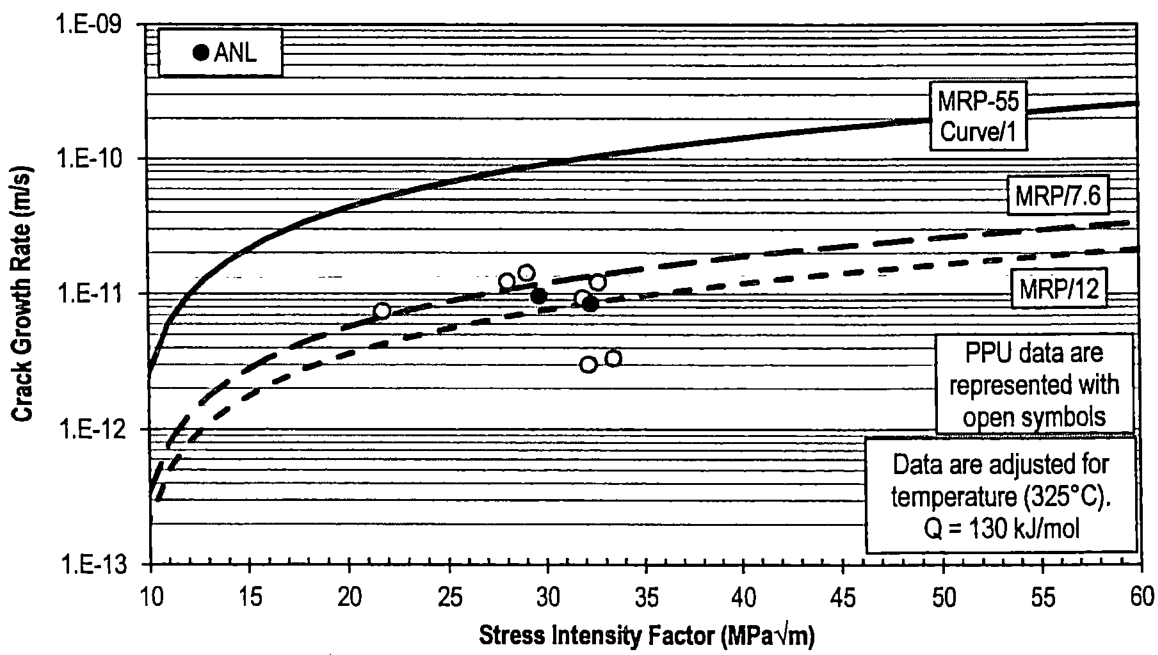

Figure 4. Plot of *da/dt* versus K_I for Alloy 690 HAZ Data from Heat WP142

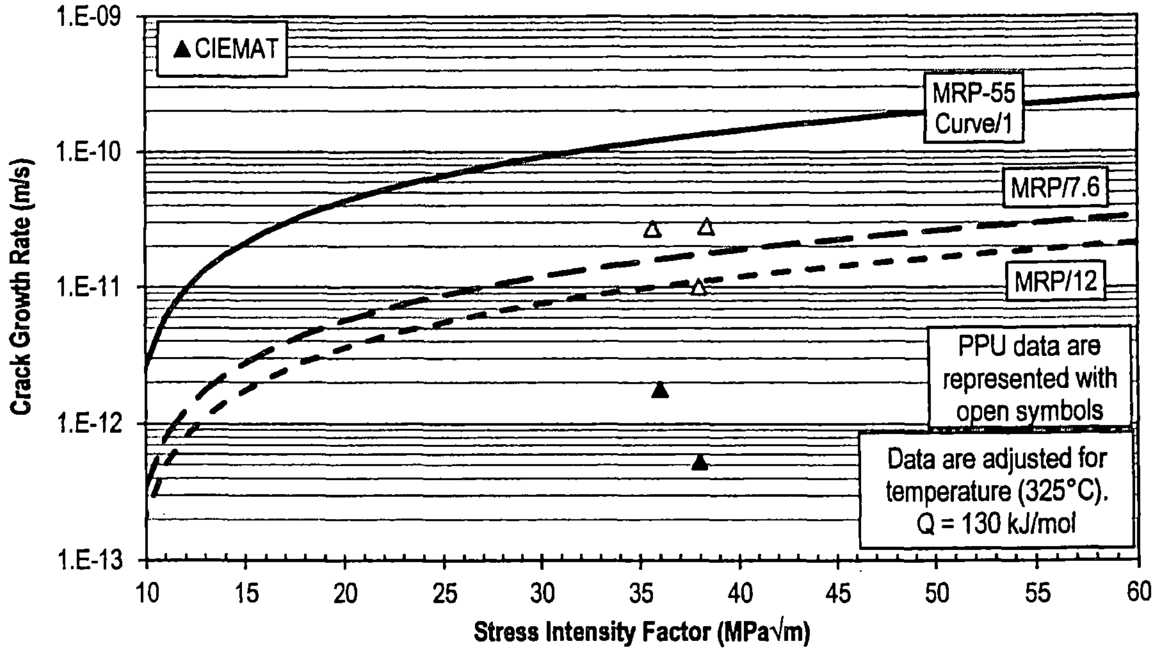

Figure 5. Plot of *da/dt* versus K_I for Alloy 690 HAZ Data from Plate Material Tested by CIEMAT

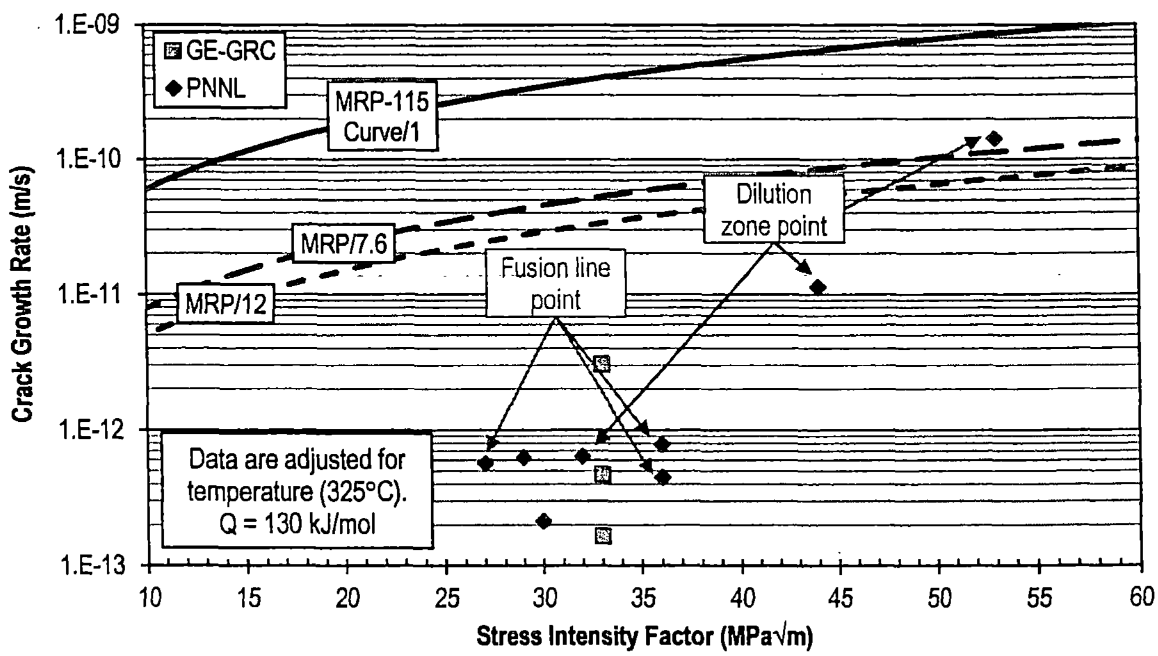

Figure 6. Plot of *da/dt* versus K_I for Alloy 152 Data from Heat WC83F8

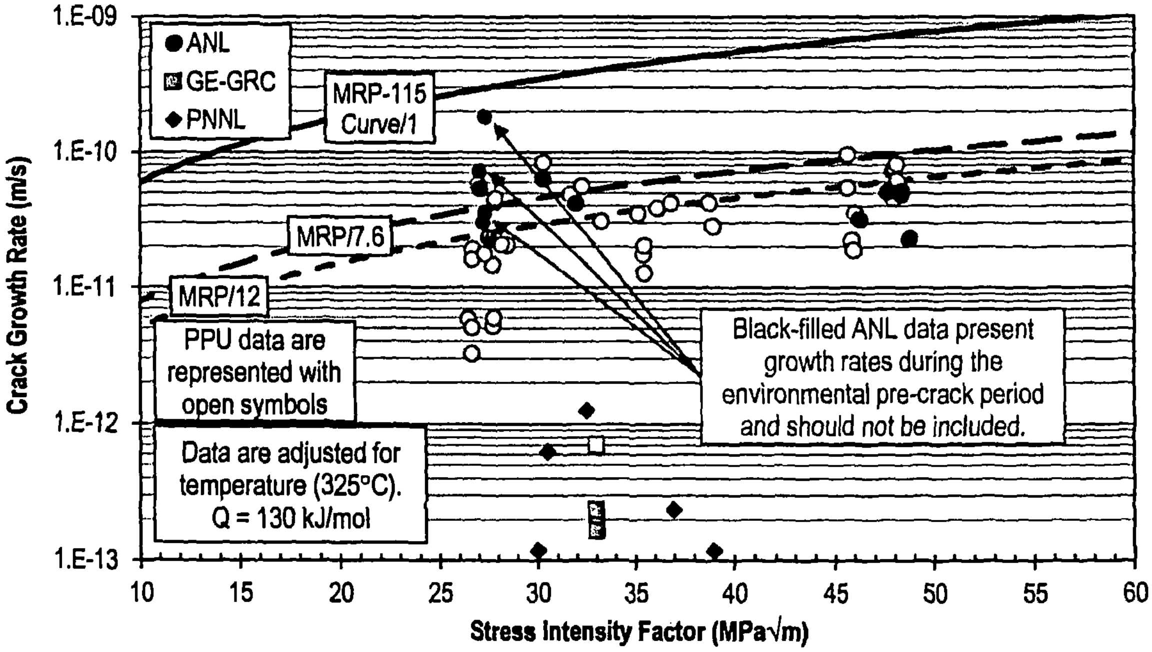

Figure 7. **Plot of *da/dt* versus K_I for Alloy 152 Data from Heat WC04F6**

Data Most Applicable to Plant Conditions

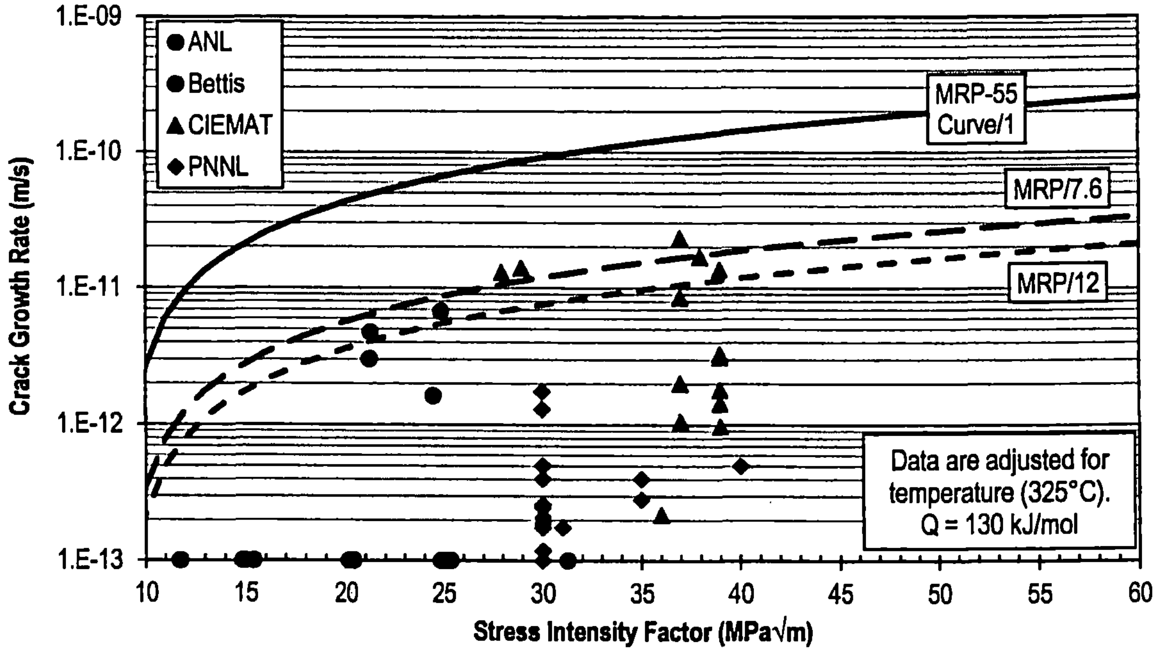

Figure 8. Plot of *da/dt* versus K_I for Alloy 690 Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I

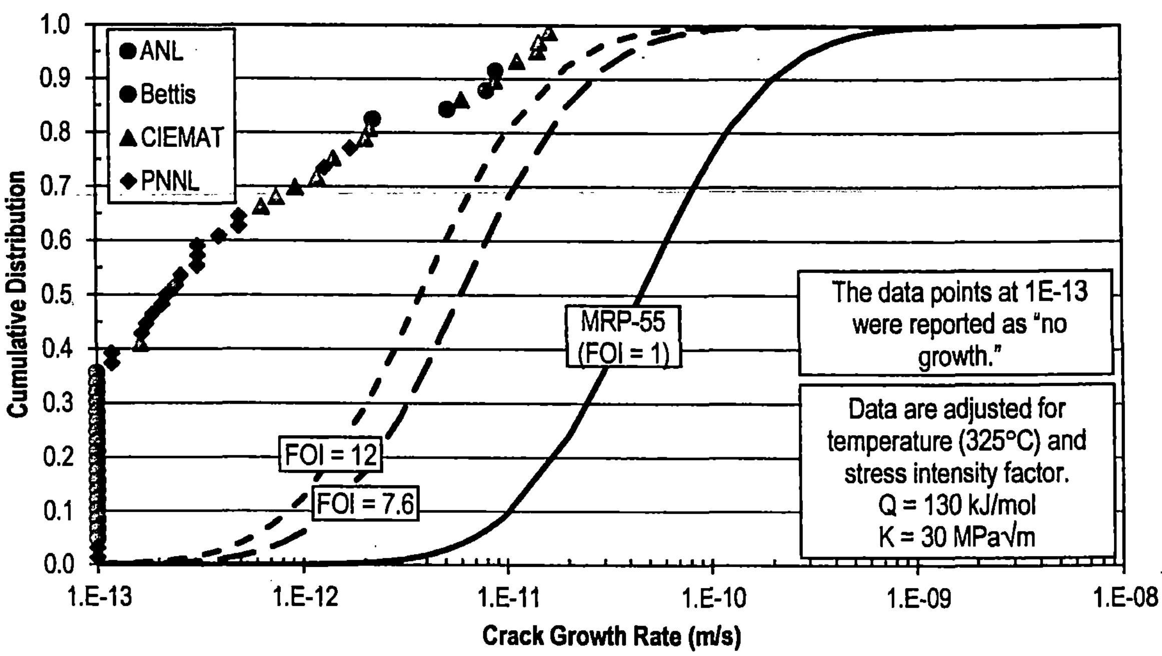

Figure 9. Cumulative Distribution Function of Adjusted *da/dt* for Alloy 690 Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I

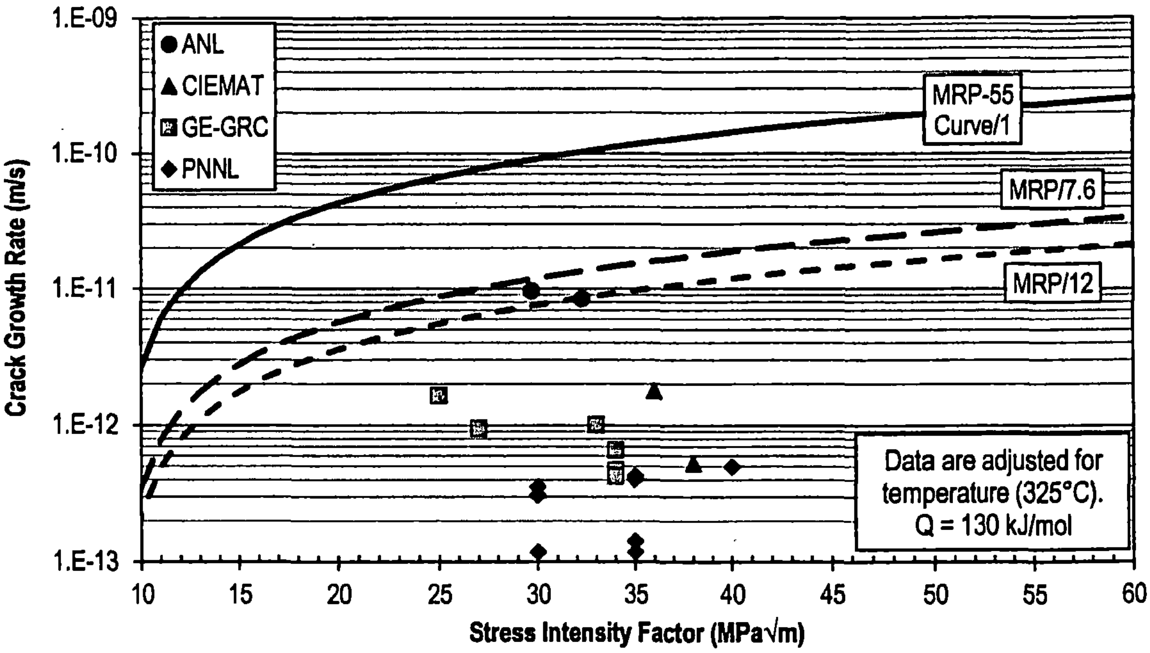

Figure 10. **Plot of *da/dt* versus K_I for Alloy 690 HAZ Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I**

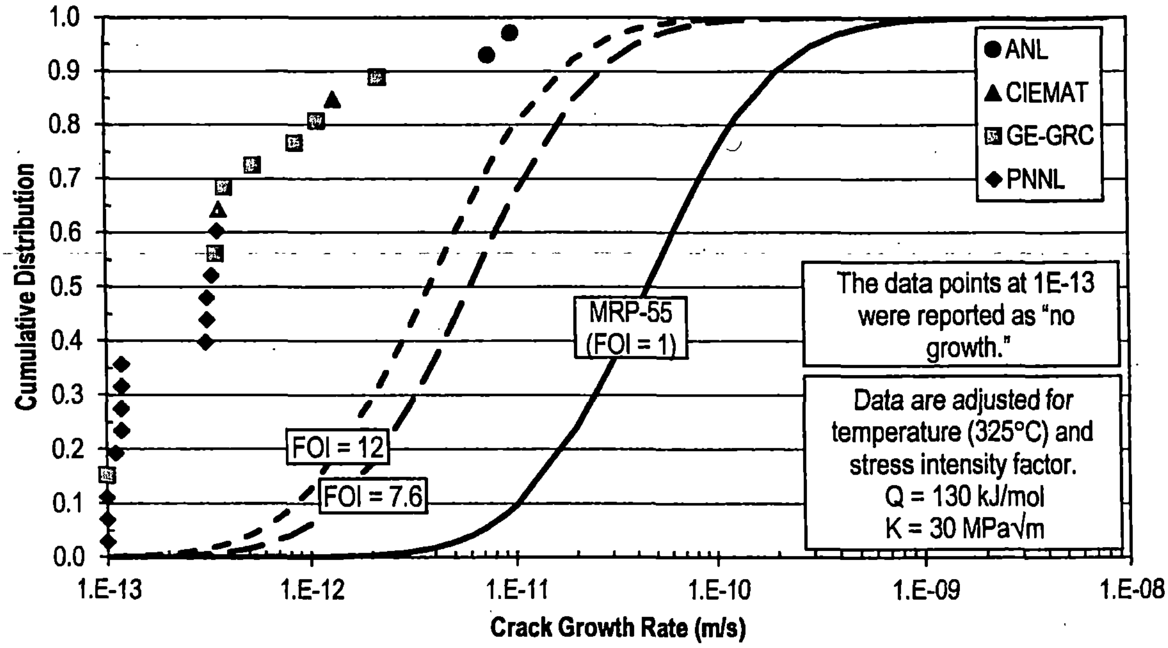

Figure 11. **Cumulative Distribution Function of Adjusted *da/dt* for Alloy 690 HAZ Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I**

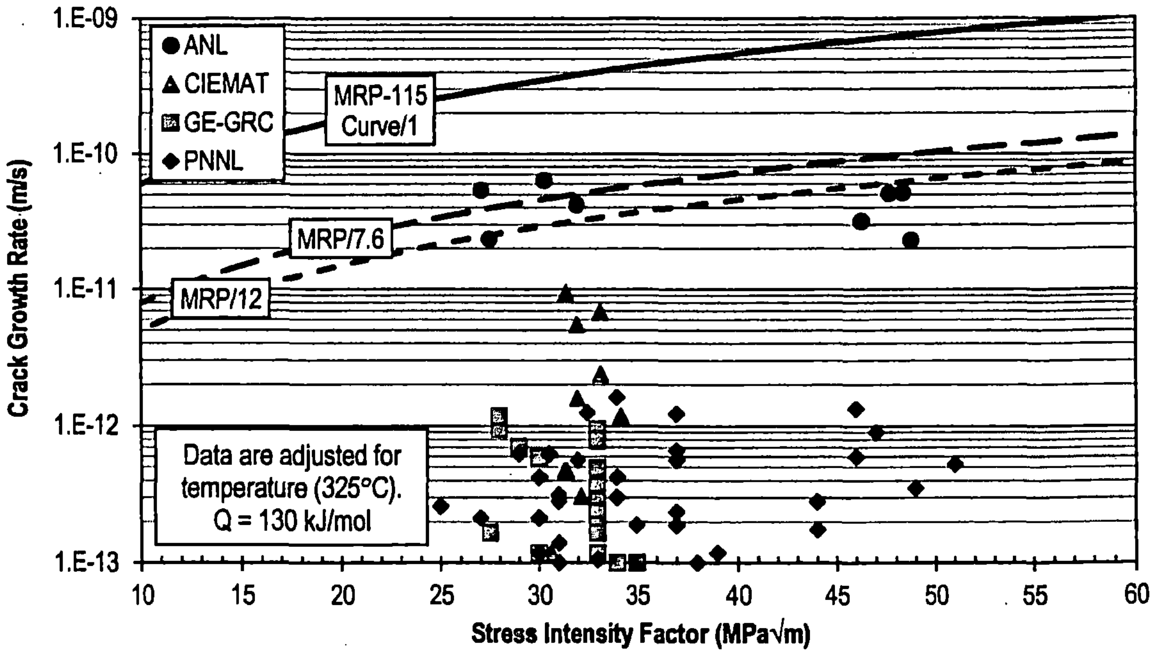

Figure 12. **Plot of *da/dt* versus K_I for Alloy 52/152 Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I**

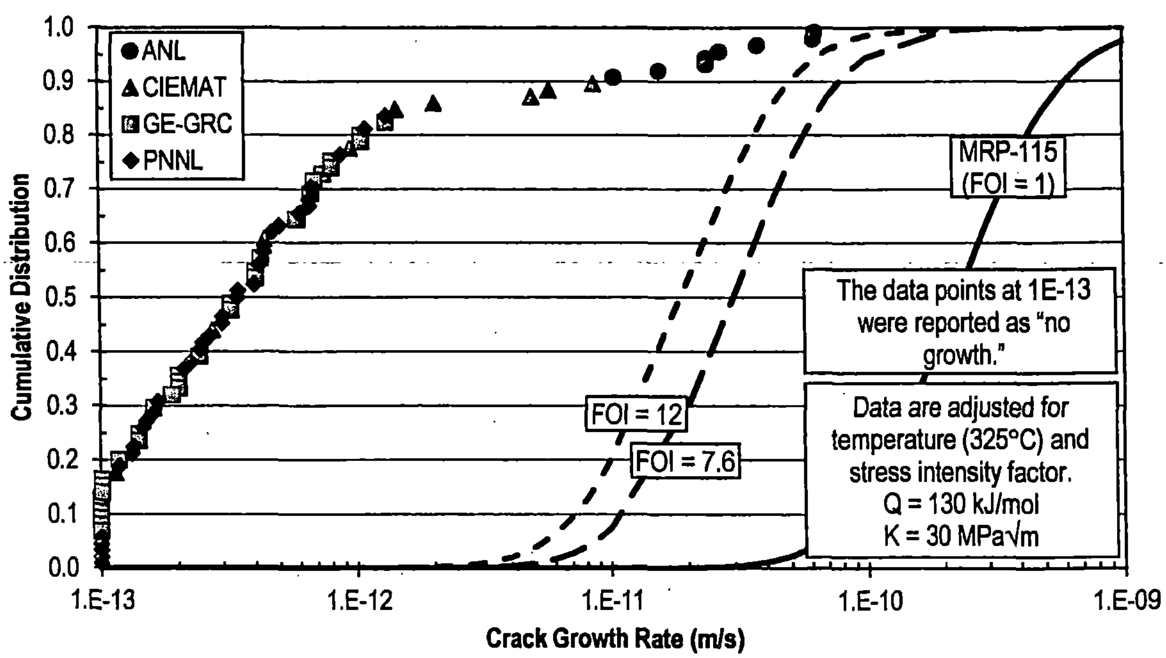

Figure 13. **Cumulative Distribution Function of Adjusted *da/dt* for Alloy 52/152 Data from All Laboratories, ≤ 10% Cold Work, Constant Load or K_I**

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ATTACHMENT 3

Calculation No. C-3688-00-01, "Minimum Factor of Improvement Implied by Extension of CCNPP RVCH UT Inspection Interval," Revision 0, December 1, 2014.

RECORD OF REVISIONS

Rev.	Description	Prepared by Date	Checked by Date	Reviewed by Date	Approved by Date
0	Original Issue	K.J. Fuhr 12/1/2014 K. J. Fuhr Associate Engineer	G.A. White 12/1/2014 G. A. White Principal Engineer	G.A. White 12/1/2014 G. A. White Principal Engineer	G.A. White 12/1/2014 G. A. White Principal Engineer

The last revision number to reflect any changes for each section of the calculation is shown in the Table of Contents. The last revision numbers to reflect any changes for tables and figures are shown in the List of Tables and the List of Figures. Changes made in the latest revision, except for Rev. 0 and revisions which change the calculation in its entirety, are indicated by a double line in the right hand margin as shown here.

TABLE OF CONTENTS

1 PURPOSE

This document presents a simple calculation of the minimum factor of improvement (FOI) on crack growth time that is implied by an extension of the volumetric or surface inspection interval for the Alloy 690 nozzles of the replacement reactor vessel closure head (RVCH) at Calvert Cliffs Nuclear Power Plant Units 1 and 2, CCNPP. ASME Code Case N-729-1 [1], which has been mandated by 10 CFR 50.55a(g)(6)(ii)(D) with conditions, specifies a reexamination interval of no more than one Code inspection interval (nominally 10 calendar years). For the purpose of this calculation, an extension of the 10-year interval to 20 years is assumed.

2 SUMMARY OF RESULTS

This simple analysis shows that the minimum FOI on crack growth time implied by a 20-year reexamination interval for CCNPP is 7.6.

3 INPUT REQUIREMENTS

1. For heads with Alloy 600 nozzles, Code Case N 729-1 [1] as conditioned by 10 CFR 50.55a limits the interval between subsequent volumetric or surface inspections to RIY = 2.25.
2. The Calvert Cliffs Unit 1 and Unit 2 replacement heads are manufactured with Alloy 690TT nozzle material and Alloy 52 weld material [2].
3. The Calvert Cliffs Unit 1 and Unit 2 replacement heads have an operating temperature of 594°F ([3] and [4]).

4 ASSUMPTIONS

1. A 10 calendar year extension beyond the 10 calendar year examination requirement of Code Case N-729-1 Item B4.40 is assumed [5].
2. It is conservatively assumed that one effective full power year (EFPY) is accumulated for each calendar year during the reexamination interval (i.e., operation at the assumed head operating temperature for 100% of the calendar time).
3. The same standard activation energy applied to model growth of Alloys 600/82/182 (31 kcal/mol or 130 kJ/mol [1]) is applied to model the temperature sensitivity of growth of a hypothetical PWSCC flaw in the Alloy 690/52/152 material of the replacement RVCH. This assumption is supported by available laboratory crack growth rate testing data for Alloy 690 wrought material:

- Results from ANL indicate that Alloy 690 with 0-26% cold work has an activation energy between 100 and 165 kJ/mol (24-39 kcal/mol) [6]. NUREG/CR-7137 [6] concludes that the activation energy for Alloy 690 is comparable to the standard value for Alloy 600 (130 kJ/mol).
- Testing at PNNL found an activation energy of about 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work [7].
- Additional PNNL testing determined an activation energy of 123 kJ/mol (29.4 kcal/mol) for Alloy 690 with 31% cold work [8].

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.

5 ANALYSIS

ASME Code Case N-729-1 [1] addresses the effect of differences in operating temperature on the required volumetric or 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. The RIY parameter, which is referenced to a head temperature of 600°F, limits the time available for potential crack growth between inspections. Per Input #1, for heads with Alloy 600 nozzles, Code Case N 729-1 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) limits the interval between subsequent volumetric or surface inspections to RIY = 2.25. This RIY value is based on the conclusion of the technical basis documents ([9], [10], and [11]) that a reexamination interval between volumetric or 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. 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 ([12] and [13]).

5.1 RIY Parameter Describing the Potential for Crack Propagation

The RIY parameter, which quantifies the potential for crack propagation between successive volumetric or surface examinations, is defined by ASME Code Case N-729-1 [1] as follows:

$$RIY = \sum_{j=n1}^{n2} \left\{ \Delta EFPY_j \exp\left[-\frac{Q_g}{R} \left(\frac{1}{T_{head,j}} - \frac{1}{T_{ref}} \right) \right] \right\} \quad [5\text{-}1]$$

where:

RIY	=	Reinspection Years, normalized to a reference temperature of 600°F
$\Delta EFPY_j$	=	effective full power years accumulated during time period j
Q_g	=	activation energy for crack growth (31 kcal/mole)
R	=	universal gas constant (1.103×10^{-3} kcal/mol-°R)
$T_{head,j}$	=	absolute 100% power head temperature during time period j (°R = °F + 459.67)
T_{ref}	=	absolute reference temperature (1059.67°R)
$n1$	=	number of the first time period with distinct 100% power head temperature[1] since time of most recent volumetric or surface NDE
$n2$	=	number of the most recent time period with distinct 100% power head temperature

The RIY expression simplifies to the following assuming a single representative head temperature over the period between successive examinations:

$$RIY = \Delta EFPY \exp\left[-\frac{Q_g}{R} \left(\frac{1}{T_{head}} - \frac{1}{T_{ref}} \right) \right] \quad [5\text{-}2]$$

Conservatively assuming that the EFPYs of operation accumulated at each Calvert Cliffs unit since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended period at Calvert Cliffs is calculated as follows:

$$RIY = (20.0)\exp\left[-\frac{31}{1.103\times10^{-3}} \left(\frac{1}{594+459.67} - \frac{1}{600+459.67} \right) \right] = (20.0)(0.860) = 17.20 \quad [5\text{-}3]$$

[1] Head temperature at 100% power may have been changed during the life of the plant due to design changes, power uprates, etc., and the summation is over the number of distinct periods since the last volumetric or surface NDE.

5.2 Factor of Improvement (FOI) Implied by RIY

The FOI implied by this RIY value for Calvert Cliffs (relative to the limiting RIY for heads with Alloy 600 nozzles) is calculated as the following ratio:

$$FOI = \frac{RIY_{Alloys\ 690/52/152}}{RIY_{Alloys\ 600/82/182}} = \frac{(20.0)(0.860)}{2.25} = \frac{17.20}{2.25} = 7.6 \qquad [5\text{-}4]$$

This FOI value may be compared to laboratory PWSCC crack growth rate data for Alloys 690/52/152 when they are considered relative to standard statistical distributions describing the variability in the crack growth rate for Alloy 600 [14] and Alloy 182 [15].

Note that the temperature factor calculated above (0.860) is relatively modest such that the FOI result is relatively insensitive to the assumed activation energy. For an activation energy of 40 kcal/mole, the calculated FOI would be 7.3 instead of 7.6.

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