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AEP-NRC-2018-50
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, Unit 1 and Unit 2
Request for Alternative from Volumetric/Surface Examination
Frequency Requirements of ASME Code Case N-729-4

Pursuant to 10 CFR 50.55a(z)(1), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, hereby requests approval of a proposed alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, reactor pressure vessel head examination frequency requirements for CNP Unit 1 and Unit 2. The proposed alternative is included as the enclosure to this letter.

I&M would like to request U. S. Nuclear Regulatory Commission (NRC) review and approval of the proposed alternative by January 31, 2019, to facilitate planning for the next CNP Unit 1 refueling outage currently scheduled to occur during spring 2019.

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

Sincerely,

Quinton S. Lies

Q. Shane Lies
Site Vice President

BMC/mll

Enclosure: 10 CFR 50.55a Relief Request Number ISIR 04-09, Request for Alternative from Reactor Vessel Closure Head (RVCH) Examination Frequency Requirements

A047
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Enclosure to AEP-NRC-2018-50

10 CFR 50.55a Relief Request Number ISIR-4-09

Request for Alternative from Reactor Vessel Closure Head (RVCH) Examination Frequency Requirements

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

Pursuant to 10 CFR 50.55a(z)(1), Indiana Michigan Power Company (I&M), the licensee for Donald C. Cook Nuclear Plant (CNP) Unit 1 and Unit 2, hereby requests approval of a proposed alternative to the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section XI, reactor pressure vessel head examination frequency requirements for CNP Unit 1 and Unit 2. The affected components are the CNP Unit 1 and Unit 2 ASME Class 1 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 Boiler and Pressure Vessel (BPV) Code Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," 2004 Edition, with no Addenda.

3. Applicable Code Requirement

The Code of Federal Regulations 10 CFR 50.55a(g)(6)(ii)(D)(*1*), requires, (in part):

> 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)(*2*) conditions ASME Code Case N-729-4 (Reference 1) by stating:

> Appendix I of ASME BPV Code Case N-729-4 shall not be implemented without prior U. S. Nuclear Regulatory Commission (NRC) approval.

10 CFR 50.55a(g)(6)(ii)(D)(*3*) conditions ASME Code Case N-729-4 bare metal visual examination frequency and is not relevant to this request.

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

> In addition to the requirements of Paragraph–3132.1(b) of ASME BPV Code Case N-729-4, a component whose surface examination detects rounded indications greater than allowed in Paragraph NB–5352 in size on the partial-penetration or associated fillet weld shall be

classified as having an unacceptable indication and corrected in accordance with the provisions of [P]aragraph–3132.2 of ASME BPV Code Case N-729–4.

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

- Volumetric or surface examination of all nozzles every inspection interval (nominally ten calendar years) provided that flaws attributed to PWSCC have not previously been identified in the head; and
- Direct visual examination of the outer surface of the head for evidence of leakage every third refueling outage or five calendar years, whichever is less.

4. Reason for Request

ASME Code Case N-729-4 (Reference 1) with the conditions of 10 CFR 50.55a(g)(6)(ii)(D) requires volumetric/surface examination of the RPV upper head nozzles and welds at an interval of once per nominal ten calendar years after the head was placed into service.

The CNP RVCH penetration nozzles and associated welds are made from Alloys 690/52/152. As discussed in Electric Power Research Institute (EPRI) Materials Reliability Program (MRP) report, "Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386)" (Reference 2), compared to Alloys 600/82/182, these materials have much greater PWSCC resistance. Excellent operating experience with no observations of PWSCC in almost 30 years of service supports the superiority of Alloy 690 relative to Alloy 600 in pressurized water reactor (PWR) primary water environments, as does extensive laboratory testing.

As stated in EPRI MRP report: "Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles (MRP-375)" (Reference 3):

> ...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. Since that time, plant experience and laboratory testing have continued to demonstrate the much greater resistance of these replacement alloys to PWSCC compared to that for Alloys 600/82/182 for the material conditions relevant to partial-penetration welded nozzles. Although laboratory research is ongoing to investigate and understand the times to crack initiation and the crack growth rates for these materials under various conditions, there are now sufficient data available to develop an improved technical basis for inspection of these components.

Research documented in MRP-386 (Reference 2) further demonstrates the much greater resistance of these replacement alloys to PWSCC as compared to Alloys 600/82/182 for the conditions relevant to partial-penetration welded nozzles.

The technical bases of References 2 and 3 together demonstrate that the reexamination interval can be extended to a 15 year interval length or longer while maintaining an acceptable level of quality and safety. 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 10.55a(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 Case N-729-4 (Reference 1). Specifically, I&M requests to extend the frequency of the volumetric/surface examination of the CNP Unit 1 and Unit 2 RVCH of Table 1, Item B4.40 of ASME Code Case N-729-4 for one additional fuel cycle beyond that approved by the United States (U.S.) NRC in accordance with CNP Relief Request ISIR 04-05 (Reference 4).

- For CNP Unit 1, this request would extend the volumetric/surface examination currently scheduled for the Cycle 29 refueling outage in spring 2019 to the Cycle 30 refueling outage that is scheduled for fall 2020. At that point the Unit 1 RVCH will have been in service for approximately 13.9 calendar years.

- For CNP Unit 2, this request would extend the volumetric/surface examination currently scheduled for the Cycle 25 refueling outage in fall 2019 to the Cycle 26 refueling outage that is scheduled for spring 2021. At that point the Unit 2 RVCH will have been in service for approximately 13.4 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 (Reference 1) 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 NRC approval of Relief Request ISIR 04-05 (Reference 4), I&M will be 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 29 refueling outage that is scheduled for spring 2019 (12.4 calendar years following replacement)

- For CNP Unit 2, during the Cycle 25 refueling outage that is scheduled for fall 2019 (11.9 calendar years following replacement).

The basis for volumetric and/or surface examination frequency for heads with Alloy 690 nozzles of ASME Code Case N-729 through N-729-4 comes, in part, from the analysis performed in EPRI report MRP-111 (Reference 5), which was summarized in the safety assessment for RVCHs in MRP-110 (Reference 6). The material improvement factor for PWSCC of Alloy 690/52/152 materials over that of mill-annealed Alloys 600/82/182 was shown by this report to be in the order of 26 or greater. In August of 2009 EPRI released MRP-258 (Reference 7), which updates MRP-111, stating that "Relative improvement factors of 40 - 100 times versus Alloy 600 can now be derived for the initiation of cracking, but these numbers are clearly conservative, due to an absence of PWSCC in almost all Alloy 690 specimens within the test duration."

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 ASME Code Case N-729 (Revisions 1 through 4) was developed in 2004 with deterministic and probabilistic evaluations to assess the improved PWSCC resistance of Alloys 690/52/152 relative to Alloys 600/82/182. Additional research was recently performed under an EPRI MRP initiative provided in MRP-386 (Reference 2). This report compiled over 530 Alloy 690 Crack Growth Rate (CGR) data points and over 130 Alloy 52/152 CGR data points from seven research laboratories further supporting the improved PWSCC resistance of Alloys 690/52/152.

Evaluation of Alloys 690/52/152 Data and Experience by MRP-375 (Reference 3)

As documented in MRP-375, operating experience for replacement and repaired components using Alloys 690/52/152 has shown a proven record of resistance to PWSCC during numerous examinations in the nearly 30 years of its application. This includes steam generators, pressurizers, and RVCHs.

In particular, at the completion of the spring 2018 refueling outage season, Alloy 690/52/152 operating experience includes inservice volumetric or surface examinations performed on 16 of the 40 currently operating plant replacement RVCHs in the U. S. in accordance with the Augmented ISI Requirements for Reactor Vessel Head Inspections.

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

Table 3-6 of MRP-375 (Reference 3), provides a summary of CGR and crack initiation data. For crack initiation, FOIs reported, although significant, are conservative because in many cases 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. Additionally, many of the Alloy 690 CGR tests were performed on specimens with considerable amounts of cold work (up to 40%), which is known to accelerate CGRs to rates that are not representative of cold work levels applicable to reactor vessel head penetrations.

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

The deterministic crack growth evaluation provides a precursor to the probabilistic evaluation to directly illustrate the relationship between the improved PWSCC growth resistance of Alloys 690/52/152 and the time to certain adverse conditions. These evaluations apply conservative CGR 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 which 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 inside diameter axial cracking and of 10 to outside diameter axial cracking for Alloys 690/52/152 versus Alloys 600 and 182, 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 (Reference 3) 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 Alloy 600 RVCHs. The integrated probabilistic model in MRP-375 includes submodels for simulating component and crack stress conditions, PWSCC initiation, PWSCC growth, and flaw examination. The submodels for crack initiation and growth prediction for Alloy 600 RPV head penetration nozzles in MRP-335, Revision 1, were adapted for Alloy 690 RVCHs by applying FOIs to account for its superior PWSCC resistance. The probabilistic calculations are based on a Monte Carlo simulation model including PWSCC initiation, crack growth, and flaw detection via ultrasonic testing. The average leakage frequency and average ejection frequency were determined using conservative FOI assumptions.

The results show that using only modest FOIs for Alloy 690/52/152 RVCHs, the potential for developing a safety significant flaw (risk of nozzle ejection) is acceptably small for a volumetric or surface examination period of 20 years.

The evaluations performed in MRP-375 (Reference 3) were prepared to bound all PWR replacement RVCH designs manufactured using Alloy 690 base material and Alloy 52/152 weld materials. The evaluations assume a bounding continuously operating RVCH temperature of 613 degrees Fahrenheit (°F) and a relatively large number of RVCH penetrations (89).

While approval of this I&M request for alternative is not contingent on NRC review and approval of MRP-375 (Reference 3), the insights gained in this technical report help substantiate the limited extension duration being requested. In particular, the tabulation of CGR 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.

Additional Evaluations Performed under MRP-386 (Reference 2)

MRP-386 summarizes years of laboratory testing by an international group of experts to quantify the PWSCC growth rates of Alloy 690 and its weld metals, Alloy 52/152, in simulated PWR primary water. Fracture mechanics-based tests were conducted under testing conditions designed to promote PWSCC in several product forms of wrought Alloy 690 and in several alloy variants of weld metal Alloy 52/152. For some Alloy 690 tests, laboratory-added plastic strain (i.e. "cold work") of up to 30% reduction in thickness was used to accelerate PWSCC growth rates. Variables known to affect PWSCC were assessed and included in the CGR model and/or disposition equations, including the mode I stress intensity factor, the test temperature, the yield strength of the material, the electrochemical potential in the test environment, and the orientation of the crack relative to the direction of added cold work. The data was vetted by an international expert panel and was then used to develop predictive models of the PWSCC growth rate in thick-walled Alloy 690 (including the heat-affected zone) and its weld metals, Alloys 52 and 152, and variants of these alloys. The lower bound FOI for Alloy 690 compared to Alloy 600 is 25, while the more realistic and recommended FOI is 38. For Alloy 52/152 compared to Alloy 182, the lower bound FOI is 253, while the recommended FOI is 324.

RVCH Design and Operation

The analysis presented in MRP-375 (Reference 3) 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 RVCH operating temperature at the time of the installation of the replacement head in 2006 was 578°F (Reference 9). In November of 2015 the NRC issued an amendment (Reference 10) allowing CNP Unit 1 to increase normal reactor coolant system (RCS) temperature and pressure consistent with previously licensed conditions, increasing full power average coolant temperature by 15°F. To ensure this alternative request remains conservative, the CNP Unit 1 RVCH operating temperature is assumed to be equal to that of CNP Unit 2 at 601°F.

- 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 (Reference 3) 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 (Reference 3) 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 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 most recently performed on the CNP Unit 1 and Unit 2 replacement RVCHs in the spring and fall of 2016, respectively, in accordance with ASME Code Case N-729-1, Table 1, Item B4.30. The visual examinations were performed 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.

FOI Approach and Minimum FOI Implied by Requested Inspection Period

ASME Code Case N-729-4 (Reference 1) is based on conclusions reached in MRP-117 (Reference 11) 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 ASME Code Case N-729-4 is a nominal ten years, which represents a minimum implied FOI of 5 over Alloy 600.

FOI Approach

Per the technical basis documents for ASME Code Case N-729-4 for heads with Alloy 600 nozzles (References 6, 11, and 12), the effect of differences in operating temperatures on the required volumetric/surface reexamination interval for heads with Alloy 600 nozzles can be easily addressed on the basis of the Re-Inspection Years (RIY) parameter. The RIY parameter adjusts the effective full power years (EFPYs) of operation between inspections for the effect of head operating temperature using the thermal activation energy appropriate to PWSCC crack growth. For heads with Alloy 600 nozzles, ASME Code Case N-729-4 (Reference 1) 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 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 reported in NUREG/CR-7137 (Reference 13) indicate that Alloy 690 with 0-26% cold work has an activation energy 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 found activation energy of approximately 120 kJ/mol (28.7 kcal/mol) for Alloy 690 materials with 17-31% cold work (Reference 14).

This testing data shows that it is reasonable to assume the same crack growth thermal activation energy as was determined for Alloys 600/82/182, namely 130kJ/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 11) 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. 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).

Minimum FOI Implied by Requested Inspection Period

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 CGR data presented in MRP-375 (Reference 3). 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 operating at a temperature of 605°F, an extension of the CNP examination interval to 14.5 years would imply a factor of 14.5/2 or 7.25 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 the minimum implied FOI for the CNP Unit 1 and Unit 2 RVCHs operating temperature of 601°F, the RIY parameter for the requested examination interval is compared with the ASME Code Case 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 (Reference 1). As discussed previously, it is appropriate to apply this standard activation energy for modeling crack growth of Alloys 690/52/152 plant components. Conservatively assuming that the EFPYs of operation accumulated at CNP Unit 1 and Unit 2 since RVCH replacement are equal to the calendar years since replacement, the RIY for extended periods for CNP Unit 1 and Unit 2 would be (1.025 temperature factor for growth rate) x (14.5 total calendar years for extended interval) = 14.9 RIY_{690}. The FOI implied by this RIY value for CNP Unit 1 and Unit 2 is 14.9/2.25 = 6.6.

Considering the statistical compilation of data provided in Figures 3-2, 3-4, and 3-6 of MRP-375 (Reference 3), this factor of improvement is conservatively less than the FOI of 10 that statistically bounds the crack growth rate data presented in MRP-375 and is less than one third the minimum FOI of 25 presented in MRP-386 (Reference 2). 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.

Conclusions

It is concluded that the Alloy 690 nozzle base and Alloys 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 and Unit 2 RVCHs in the spring and fall of 2016, respectively, 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 bounded on a statistical basis by the laboratory data compiled in MRP-375 (Reference 3). 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)(1).

6. Duration of Proposed Alternative

The proposed alternative is requested:

- For CNP Unit 1, for the duration up to and including the Cycle 30 refueling outage that is scheduled to commence in fall 2020 and which will occur in the fifth 10-year ISI interval that begins on March 1, 2020, and ends February 28, 2030.

- For CNP Unit 2, for the duration up to and including the Cycle 26 refueling outage that is scheduled to commence in spring 2021 and which will occur in the fifth 10-year ISI interval that begins on March 1, 2020, and ends February 28, 2030.

7. Precedents

There have been many submittals from multiple plants requesting an alternative from the normal 10-year interval of ASME Code Case N-729-1 (and now ASME Code Case N-729-4) for volumetric/surface examinations of RVCHs with Alloy 690 nozzles. These include Calvert Cliffs Nuclear Power Plant (Reference 17), Arkansas Nuclear One (Reference 18), Beaver Valley Power Station, Unit 1 (Reference 19), and St. Lucie Plant, Unit 1 (Reference 20). Alternative intervals greater than 15 years have previously been granted in order to align with scheduled refueling outages. The alternatives approved for three sites (References 18, 19, and 20) extended the inspection interval from an initial approved alternative to a total interval of up to 15.5 years. The approved alternative for Calvert Cliff Unit 1 and Unit 2 (Reference 17) permitted an inspection interval not to exceed 16 years in order to align with scheduled refueling outages.

8. References

1. ASME Code Case N-729-4, "Alternative Examination Requirements for PWR Reactor Vessel Upper Heads with Nozzles Having Pressure-Retaining Partial-Penetration Welds," Section XI, Division 1, approved June 22, 2012.

2. Materials Reliability Program: Recommended Factors of Improvement for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) Growth Rates of Thick-Wall Alloy 690 Materials and Alloy 52, 152, and Variants Welds (MRP-386). EPRI, Palo Alto, CA: Final Report - December 2017 (Report No. 3002010756).

3. 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: February 2014 (Report No. 3002002441).

4. Letter from NRC to Indiana Michigan Power Company, "Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2 - Request for Use of Alternative ISIR 04-05, Revision 1, Associated with Reactor Vessel Closure Head Volumetric/Surface Examination Frequency Requirements for the Inservice Inspection Program (EPID L-2018-LLR-0027)," dated April 19, 2018 (NRC ADAMS Accession No. ML18103A059).

5. 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 and U.S. Department of Energy, Washington, DC: March 2004 (Report No. 1009801, NRC ADAMS Accession No. ML041680546).

6. Materials Reliability Program: Reactor Vessel Closure Head Penetration Safety Assessment for U.S. PWR Plants (MPR-110 NP). EPRI, Palo Alto, CA: April 2004 (Report No. 1009807-NP, NRC ADAMS Accession No. ML041680506).

7. Materials Reliability Program: Resistance to Primary Water Stress Corrosion Cracking of Alloy 690 in Pressurized Water Reactors (MRP-258). EPRI, Palo Alto, CA: August 2009 (Report No. 1019086).

8. Materials Reliability Program: Topical Report for Primary Water Stress Corrosion Cracking Mitigation by Surface Stress Improvement (MRP-335, Revision 1). EPRI, Palo Alto, CA: January 2013 (Report No. 3002000073).

9. PWR Materials Reliability Program: Response to NRC Bulletin 2001-01 (MRP-48). EPRI, Palo Alto, CA: August 2001 (Report No. 1006284).

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