FPL.

MAR 0 7 2016

L-2016-052
10 CFR 50.4
10 CFR 50.55a

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

Re: St. Lucie Unit 2
Docket No. 50-389
In-Service Inspection Plans
Fourth Ten-Year Interval
Unit 2 Relief Request 11

Pursuant to 10CFR50.55a(z)(1), Florida Power & Light (FPL) 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 St Lucie Unit 2 RVCH penetrations. Electric Power Research Institute (EPRI) 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 re-examination interval can be extended while maintaining an acceptable level of quality and safety.

The justification for this relief is contained in Attachment 1, Attachment 2 provides a calculation summary for the minimum factor of improvement (FOI) on crack growth, and Attachment 3 provides further support for the requested alternative inspection interval based PWSCC crack growth rate data, the FOI approach, and addresses requests for additional information that the NRC has transmitted to other licensees in the context of similar relief requests.

Please contact Ken Frehafer at 772-467-7748 if there are any questions about this submittal.

Very truly yours,

SCISCANTE FOR ESK

Eric Katzman
Licensing Manager
St. Lucie Plant

Attachment

ESK/KWF

cc: USNRC Regional Administrator, Region II
USNRC Senior Resident Inspector, St. Lucie Units 1 and 2

A047
NRR

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER11, Rev. 0

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

1. ASME CODE COMPONENT(S) AFFECTED:

The affected components are ASME Class 1 Pressurized Water Reactor (PWR) Reactor Vessel Upper Head (Closure Head) (RVCH) nozzles and partial-penetration welds fabricated with primary water stress corrosion cracking (PWSCC)-resistant materials. St Lucie Unit 2 penetration nozzles and vent pipe are fabricated from Alloy 690 with Alloy 52/152 attachment welds.

2. APPLICABLE CODE EDITION AND ADDENDA:

The Fourth Ten Year ISI interval Code of record for St Lucie Unit 2 is the 2007 Edition with 2008 Addenda of 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. APPLICABLE CODE REQUIREMENT:

The Code of Federal Regulations (CFR) 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."

10 CFR 50.55a(g)(6)(ii)(D)(*3*) conditions ASME Code Case N-729-1 [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, -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. REASON FOR REQUEST:

Code Case N-729-1 [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 [2]. Using plant and laboratory data, Electric Power Research Institute (EPRI) 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 re-examination interval can be extended to a 20 year interval length while maintaining an acceptable level of quality and safety. FPL is requesting approval of this alternative to allow the use of the ISI interval extension for the St Lucie Unit 2 Alloys 690/52/152 reactor vessel closure head penetrations.

In addition to maintaining an acceptable level of quality and safety with the proposed examination frequency, elimination of one examination occurrence will reduce personnel radiation exposure by an estimated 2 to 3 REM and reduce industrial injury risk by eliminating the examination at the 10 year period.

5. PROPOSED ALTERNATIVE AND BASIS FOR USE:

Florida Power & Light Co. (FPL) 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 St Lucie Unit 2 RVCH penetrations not to exceed one inspection interval (nominally 10 calendar years). The St. Lucie Unit 2 replacement RVCH went into service when the unit started up in January 2008. Specifically, this would allow volumetric or surface examinations currently scheduled for the Spring 2017 Outage to be extended to the Spring 2023 refueling outage (nominally, 15.5 calendar years from installation). This request applies to the Item B4.40 inspection frequencies only.

As discussed in the original ASME technical basis document [2], 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 [3], which was summarized in the safety assessment for RVCHs in MRP-110 [4]. The material improvement factor for primary water stress corrosion cracking (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 [2].

Further evaluations were performed to demonstrate the acceptability of extending the inspection intervals for Code Case N-729-1, 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 second interval (nominally 20 calendar years) is based on plant service experience, factor of improvement studies using laboratory data, deterministic study results, and probabilistic study results.

Per MRP-375, much of the laboratory data indicated a factor of improvement of 100 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 RVCHs 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.

Two of the replacement heads that were volumetrically examined in accordance with N-729-1 were Turkey Point Units 3 and 4, owned by FPL. The Turkey Point heads were replaced in 2004 and 2005 respectively, and examined during their 2014 refueling outages. The St. Lucie Unit 2 head and the Turkey Point Units 3 & 4 head were fabricated by the same manufacturer (AREVA), using thermally treated Alloy 690 nozzle material produced by the same material supplier (Valinox Nucleaire), per the same ASME SB-167 nozzle material specifications with identical supplemental requirements as the previously examined Turkey Point Units 3 and 4 heads. The nozzle J-groove attachment welds for the Turkey Point and St. Lucie Heads utilized PWSCC resistant ERNiCrFe-7 (UNS N06052 and/or ENiCrFe-7 UNS W86152) weld materials. The St. Lucie Unit 2 and Turkey Point Units 3 & 4 were all procured to ASME Section III, 1989 Edition, no addenda. 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 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 main 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 approximately 2.0 EFPYs or a 2 year operating cycle. Thus, a nominal interval of 15.5 calendar years for the St. Lucie Unit 2 replacement head implies a FOI of 7.35 (see Attachment 2 of FPL Letter L-2016-052) versus the standard interval for heads with Alloy 600 nozzles. It is emphasized that the FOI of 7.35 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 3 of FPL Letter L-2016-052, Dominion Engineering Inc. Technical Note TN-5696-00-02 Rev 0, provides further support for the requested alternative inspection interval based on the available laboratory PWSCC crack growth rate data and the FOI approach and addresses requests for additional information that NRC has transmitted to other licensees in the context of similar relief requests (see Section 7 Precedent). Attachment 3 describes the materials tested for data points that are located above the curves that are a factor of 12 below the MRP-55 [7] and MRP-115 [6] crack growth rate curves for the 75th percentile of material variability. As the attachment discusses data points above the curves that are a factor of 12 below the MRP-55 and MRP-115 curves, the discussion bounds the needed FOI of 7.35 for this St. Lucie Unit 2 inspection interval extension.

Attachment 3 of FPL Letter L-2016-052 also identifies that much of the Alloy 690 CRDM material included in the MRP-375 data compilation was supplied by the Valinox Nucleaire, the same material supplier for the St. Lucie Unit 2 head (and the previously examined Turkey Point Unit 3 & 4 heads). Therefore this crack growth rate data and the FOI approach are applicable as a basis for the St. Lucie Unit 2 requested frequency extension. 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 St. Lucie Unit 2 replacement head.

Design Features Further Increasing the Resistance of the St. Lucie Unit 2 *Replacement Head to PWSCC*

In addition to the standard Alloy 690 materials (plate and CRDM nozzle material) test data reported in MRP-375, FPL imposed supplemental requirements on the St. Lucie Unit 2 nozzle materials (identical to and the previously examined Turkey Point Unit 3 & 4 heads) to increase the material resistance to PWSCC. These supplemental requirements included; thermal treatment (TT), prohibition of cold straightening after TT, ingot remelting to reduce impurities, additional chemistry requirements, microstructure and grain size requirements. These methods substantially reduce PWSCC susceptibility beyond that assumed in the generic MRP-375 study, resulting in additional assurance that the St. Lucie Unit 2 head can be operated for 15.5 years from replacement prior to their next volumetric and/or surface examination with an acceptable level of quality and safety.

Previous Examinations of the St. Lucie Unit 2 Replacement Head

A preservice volumetric examination of the replacement RVCH partial-penetration welded nozzles was performed prior to head installation that went into service with the start up in January 2008 at St Lucie 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.

A bare metal visual examination (VE) was performed of the St Lucie Unit 2 replacement RVCH in 2012 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.

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 degrees Fahrenheit (°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 Effective Full Power Years (EFPY). 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 EFPY. 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 upper head operating temperature of the St Lucie Unit 2 is 602.6°F and the FOI for the extension to a 15.5 year examination frequency is 7.35 and is 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 [8] 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 [9] technical basis for Alloy 600 heads,

2) And 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 second interval (nominally 20 calendar years) is based on plant service experience, factor of improvement studies using laboratory initiation and growth data, deterministic modeling, and probabilistic study results. The results of the analysis show that the alternative proposed extension to a 15.5 year examination 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 of 7.35 implied by the requested extension to a 15.5 year examination frequency 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 St Lucie Unit 2 replacement head such as thermal treatment of nozzle material and by the 2014 indication free inspection results of the Turkey Point Unit 3 and 4 heads with similar material. 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.

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

6. DURATION OF PROPOSED ALTERNATIVE:

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

7. PRECEDENTS:

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 this request are shown below:

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER11, Rev. 0

Plant	NRC ADAMS Accession No.				Status
	Relief Request	Request for Additional Information (RAI)	RAI Response	NRC Safety Evaluation	
Arkansas Nuclear One, Unit 1	ML14118A477	ML14258A020	ML14275A460	ML14330A207	Accepted
Beaver Valley, Unit 1	ML14290A140			ML14363A409	Accepted
Calvert Cliffs Unit 1 & 2	ML15009A035 ML15201A067			ML15327A367	Accepted
Comanche Peak Unit 1	ML15120A038			ML15259A004	Accepted
D.C. Cook Units 1 & 2	ML15023A038			ML15156A906	Accepted
J.M. Farley, Unit 2	ML14280A260 ML15111A387			ML15104A192	Accepted
North Anna, Unit 2	ML14283A044			ML15091A687	Accepted
Prairie Island, Units 1 and 2	ML14258A124	ML15030A008	ML15036A252	ML15125A361	Accepted
H.B. Robinson, Unit 2	ML14251A014	ML14294A587	ML14325A693	ML15021A354	Accepted
Salem Unit 1	ML15098A426			ML15349A956	Accepted
St. Lucie, Unit 1	ML14206A939	ML14251A222	ML14273A011	ML14339A163	Accepted

St. Lucie Unit 2
Fourth Inspection Interval
RELIEF REQUEST NUMBER11, Rev. 0

---

8. 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. ASME Section XI, Code Case N-729, "Technical Basis Document," dated September 14, 2004.
3. *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]
4. *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]
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. *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]
9. *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]

Calculation of the Minimum Factor of Improvement Needed by Extension of Reactor Vessel Closure Head Volumetric/Surface Inspection Interval to Support Relief Requests Citing MRP-375

1 Introduction

This document presents a calculation of the minimum factor of improvement (FOI) on crack growth time that is implied by an extension of the volumetric/surface inspection interval for the Alloy 690 nozzles of a replacement reactor vessel closure head (RVCH) for St. Lucie Unit 2. This is the minimum FOI value that crack growth rate testing should demonstrate in order to directly support the requested inspection interval. ASME Code Case N-729-1 [1], which has been mandated by 10CFR50.55 a(g)(6)(ii)(D) with conditions, specifies a reexamination interval of no more than one Section XI inspection interval (nominally 10 calendar years). MRP-375 [2] supports an extension of this interval to two Section XI inspection intervals.

2 Summary of Results

This calculation is performed for St. Lucie Unit 2, which has a replacement head manufactured with Alloy 690/52/152 material, has a head operating temperature of 602.6°F, and requests a five and a half (5.5) calendar year examination interval extension beyond the 10 calendar year examination requirement of Code Case N-729-1. This analysis shows that the minimum FOI on crack growth time implied by a 15.5-year reexamination interval for St. Lucie Unit 2 (a 5.5-year extension relative to Code Case N-729-1) is 7.35.

3 Analysis

ASME Code Case N-729-1 [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, 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 MRP-117 [3] technical basis document for heads with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate reexamination interval to detect any PWSCC in a timely fashion.

The RIY parameter for heads with Alloy 600 nozzles is adjusted to the reference head temperature using an activation energy of 130 kJ/mole (31 kcal/mole) [1]. Based on the available laboratory data, the same activation energy is applicable to model the temperature sensitivity of growth of a hypothetical PWSCC flaw in the Alloy 690/52/152 material of the replacement RVCH. Key laboratory crack growth rate testing data for Alloy 690 wrought material investigating the effect of temperature are as follows:

(1) 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) [4]. NUREG/CR-7137 [4] concludes that the activation energy for Alloy 690 is comparable to the standard value for Alloy 600 (130 kJ/mole).
(2) Testing at PNNL found an activation energy of about 120 kJ/mol (28.7 kcal/mole) for Alloy 690 materials with 17-31% cold work [5].
(3) Additional PNNL testing determined an activation energy of 123 kJ/mole (29.4 kcal/mole) for Alloy 690 with 31% cold work [6].

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.

3.1 RIY Parameter Describing the Potential for Crack Propagation

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

$$RIY = \sum_{j=n1}^{n2}\left\{\Delta EFPY_j \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T_{head,j}}-\frac{1}{T_{ref}}\right)\right]\right\} \qquad [3\text{-}1]$$

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/surface NDE

[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/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:

$$RIY = \Delta EFPY \exp\left[-\frac{Q_g}{R}\left(\frac{1}{T_{head}} - \frac{1}{T_{ref}}\right)\right] \quad [3\text{-}2]$$

Conservatively assuming that the EFPYs of operation accumulated at St. Lucie Unit 2 since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended period at St. Lucie Unit 2 is calculated as follows:

$$RIY = (15.5\ \text{EFPY})\exp\left[-\frac{31}{1.103\times10^{-3}}\left(\frac{1}{602.6+459.67} - \frac{1}{600+459.67}\right)\right] = (15.5)(1.07) = 16.54 \quad [3\text{-}3]$$

3.2 Factor of Improvement (FOI) Implied by RIY

The FOI implied by this RIY value for St. Lucie Unit 2 (relative to the limiting RIY for heads with Alloy 600 nozzles) is calculated as the following ratio:

$$FOI = \left[\frac{RIY_{Alloys\ 690,52,152}}{RIY_{Alloys\ 600,82,182}}\right] = \frac{(15.5)(1.07)}{2.25} = \frac{16.54}{2.25} = 7.35 \quad [3\text{-}4]$$

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 [7] and Alloy 182 [8]. Alloy 600 wrought material is the appropriate reference for defining the FOI for Alloy 690 wrought material. As discussed in Section 3.1 of MRP-375, Alloy 182 weld metal is chosen as the reference for defining the FOI for Alloys 52 and 152 weld metals because Alloy 182 is more susceptible on average to PWSCC initiation and growth than Alloy 82 (due to the higher Cr content of Alloy 82).

Note that the temperature factor calculated above (1.07) 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 7.49 instead of 7.35.

4 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: 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 downloadable at www.epri.com]
3. *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 downloadable at www.epri.com]
4. 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. [NRC ADAMS Accession No. ML12199A415]
5. *Materials Reliability Program: Resistance of Alloys 690, 152, and 52 to Primary Water Stress Corrosion Cracking (MRP-237, Rev.2): Summary of Findings Between 2008 and 2012 from Completed and Ongoing Test Programs*, EPRI, Palo Alto, CA: 2013. 3002000190. [freely downloadable at www.epri.com]
6. 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.
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 downloadable at www.epri.com]
8. *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 downloadable at www.epri.com]