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10 CFR 50.55a

Docket Nos.: 50-315
50-316

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

Donald C. Cook Nuclear Plant, Units 1 and 2
REQUEST FOR ALTERNATIVE FROM VOLUMETRIC/SURFACE EXAMINATION FREQUENCY REQUIREMENTS OF ASME CODE CASE N-729-4, REQUEST NUMBER ISIR 04-05, REVISION 1

References:

1. Letter from J. P. Gebbie, Indiana Michigan Power (I&M), to U. S. Nuclear Regulatory Commission (NRC), "Donald C. Cook Nuclear Plant, Units 1 and 2, Request for Alternative from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1, Request Number ISIR 04-02," dated January 20, 2015, [Agencywide Documents Access and Management System (ADAMS) Accession Number ML15023A038].

2. Letter from D. L. Pelton, NRC, to L. J. Weber, I&M, "Donald C. Cook Nuclear Plant, Units 1 and 2 – Request for Use of Alternative ISIR 04-02 Associated with Reactor Vessel Closure Head Volumetric/Surface Examination Frequency Requirements for the Inservice Inspection Program (TAC Nos. MF5606 and MF5607)," dated June 11, 2015, ADAMS Accession Number ML15156A906.

Pursuant to 10 CFR 50.55a(z), Indiana Michigan Power Company, the licensee for Donald C. Cook Nuclear Plant (CNP) Units 1 and 2, hereby requests approval of a proposed alternative for the CNP Units 1 and 2 Inservice Inspection program. This request is associated with the examination frequency requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, Code Case N-729-4, which specifies that reactor vessel welded Alloy 690 head penetration nozzles shall undergo volumetric/surface examinations on a frequency of all nozzles in a nominal ten-year inspection interval.

The proposed alternative, included as the enclosure to this letter, would allow deferral of the volumetric/surface examinations of the replacement reactor pressure vessel closure heads of both Units 1 and 2 for two fuel cycles beyond the nominal ten-year inspection interval required by Code Case N-729-4. The alternative inspection interval provides an acceptable level of quality and safety. The proposed alternative was previously submitted by Reference 1 and was previously approved by Reference 2. This request is to acknowledge a change to the applicable revision of the Code Case from N-729-1 to N-729-4. Additionally, the request number has been updated to accurately reflect the correct number of ISIR 04-05, Revision 1. During the previous submittal the

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request number was incorrectly documented as ISIR 04-02; this was entered into CNP's corrective action program and is being corrected with this submittal.

Approval of the proposed relief request is requested prior to entry into Mode 5 of the current Unit 2 refueling outage, when the reactor vessel head is reinstalled. Mode 5 is currently scheduled to occur on April 27, 2018. Copies of this letter are being transmitted to the Michigan Public Service Commission and Michigan Department of Environmental Quality, in accordance with the requirements of 10 CFR 50.91.

There are no new regulatory commitments made in this letter. Should you have any questions, please contact Mr. Michael K. Scarpello, Regulatory Affairs Manager, at (269) 466-2649.

Sincerely,

Quinton S. Lies

Q. Shane Lies
Site Vice President

DB/kmh

Enclosure: 10 CFR 50.55a Request Number ISIR 04-05, Revision 1, Proposed Alternative in Accordance with 10 CFR 50.55a(z).

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10 CFR 50.55a Request Number ISIR 04-05, Revision 1

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

--Alternative Provides Acceptable Level of Quality and Safety--

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

The affected components are the Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2 ASME Class 1 reactor pressure vessel closure head (RVCH) nozzles and partial-penetration welds fabricated from Primary Water Stress Corrosion Cracking (PWSCC) - resistant materials. Each unit's RVCH nozzle penetration tubes, vent pipe, and reactor vessel level indication system (RVLIS) pipe are fabricated from Alloy 690 material with Alloy 52/152 attachment welds.

2. Applicable Code Edition and Addenda

The applicable Code edition for the CNP fourth Inservice Inspection (ISI) interval that began on March 1, 2010, is ASME Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2004 Edition, with no Addenda.

3. Applicable Code Requirement

10 CFR 50.55a(g)(6)(ii)(D)(*1*), requires (in part) that:

> "Holders of operating licenses or combined licenses for pressurized-water reactors as of or after August 17, 2017 shall implement the requirements of ASME BPV Code Case N-729-4 instead of ASME BPV Code Case N-729-1, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of this section, by the first refueling outage starting after August 17, 2017."

10 CFR 50.55a(g)(6)(ii)(D)(*3*) conditions ASME Code Case N-729-4 [Reference 4] by stating:

> "Instead of Note 4 of ASME BPV Code Case N-729-4, the following shall be implemented. If effective degradation years (EDY) <8 and if no flaws are found that are attributed to primary water stress corrosion cracking:
>
> (i) A bare metal visual examination is not required during refueling outages when a volumetric or surface examination is performed; and
>
> (ii) If a wetted surface examination has been performed of all of the partial penetration welds during the previous non-visual examination, the reexamination frequency may be extended to every third refueling outage or 5 calendar years, whichever is less, provided an IWA-2212 VT-2 visual examination of the head is performed under the insulation through multiple

access points in outages that the VE is not completed. This IWA-2212 VT-2 visual examination may be performed with the reactor vessel depressurized."

ASME Code Case N-729-4 specifies that the reactor vessel upper head components shall be examined on a frequency in accordance with Table 1 of this code case. The basic inspection requirements of Code Case N-729-4 for partial-penetration welded Alloy 690 head penetration nozzles are as follows:

- Volumetric/surface examination of all nozzles every ASME Section XI ten-year ISI interval (provided that flaws attributed to PWSCC have not previously been identified in the head) (Item B4.40);
- Direct visual examination of the outer surface of the head for evidence of leakage every third refueling outage or 5 calendar years, whichever is less (Item B4.30).

4. Reason for Request

Treatment of Alloy 690 RVCHs in Code Case N-729-4 was intended to be conservative and subject to reassessment once additional laboratory data and plant experience on the performance of Alloy 690 and Alloys 52/152 weld metals became available [Reference 2 and 3]. Using plant and laboratory data, Electric Power Research Institute (EPRI) document Materials Reliability Program (MRP)-375 [Reference 3] was developed to support a technically based volumetric/surface reexamination interval using appropriate analytical tools. This technical basis demonstrates that the reexamination interval can be extended to the interval length requested below while maintaining an acceptable level of quality and safety. Therefore, Indiana Michigan Power Company (I&M) is requesting approval of this alternative to allow the use of the ISI interval extension for the affected CNP Unit 1 and Unit 2 components.

5. Proposed Alternative and Basis for Use

Proposed Alternative

Pursuant to 10 CFR 50.55a(z)(1), I&M requests an alternative from performing the required volumetric/surface examinations for the CNP Unit 1 and Unit 2 RVCH components identified above at the frequency prescribed in ASME Code, Section XI, Code Case N-729-4. Specifically, I&M requests to extend the frequency of the volumetric/surface examination of the CNP Unit 1 and Unit 2 RVCHs of Table 1, Item B4.40 of ASME Code Case N-729-4 for two fuel cycles beyond the one inspection interval (nominally ten calendar years) from installation of the replacement RVCHs.

- For CNP Unit 1, this request would extend the volumetric/surface examination to the Cycle 29 refueling outage that is scheduled for spring 2019. At that point, the Unit 1 RVCH will have been in service for approximately 12.4 calendar years.
- For CNP Unit 2, this request would extend the volumetric/surface examination to the Cycle 25 refueling outage that is scheduled for fall 2019. At that point, the Unit 2 RVCH will have been in service for approximately 11.9 calendar years.

No alternative examination processes are proposed to those required by ASME Code Case N-729-4, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D). The visual examinations and acceptance criteria as required by Item B4.30 of Table 1 of ASME Code Case N-729-4 are not affected by this request and will continue to be performed on a frequency of every third refueling outage or five calendar years, whichever is less.

Basis for Use

The original CNP Unit 1 and Unit 2 RVCHs, which were manufactured with Alloy 600/82/182 materials, were replaced with new RVCHs using Alloy 690/52/152 materials during the refueling outages that returned to operation in November 2006 and November 2007, respectively. In accordance with Table 1 of ASME Code Case N-729-1, Item B4.40, as conditioned by 10 CFR 50.55a(g)(6)(ii)(D)(3), I&M was previously required to perform a volumetric and/or surface examination of essentially 100 percent (%) of the required volume or equivalent surfaces of the nozzle tubes as follows:

- For CNP Unit 1, during the Cycle 27 refueling outage that was scheduled for spring 2016 (9.4 calendar years following replacement). This examination was not performed as approved by Reference 29.

- For CNP Unit 2, during the Cycle 23 refueling outage that was scheduled for fall 2016 (8.9 calendar years following replacement). This examination was not performed as approved by Reference 29.

The basis for the inspection frequency for ASME Code Case N-729-4 comes, in part, from the analysis of laboratory and plant data presented in report MRP-111 [Reference 4], which was summarized in the safety assessment for RVCHs in MRP-110 [Reference 5]. The material improvement factor for PWSCC of Alloy 690/52/152 materials over that of mill-annealed Alloys 600 and 182 was shown by this report to be on the order of 26 or greater.

Further evaluations were performed to demonstrate the resistance of Alloys 690/52/152 to PWSCC under an EPRI MRP initiative provided in MRP-375 [Reference 3]. This report combines an assessment of the test data and operating experience developed since the technical basis for the 10-year interval of Case N-729-1 was developed in 2004 [Reference 2] with deterministic and probabilistic evaluations to assess the improved PWSCC resistance of Alloys 690/52/152 relative to Alloys 600/82/182.

Evaluation of Alloys 690/52/152 Data and Experience by MRP-375

Operating experience to date for replacement and repaired components using Alloys 690/52/152 has shown a proven record of resistance to PWSCC during numerous examinations in the approximate 25 years of its application. This experience includes steam generators, pressurizers, and RVCHs. In particular, at the completion of the spring 2017 refueling outage season, Alloy 690/52/152 operating experience includes inservice volumetric/surface examinations performed in accordance with ASME Code Case N-729-1 on 13 of the 40 replacement RVCHs currently operating in the United States (U.S.). Some of

these examined heads had continuous full power operating temperatures that may approach 613 degrees Fahrenheit (°F). None of these examinations revealed PWSCC cracking, and these examination results further support the low likelihood of the potential for the RVCHs to experience PWSCC during the extension periods.

The evaluation performed in MRP-375 considers a simple Factor of Improvement (FOI) approach applied in a conservative manner to model the increased resistance of Alloys 690/52/152 compared to Alloys 600 and 182 at equivalent temperature and stress conditions.[1] FOIs were estimated for the material improvements of Alloy 690/52/152 materials using an extensive database of test data. Results for both crack initiation and crack growth conclude a substantially improved resistance to PWSCC for Alloy 690 base material and Alloy 52/152 weld materials. Figures 3-2, 3-4, and 3-6 of MRP-375 provide crack growth rate data for Alloy 690/52/152 materials and heat affected zones with curves plotting FOIs of 1, 5, 10, and 20 on a statistical basis reflecting the material variability exhibited in MRP-55 [Reference 6] for Alloy 600 material and in MRP-115 [Reference 7] for Alloy 82/182/132 weld material.[2] An FOI of 20 bounds most of the data plotted, and an FOI of 10 essentially bounds all of the crack growth rate data. Table 3-6 of MRP-375 provides a summary of FOIs determined on the basis of crack growth rate and crack initiation data. For crack initiation, FOIs reported, although significant, are conservatively small because crack initiation of Alloys 690/52/152 was not observed during testing; instead, the initiation time was assumed to be equivalent to the test duration.

Additional Evaluations Performed under MRP-375

MRP-375 applied the FOI results to perform a combination of deterministic and probabilistic evaluations to establish an appropriately conservative inspection interval for Alloy 690 RVCHs. The deterministic technical basis applies industry-standard crack growth calculation procedures to predict time to certain adverse conditions under various conservative assumptions. A probabilistic evaluation is then applied to make predictions for leakage and ejection risk, generally using best-estimate inputs and assumptions, with uncertainties treated using statistical distributions.

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[1] 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 [Reference 3], Alloy 182 weld metal is chosen as the reference for defining the FOI for Alloys 52 and 152 weld metals because Alloy 182 is generally more susceptible to PWSCC initiation and growth than Alloy 82 (due to the higher chromium content of Alloy 82).

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

The deterministic crack growth evaluation provides a precursor to the probabilistic evaluation to directly illustrate the relationship between the improved PWSCC growth resistance of Alloys 690/52/152 and the time to certain adverse conditions. These evaluations apply conservative crack growth rate predictions and the assumption of an existing flaw (which is replaced with a PWSCC initiation model for probabilistic evaluation). The evaluations provide a reasonable lower bound on the time to adverse conditions, from which a conservative inspection interval may be recommended. This evaluation draws from various EPRI MRP and industry documents that evaluate, for Alloys 600/82/182, the time from a detectable flaw being created to leakage occurring and from a leaking flaw to the time that net section collapse (nozzle ejection) would be predicted to occur. Applying a conservative crack growth FOI of 20 to circumferential and ID axial cracking and an FOI of 10 to OD axial cracking for Alloy 690 versus Alloy 600, the results show that more than 20 years is required for leakage to occur and that more than 120 years would be required to reach the critical crack size subsequent to leakage.

The probabilistic model in MRP-375 was developed to predict PWSCC degradation and its associated risks in RVCHs. The model utilized in this probabilistic evaluation is modified from the model presented in Appendix B of MRP-335, Revision 1 [Reference 8] that evaluated surface stress improvement of RVCHs with Alloy 600 nozzles. The integrated probabilistic model in MRP-375 includes sub-models for simulating component and crack stress conditions, PWSCC initiation, PWSCC growth, and flaw examination. The sub-models for crack initiation and growth prediction for Alloy 600 reactor pressure vessel head penetration nozzles in MRP-335, Revision 1, were adapted for RVCHs with Alloy 690 nozzles by applying FOIs to account for the superior PWSCC resistance of Alloys 690/52/152. The average leakage frequency and average ejection frequency were determined using the Monte Carlo simulation model with conservative FOI assumptions. The results show that, using only modest FOIs for Alloys 690/52/152, the potential for developing a safety significant flaw (risk of nozzle ejection) is acceptably small for a volumetric/surface examination period up to 40 years.

The evaluations performed in MRP-375 were prepared to bound all Pressurized Water Reactor (PWR) replacement RVCH designs that are manufactured using Alloy 690 base material and Alloy 52/152 weld materials. The evaluations assume a continuously operating RVCH temperature of 613°F and a relatively large number of RVCH penetrations (89).

While approval of this I&M request for alternative is not contingent on U. S. Nuclear Regulatory Commission (NRC) review and approval of MRP-375, the insights gained in this technical report help substantiate the limited extension duration being requested. In particular, the tabulation of crack growth rate data for Alloys 690/52/152 (Section 3 of MRP-375) and review of inspection experience for Alloy 690/52/152 plant components (Section 2 of MRP-375) are sufficient to demonstrate the acceptability of the limited extension duration being requested. This request is not dependent on the more detailed probabilistic calculations presented in Section 4 of MRP-375.

RVCH Design and Operation

The analysis presented in MRP-375 was intended to cover all replacement heads in U. S. PWRs, including the CNP Unit 1 and Unit 2 RVCHs. The MRP-375 analyses assume a

reactor vessel head operating temperature of 613°F to bound the known reactor vessel head temperatures of all U. S. PWRs currently operating. RVCH operating temperature considerations for CNP are as follows:

- For CNP Unit 1, the average RVCH operating temperature over the operating period from installation of the replacement head in 2006 until the Cycle 27 refueling outage that was scheduled for spring 2016 is 578°F [Reference 9].[3] During that outage, the Unit 1 reactor coolant system (RCS) was restored to "normal" operating pressure and temperature[4], after which the Unit 1 RVCH operating temperature is conservatively assumed to increase to 601°F (the same as Unit 2). Therefore, the final two operating cycles before the end of the requested volumetric/surface inspection period are assumed to be at the higher temperature.
- For CNP Unit 2, the average RVCH operating temperature over the operating period from installation of the replacement head in 2007 until the end of the requested volumetric/surface inspection period is 601°F [Reference 9].

Based on the above, the CNP Unit 1 and Unit 2 RVCH average operating temperature, which is the measure of temperature relevant to potential PWSCC degradation, is bounded by the MRP-375 evaluation that assumes 613°F for its main deterministic and probabilistic calculations.

The CNP Unit 1 and Unit 2 RVCHs each contain 60 nozzle penetrations, of which 53 are used for control rod drive mechanisms, five are used for in-core thermocouples, and two are small-diameter penetrations near the center of the RVCH used for vent and RVLIS pipes. The replacement RVCHs were manufactured by Framatome ANP, Inc. and placed in service in November 2006 and November 2007, respectively. The replacement RVCHs were manufactured as single forgings, which eliminated all circumferential and meridional welds in the original RVCHs. The replacement RVCHs are fabricated from SA-508, Class 3 low-alloy steel and clad with an initial layer of 309L stainless steel followed by subsequent layers of 308L stainless steel. The nozzle housing penetrations and small diameter vent and RVLIS connections on the replacement RVCHs are fabricated from SB-167 (Alloy 690) UNS N06690. The penetration nozzle J-groove welds utilized ERNiCrFe-7 (UNS N06052) and ENiCrFe-7 (UNS W86152) weld materials.

Note that the probabilistic analysis in MRP-375 was performed assuming a head with 89 partial-penetration welded nozzles, which bounds the number of penetrations in the CNP replacement RVCHs. The number of penetrations included in the probabilistic model is not a key variable, and the assumed number of penetrations results in a small change in results

[3] The CNP Unit 1 and Unit 2 RVCH operating temperatures cited from MRP-48 [Reference 9] were developed by Westinghouse in support of evaluations of the original Alloy 600 RVCH nozzles, but continue to be representative of current plant operation.

[4] I&M previously submitted a license amendment request [Reference 17] to allow restoration of CNP Unit 1 RCS temperature and pressure to typical PWR conditions. The NRC granted approval of this request on November 30, 2015 [Reference 27].

relative to other sensitivity cases. Thus, the probabilistic calculations of MRP-375 cover all U. S. replacement RVCHs regardless of the precise number of penetrations.
Preservice volumetric examinations of the CNP Unit 1 and Unit 2 replacement RVCH partial-penetration welded nozzles were performed prior to installation using eddy current (ET) and ultrasonic (UT) examination techniques. The volumetric examinations included scanning the nozzles to the fullest extent possible, from the end of the nozzle to a minimum of 2 inches above the root of the J-groove weld on the uphill side. No ET or UT responses indicative of planar degradation were found in any of the penetrations or welds.

Bare metal visual examinations were performed on the CNP Unit 1 replacement RVCH in 2011 and 2016, and on the Unit 2 replacement RVCH in 2012 and 2016, in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. The visual examinations were performed by visual testing (VT)-2 qualified examiners on the outer surface of the RVCHs including the annulus area of the penetration nozzles. The examinations did not reveal any surface or nozzle penetration boric acid that would be indicative of nozzle leakage. These examinations will be performed again in the Unit 1 Cycle 29 refueling outage scheduled for spring 2019 and the Unit 2 Cycle 25 refueling outage scheduled for fall 2019, unless relief is granted by this request. Should this relief be granted, only Volumetric and Surface examinations will be performed as specified in 10 CFR 50.55a(g)(6)(ii)(D)(3)(i).

Minimum FOI Implied by Requested Inspection Period

ASME Code Case N-729-4 is based upon conclusions reached [Reference 10] that a reexamination interval between volumetric/surface examinations of one 24-month operating cycle is acceptable for a head with Alloy 600 nozzles and operating at a temperature of 605°F. The inspection period for heads with Alloy 690 nozzles in Case N-729-4 is a nominal 10 years, which represents a minimum implied FOI of 5 over Alloy 600.

<u>FOI Approach</u>

Per the technical basis documents for ASME Code Case N-729-4 for heads with Alloy 600 nozzles [References 5, 10, and 11], the effect of differences in operating temperature on the required volumetric/surface reexamination interval for heads with Alloy 600 nozzles can be addressed on the basis of the Re-Inspection Years (RIY) parameter. The RIY parameter adjusts the effective full power years (EFPYs) of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth. For heads with Alloy 600 nozzles, ASME Code Case N-729-4 as conditioned by 10 CFR 50.55a(g)(6)(ii)(D) limits the interval between subsequent volumetric/surface inspections to RIY = 2.25. The RIY parameter, which is referenced to a head temperature of 600°F, limits the time available for potential crack growth between inspections.

The RIY parameter for heads with Alloy 600 nozzles is adjusted to the reference head temperature using an activation energy of 130 kilojoules per mole (kJ/mol) [31 kilocalories per mole (kcal/mol)] [Reference 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 Argonne National Laboratory (ANL) reported in NUREG/CR-7137 [Reference 12] indicate that Alloy 690 with 0-26% cold work has an activation energy value between 100 and 165 kJ/mol (24-39 kcal/mol). NUREG/CR-7137 concludes that the activation energy for Alloy 690 is comparable to the standard value for Alloy 600 (130 kJ/mol).

(2) Testing at Pacific Northwest National Laboratory (PNNL) found an activation energy value of about 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work [Reference 13].

(3) Additional PNNL testing determined an activation energy value of 123 kJ/mol (29.4 kcal/mol) for Alloy 690 with 31% cold work [Reference 14].

These data show that it is reasonable to assume the same crack growth thermal activation energy as was determined for Alloys 600/82/182, namely 130 kJ/mol (31 kcal/mol), for modeling growth of hypothetical PWSCC flaws in Alloys 690/52/152 PWR plant components.

As discussed in the MRP-117 [Reference 10] technical basis document for RVCHs with Alloy 600 nozzles, effective time for crack growth is the principal basis for setting the appropriate reexamination interval to detect any PWSCC in a timely fashion. U. S. PWR inspection experience for heads with Alloy 600 nozzles has confirmed that the RIY = 2.25 interval results in a suitably conservative inspection program. There have been no reports of nozzle leakage or of safety-significant circumferential cracking for times subsequent to the time that the Alloy 600 nozzles in a head were first examined by non-visual inservice non-destructive examination [References 15 and 16].

FOI Implied by Requested Inspection Period for CNP Unit 1 and Unit 2

I&M has assessed the minimum Alloy 690/52/152 FOI that supports the requested CNP Unit 1 and Unit 2 extension periods for comparison with the laboratory crack growth rate data presented in MRP-375. Based on the previously stated conclusion that a reexamination interval between volumetric/surface examinations of one 24-month operating cycle is acceptable for a head with Alloy 600 nozzles and operating at a temperature of 605°F, an extension of the CNP examination interval to 13 years[5] would imply a factor of 13/2 or 6.5 for Alloys 690/52/152 relative to Alloys 600 and 182 for the proposed period between volumetric/surface examinations for a head operated at a temperature of 605°F. To calculate

[5] The nominal interval of 13 years was selected by conservatively adding two 18-month fuel cycles to the nominal inspection interval of 10 years. The actual proposed intervals between RVCH installation and the first volumetric inspection for CNP Unit 1 and Unit 2 are 12.4 and 11.9 calendar years, respectively. The projected EFPYs at the end of the proposed RVCH inspection intervals for CNP Unit 1 and Unit 2 are 10.3 and 11.0 EFPYs, respectively.

the minimum implied FOI for the Unit 1 and Unit 2 RVCHs operating temperature of 601°F[6], the RIY parameter for the requested examination interval is compared with the N-729-4 interval for Alloy 600 nozzles of RIY = 2.25.

The representative CNP Unit 1 and Unit 2 RVCH operating temperature of 601°F corresponds to an RIY temperature adjustment factor of 1.025 (versus the reference temperature of 600°F) using the activation energy of 130 kJ/mol (31 kcal/mol) for crack growth of ASME Code Case N-729-4. As discussed previously, it is appropriate to apply this standard activation energy for modeling crack growth of Alloy 690/52/152 plant components. Conservatively assuming that the EFPYs of operation accumulated at CNP Unit 1 and Unit 2 since RVCH replacement is equal to the calendar years since replacement, the RIY for the requested extended periods for CNP Unit 1 and Unit 2 would be (1.025 temperature factor for growth rate) x (13 total calendar years for extended interval) = 13.33 RIY_{690}. The FOI implied by this RIY value for CNP Unit 1 and Unit 2 is 13.33/2.25 = 5.9.

Considering the statistical compilation of data provided in Figures 3-2, 3-4, and 3-6 of MRP-375, this factor of improvement is conservatively less than the FOI of 10 that statistically bounds the crack growth rate data presented. Furthermore, as discussed in Sections 2 and 3 of MRP-375, PWR plant experience and laboratory testing have demonstrated a large improvement in resistance to PWSCC initiation of Alloys 690/52/152 in comparison to that for Alloys 600/82/182. Therefore, the demonstrated improvements in PWSCC initiation and growth confirm, on a conservative basis, the acceptability of the limited requested period of extension.

Comparison to ANL and PNNL Alloys 690/52/152 Crack Growth Data (ML14322A587)

Although the foregoing discussion of the proposed alternative RVCH examination interval in the context of MRP-375 crack growth data is consistent with current industry positions regarding the corrosion resistance of replacements RVCHs, I&M recognizes that MRP-375 is not an NRC-approved document, nor did I&M request review and approval of MRP-375 for the proposed alternative. Consequently, the following discussion of the proposed alternative inspection interval in the context of Alloys 690/52/152 crack growth data, developed for the NRC by ANL and PNNL and documented in an NRC memorandum (ML14322A587) [Reference 24], is provided.

Report ML14322A587 includes crack growth rate test results for the following:

- ANL
    - Alloy 690 and Alloy 690 heat affected zone (HAZ)
    - Alloy 152 in the as-welded condition
    - Alloys 52/152 deposited as weld overlays on Alloy 182 and SA-533 (low alloy steel), respectively

[6] The higher historical operating temperature of Unit 2 compared to Unit 1 was conservatively selected as the basis for a bounding plant assessment.

- PNNL
    - Alloy 690 and Alloy 690 HAZ, zero to 22% cold work
    - Alloys 52/152

The ANL crack growth rates for weld overlay testing are not considered pertinent to a modest requested inspection interval extension for replacement RVCH configurations and were not included in this evaluation. The remaining data were plotted against the CNP proposed FOI = 5.9, with the results shown in Figures CNP-1 through CNP-5, as discussed below:[7]

Figure CNP-1

- Figure CNP-1 provides a representation of the ANL test data reported in ML14322A587 for Alloy 690 and Alloy 690 HAZ samples in comparison to the FOI = 5.9 proposed for CNP Unit 1 and Unit 2.

- Although Figure 1 of ML14322A587 compares the ANL test data to the MRP-55 crack growth rate (CGR) curve for 325 degrees Celsius (°C), it also notes that the tests were performed in a simulated primary water environment at 320°C, which makes the standard reference curve non-conservative for comparison to the test data. Therefore, Figure CNP-1 also includes the MRP-55 CGR curve adjusted to 320°C.

- As shown in Figure CNP-1, all test data fall below the proposed FOI = 5.9 for 320°C, indicating that the proposed FOI is supported by the test results.

Figure CNP-2

- Figure CNP-2 provides a representation of the ANL test data reported in ML14322A587 for Alloy 152 welds in comparison to the FOI = 5.9 proposed for CNP Unit 1 and Unit 2.

- Figure 2 of ML14322A587 shows the MRP-115 CGR curve adjusted for the test temperature of 320°C. (Figure CNP-2 also includes the 325°C MRP-115 curve for reference.)

- As shown in Figure CNP-2, all test data fall below the proposed FOI = 5.9 for the test temperature of 320°C, with the exception of A152-TS-2 (CL), which is just slightly above the curve. Given that the MRP-115 curve was developed to statistically represent the 75^{th} percentile of Alloy 182 data, many Alloy 182 data points would fall above the MRP-115 reference curve. Therefore, it is not unexpected that an individual Alloy 152 data point may fall above the proposed FOI curve. In this particular case, the single point is only marginally above the proposed curve, with the remainder of the data falling

[7] Note that for consistency with ML14322A587, crack growth rate data from ANL are plotted in units of meters per second while crack growth rate data from PNNL are plotted in units of millimeters per second.

substantially below the curve, indicating that the proposed FOI is supported by the test results.

Figure CNP-3

- Figure CNP-3 provides a representation of the PNNL test data reported in ML14322A587 for non-cold worked Alloy 690 samples in comparison to the FOI = 5.9 proposed for CNP Unit 1 and Unit 2.

- The corresponding figure in ML14322A587 compares the PNNL test data to the MRP-55 CGR curve for 360°C since the majority of tests were performed in a simulated primary water environment at 360°C. However, since four of the sample series were tested at either 325°C or 350°C, Figure CNP-3 also includes the 325°C MRP-55 curve for reference.

- As shown in Figure CNP-3, all of the PNNL Alloy 690 non-cold worked data (tested in the range of 325°C to 360°C) fall below the proposed FOI = 5.9 for 325°C, indicating that the proposed FOI is bounded by the test results.

Figure CNP-4

- Figure CNP-4 provides a representation of the PNNL test data reported in ML14322A587 for Alloy 690 samples with zero to 22% cold work in comparison to the FOI = 5.9 proposed for CNP Unit 1 and Unit 2.

- The corresponding figure in ML14322A587 compares the PNNL test data to the MRP-55 CGR curve for 360°C since the majority of tests were performed in a simulated primary water environment at 360°C. However, since four of the sample series were tested at either 325°C or 350°C, Figure CNP-4 also includes the 325°C MRP-55 curve for reference.

- None of the samples tested at 325°C or 350°C fall above the FOI = 5.9 curve at 325°C and none of the samples tested at 360°C fall above the FOI = 5.9 curve at 360°C, indicating that the proposed FOI is bounded by the test results.

Figure CNP-5

- Figure CNP-5 provides a representation of the PNNL test data reported in ML14322A587 for Alloys 52/152 samples in comparison to the FOI = 5.9 proposed for CNP Unit 1 and Unit 2.[8]

[8] Unlike the other figures in ML14322A587, tabular data for the PNNL Alloys 52/152 sample points are not provided in the summary report. Therefore, Figure CNP-5 depicts the general area that bounds the test data as read from the corresponding figure in ML14322A587.

- The corresponding figure in ML14322A587 compares the PNNL test data to the MRP-115 CGR curves for 325°C and 360°C. However, since one of the sample series was tested at 320°C, Figure CNP-5 conservatively uses the 320°C MRP-55 curve for reference.

- As shown in Figure CNP-5, all of the plotted Alloys 52/152 data (tested from 320°C to 360°C) fall below the proposed FOI = 5.9 for 320°C, indicating that the proposed FOI is bounded by the test results.

Conclusions

It is concluded that the Alloy 690 nozzle base and Alloy 52/152 weld materials used in the CNP Unit 1 and Unit 2 replacement RVCHs provide for a superior RCS pressure boundary, where the potential for PWSCC has been shown by analysis and by years of positive industry experience to be remote. This conclusion is further supported by direct visual examination of the CNP Unit 1 replacement RVCH in 2011 and 2016, and on the Unit 2 replacement RVCH in 2012 and 2016 and the lack of PWSCC detected in the volumetric examinations performed to date of Alloy 690 nozzles in similar replacement RVCHs.

The minimum FOI implied by the requested extension period represents a level of reduction in PWSCC crack growth rate versus that for Alloys 600/82/182 that is completely bounded on a statistical basis by the laboratory data compiled in MRP-375. Given the lack of PWSCC detected to date in any PWR plant applications of Alloys 690/52/152, the simple FOI assessment clearly supports the requested period of extension.

Therefore, the requested periods of extension to perform volumetric/surface examinations of the CNP Unit 1 and Unit 2 RVCH nozzles provide an acceptable level of quality and safety in accordance with 10 CFR 50.55a(z).

6. Duration of Proposed Alternative

The proposed alternative is requested:

- For CNP Unit 1, for the duration up to and including the Cycle 29 refueling outage that is scheduled to commence in March 2019 and which will occur in the fourth 10-year ISI interval that began on March 1, 2010, and ends February 29, 2020.

- For CNP Unit 2, for the duration up to and including the Cycle 25 refueling outage that is scheduled to commence in October 2019 and which will occur in the fourth 10-year ISI interval that began on March 1, 2010, and ends February 29, 2020.

7. Precedents

There have been submittals from multiple plants to request an alternative from the frequency of ASME Code Case N-729-4 for volumetric or surface examinations of heads with Alloy 690 nozzles, as identified below. Two plants have received authorization for one-time use of the requested alternative.

- Arkansas Nuclear One, Unit 1; Request for Alternative from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1 [Reference 18]; one-time use authorized by NRC [Reference 25]

- St. Lucie Unit 1; Fourth Ten-Year Interval Unit 1 Relief Request No. 8 [Reference 19]; one-time use authorized by NRC [Reference 26]

- H. B. Robinson Unit 1; Relief Request (RR)-11 for Relief from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1; approved by NRC February 17, 2015 [Reference 20]

- Prairie Island Units 1 and 2; 10 CFR 50.55a Requests 1-RR-5-7 and 2-RR-5-7 Associated with the Fifth Ten-Year Interval for the Inservice Inspection Program; approved by NRC June 4, 2015 [Reference 21]

- Farley Unit 2; Proposed Inservice Inspection Alternative FNP-ISI-ALT-17, Version 1.0; approved by NRC May 5, 2015 [Reference 22]

- Beaver Valley Unit 1; Proposed Alternative to ASME Code Case N-729-1 Examination Frequency Requirements (Request 1TYP-4-RV-04); approved by NRC January 28, 2015 [Reference 23]

8. References

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20. Letter from S. R. Helton, NRC, to R. M. Glover, Duke Energy, "H. B. Robinson Steam Electric Plant, Unit No. 2 – Relief Request No. 11 – Relief from Volumetric/Surface Examination Frequency Requirements of ASME Code Case N-729-1 (TAC No. MF4801)," dated February 17, 2015. [ADAMS Accession No. ML15021A354]

21. Letter from D. L. Pelton, NRC to K. Davison, Xcel Energy, "Prairie Island Nuclear Generating Plant, Units 1 and 2 – Relief Requests (1-RR-5-7 and 2-RR-5-7) for Fifth 10-Year Interval for the Inservice Inspection Program (TAC Nos. MF4843 and MF4844)", dated June 4, 2015. [ADAMS Accession No. ML15125A361]

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23. Letter from M. Khanna, NRC, to E. A. Larson, First Energy Nuclear Operating Company, "Beaver Valley Power Station, Unit No. 1 – Relief Request No. 1TYP-4-RV-04 Regarding the Examination Requirements of Code Case N-729-1(TAC No. MF5049)," dated January 28, 2015. [ADAMS Accession No. ML14363A409]

24. Memorandum from M. Srinivasan, Office of Nuclear Regulatory Research, to D. W. Alley, Office of Nuclear Reactor Regulation, "Transmittal of Preliminary Primary Water Stress Corrosion Cracking Data for Alloys 690, 52, and 152," dated October 30, 2014. [ADAMS Accession No. ML14322A587]

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26. Letter from L. M. Regner, Office of Nuclear Reactor Regulation, to M. Nazar, NextEra Energy, "St. Lucie Plant Unit No. 1 – Relief Request No. 8 – Alternative From Performing Volumetric/Surface Examinations of Reactor Vessel Closure Head Components (TAC No. MF4490)," dated December 23, 2014. [ADAMS Accession No. ML14339A163]

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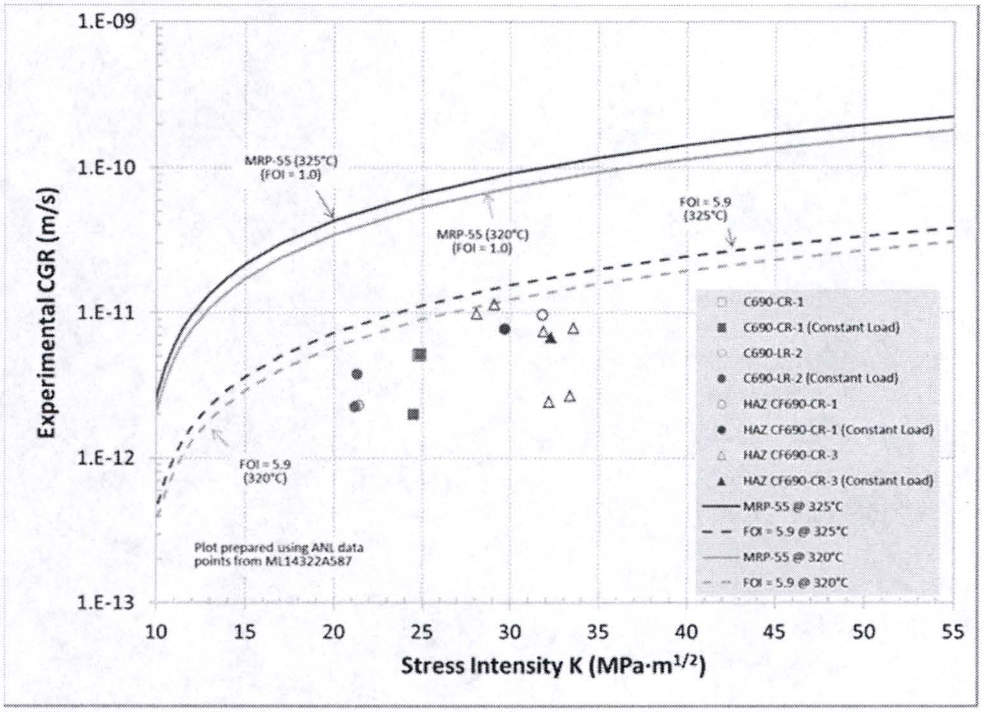

Figure CNP-1: ANL Alloy 690 and Alloy 690 HAZ

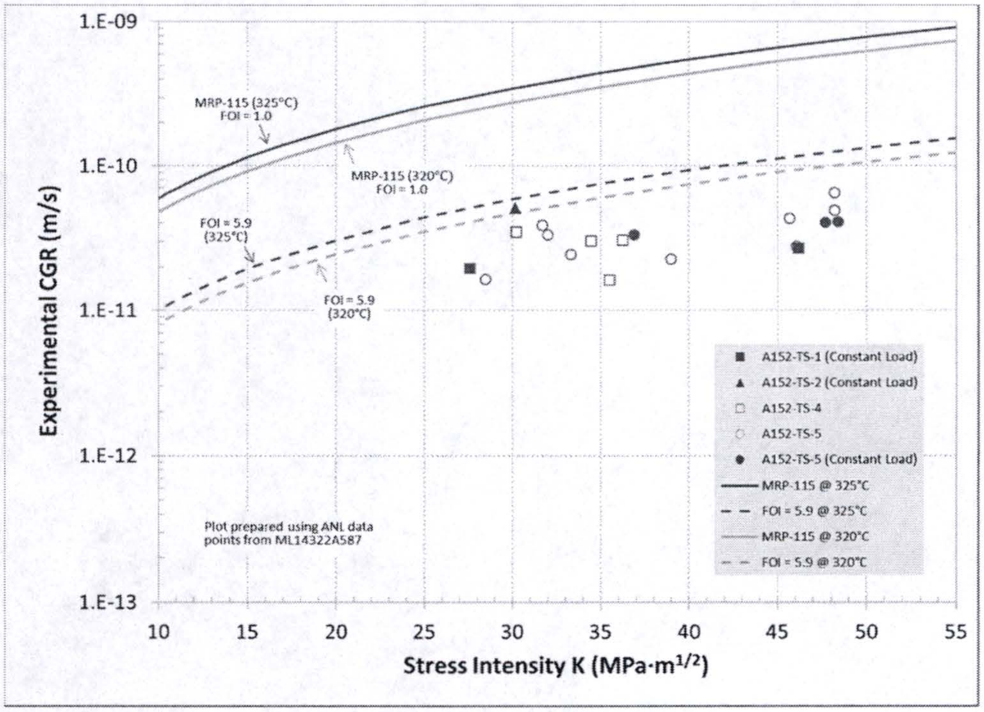

Figure CNP-2: ANL Alloy 152

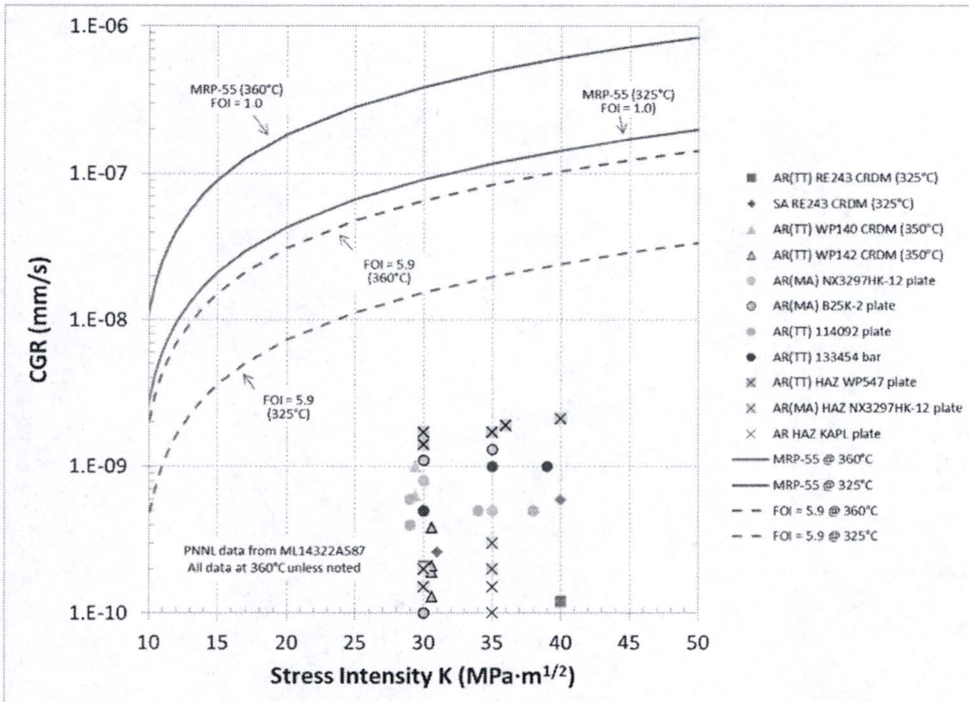

Figure CNP-3: PNNL Alloy 690 and Alloy 690 HAZ Data (No Cold Work)

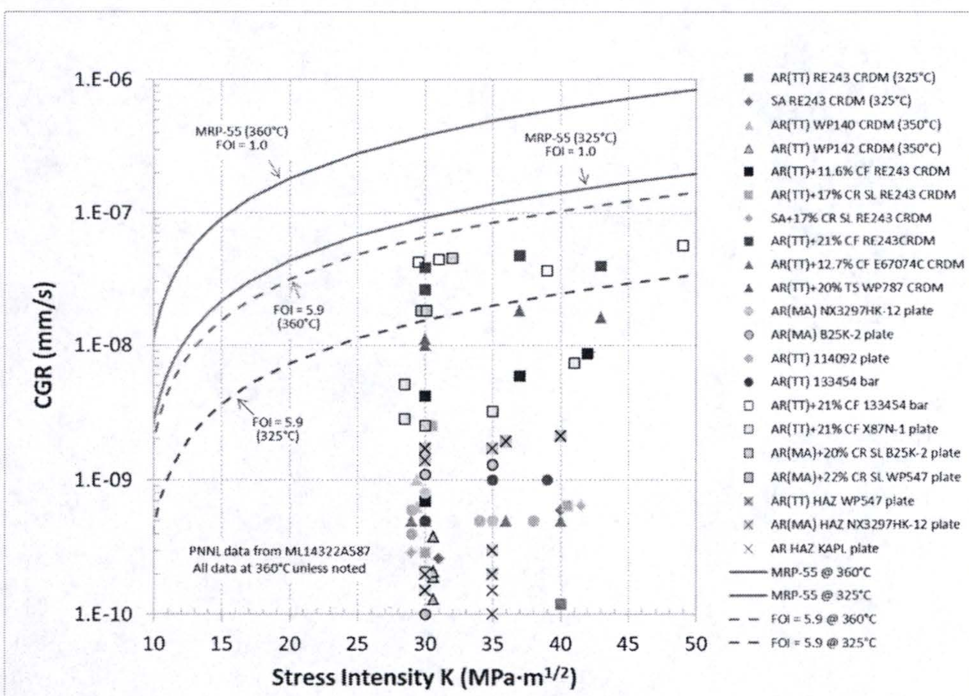

Figure CNP-4: PNNL Alloy 690 and Alloy 690 HAZ (0 - 22% Cold Work)

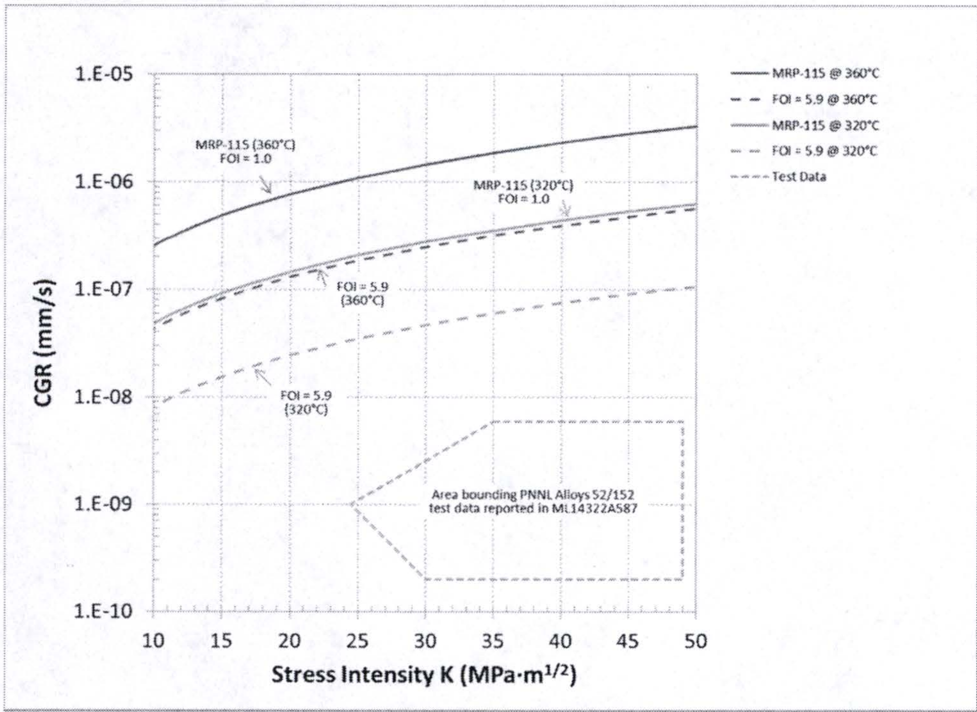

Figure CNP-5: PNNL Alloys 52/152