UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 21, 2015

Mr. Dennis L. Koehl
President and CEO/CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 1 - REQUEST FOR RELIEF NO. RR-ENG-3-17 FOR EXTENSION OF THE INSPECTION FREQUENCY OF THE REACTOR VESSEL COLD-LEG NOZZLE TO SAFE-END WELDS WITH FLAW ANALYSIS (TAC NO. MF6174)

Dear Mr. Koehl:

By letter dated April 24, 2015, as supplemented by letters dated June 25 and July 22, 2015, STP Nuclear Operating Company (STPNOC, the licensee) requested relief from the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), for the volumetric examination of the reactor pressure vessel cold-leg nozzle dissimilar metal butt welds at the South Texas Project (STP), Unit 1.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(a)(3)(ii), you proposed an alternative frequency of volumetric examination for the cold-leg nozzle butt welds on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. You requested relief from the frequency of ISI for dissimilar metal butt welds as required by ASME Code Case N-770-1, "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated with UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1."

By *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Relief requests that had been previously covered by 10 CFR paragraph 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR paragraph 50.55a(z)(1). Relief requests that had been previously covered by 10 CFR paragraph 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR paragraph 50.55a(z)(2).

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that the proposed alternative of a one-time operating cycle extension of the third 10-year ISI interval to the end of the Unit 1 spring 2017 refueling outage provides reasonable assurance of leak tightness and structural integrity. The NRC staff has determined that complying with the specified ASME Code requirement would pose a hardship to the licensee because of the additional staff radiation exposure and safety

hazards resulting from two separate core barrel lift operations. Thus, complying with ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that you have adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes relief request RR-ENG-3-17 at STP Unit 1, for the third 10-year ISI interval which began on September 25, 2010, and ends on September 24, 2020.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions concerning this matter, please contact Ms. Lisa Regner of my staff at 301-415-1906 or via e-mail at Lisa.Regner@nrc.gov.

Sincerely,

Michael T. Markley

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498

Enclosure:
Safety Evaluation

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-ENG-3-17 REGARDING DEFERRAL OF INSERVICE INSPECTION

OF REACTOR PRESSURE VESSEL COLD-LEG NOZZLE DISSIMILAR METAL BUTT WELDS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 1

DOCKET NO. 50-498

1.0 INTRODUCTION

By letter dated April 24, 2015 (Agencywide Documents Access and Management System (ADAMS) package Accession No. ML15133A119), as supplemented by letters dated June 25, and July 22, 20115 (ADAMS Accession Nos. ML15195A423 and ML15218A007, respectively), STP Nuclear Operating Company (STPNOC, the licensee) requested relief from the inservice inspection (ISI) interval requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code), for the volumetric examination of the reactor pressure vessel (RPV) cold-leg nozzle dissimilar metal butt welds at the South Texas Project (STP), Unit 1. Portions of the letter dated April 25, 2015, contain sensitive unclassified non-safeguards information (proprietary) and have been withheld from public disclosure pursuant to Section 2.390 of Title 10 of the *Code of Federal Regulations* (10 CFR).

Specifically, pursuant to 10 CFR 50.55a(a)(3)(ii), STPNOC proposed an alternative frequency of the volumetric examination for the cold-leg nozzle butt welds on the basis that complying with the specified requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. The licensee requested relief from the frequency of ISI for dissimilar metal butt welds as required by ASME Code Case N-770-1 "Alternative Examination Requirements and Acceptance Standards for Class 1 PWR Piping and Vessel Nozzle Butt Welds Fabricated With UNS N06082 or UNS W86182 Weld Filler Material With or Without Application of Listed Mitigation Activities, Section XI, Division 1."

2.0 REGULATORY EVALUATION

By *Federal Register* notice dated November 5, 2014 (79 FR 65776), which became effective on December 5, 2014, the paragraph headings in 10 CFR 50.55a were revised. Relief requests that had been previously covered by 10 CFR paragraph 50.55a(a)(3)(i) are now covered under the equivalent 10 CFR paragraph 50.55a(z)(1). Relief requests that had been previously covered by 10 CFR paragraph 50.55a(a)(3)(ii) are now covered under the equivalent 10 CFR paragraph 50.55a(z)(2).

The licensee requested relief from the frequency of ISIs for the RPV inlet cold-leg nozzle dissimilar metal butt welds as required by ASME Code Case N-770-1. For augmented ISI requirements for Class 1 piping and nozzle dissimilar-metal butt welds, 10 CFR 50.55a(g)(6)(ii)(F) states that the examination requirements for licensees of existing, operating pressurized-water reactors as of July 21, 2011, must implement the requirements of ASME Code Case N-770-1, subject to the conditions specified in paragraphs (g)(6)(ii)(F)(2) through (10) of 10 CFR 50.55a, by the first refueling outage after August 22, 2011.

Pursuant to 10 CFR 50.55a(g)(4), the ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except the design and access provisions and the preservice examination requirements, set forth in the ASME Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of 10 CFR 50.55a(g) may be used when authorized by the Director, Office of Nuclear Reactor Regulation. A proposed alternative must be submitted and authorized prior to implementation. The licensee must demonstrate (1) the proposed alternative would provide an acceptable level of quality and safety; or (2) compliance with the specified requirements of this section would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Based on the above, and subject to the following technical evaluation, the U.S. Nuclear Regulatory Commission (NRC) staff finds that regulatory authority exists for the licensee to request and the NRC to authorize the alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Component Affected

The components affected are ASME Code Class 1 RPV cold-leg nozzle to safe-end dissimilar metal butt welds. In accordance with ASME Code Case N-770-1, the licensee classified the components under Inspection Item B. The affected components, listed in Table of the licensee's letter dated April 24, 2015, are:

1. RPV1-N2ASE, Safe End to RPV Loop A Inlet Nozzle;
2. RPV1-N2BSE, Safe End to RPV Loop B Inlet Nozzle;
3. RPV1-N2CSE, Safe End to RPV Loop C Inlet Nozzle; and
4. RPV1-N2DSE, Safe End to RPV Loop D Inlet Nozzle.

The licensee stated that these welds are made of nickel-based alloys, specifically Alloy 600, that are known to be susceptible to primary-water stress-corrosion cracking (PWSCC). Each of the welds is exposed to a cold-leg temperature of 563 degrees Fahrenheit (°F) and pressure of 2250 pounds per square inch absolute (psia) during normal plant operation. The geometrical dimensions of these welds are: 33.05 inches outside diameter (OD), 27.47 inches inside diameter (ID), and 2.79 inches wall thickness.

3.2 Applicable Code Edition and Addenda

The Code of record for the third 10-year ISI interval is the 2004 Edition and no addenda of the ASME Code.

3.3 Duration of Relief Request

The licensee submitted this relief request for the third 10-year ISI interval, which commenced on September 25, 2010, and will end on September 24, 2020.

3.4 ASME Code Requirement

ASME Code Case N-770-1, Table 1, Inspection Item B, requires that the unmitigated butt welds at cold-leg temperatures greater than or equal to 525 °F and less than 580 °F are volumetrically examined every second inspection period not to exceed 7 years.

3.5 Proposed Alternative

The licensee's next volumetric examination is required during the fall 2015 (1RE19) refueling outage. The licensee's proposed alternative frequency of volumetric examinations is a one-time, one operating cycle (approximately 18 months) extension to the required examination frequency in accordance with ASME Code Case N-770-1, Table 1, Inspection Item B. The licensee proposes to perform the volumetric examination in accordance with ASME Code Case N-770-1, Table 1, Inspection Item B, in the spring 2017 (1RE20) refueling outage. The licensee requests a one-time extension to perform the volumetric examinations once every five refueling outages (6.7 effective full power years (EFPY)) not to exceed 8 years, instead of every second inspection period not to exceed 7 years as required by ASME Code Case N-770-1.

3.6 Basis for Hardship

The licensee stated that the ASME Code Case N-770-1 required volumetric examination of the subject welds from the OD surface of the pipes is not possible due to limited access to the welds from the OD. The licensee also stated that in order to perform the Code Case N-770-1 required ultrasonic test (UT) from the ID surface of the subject welds, removal of the RPV lower internal (core barrel) assembly is necessary. The proposed alternative permits the volumetric examination of the welds under consideration to take place during the same refueling outage that the licensee plans to mitigate these welds by the non-welded stress improvement process. The alignment of activities requiring a core barrel lift in a single refueling outage reduces unnecessary personnel radiation and safety hazard exposure, and minimizes potential damage to the assembly.

The licensee stated that removing the core barrel assembly is considered a "critical lift" due to the weight of the component, tight clearances, and the radiation emitted by the assembly. This critical lift would pose personnel safety hazards and potential damage to the assembly. For removal, the core barrel assembly is raised above the refueling cavity water level during transfer from the reactor vessel to the storage stand location (the lower internal storage area (LISA)). The licensee estimated the radiation dose for the actual work activities (e.g., removing, transferring, and installing the core barrel assembly) during 2RE14 refueling outage as

123 milli-roentgen-equivalent-man (mrem). During the 13 days that the core barrel assembly was stored in the LISA, workers could receive additional dose of approximately 487.5 mrem. The total dose associated with core barrel assembly removal activities was estimated by the licensee as 610.5 mrem.

3.8 Basis for Use

The licensee stated that during the 1999 (1RE08) refueling outage, it preformed UT of the subject RPV cold-leg nozzle welds without recordable indications.

The licensee stated that in the fall 2009 (1RE15) refueling outage, it performed the UT of the subject RPV cold-leg welds from the ID surface using a mechanized and encoded technique without recordable indications. The UT was qualified in accordance with Appendix VIII, Section XI of the ASME Code. The licensee also performed a supplemental surface examination of the subject welds using the eddy current testing (ET), and did not identify any recordable ID surface connected indications.

By letter dated June 25, 2015, in response to an NRC staff request for additional information (RAI) dated June 18, 2015 (ADAMS Accession No. ML15170A066), the licensee stated that the examinations of the subject RPV cold-leg welds performed in the fall 2009 (1RE15) refueling outage were credited toward the ASME Code Case N-770-1 baseline examination required by 10 CFR 50.55a(g)(6)(ii)(F)(1). Additionally, the licensee stated that in April 2014, it performed the UT and ET of the Unit 1, RPV hot-leg welds, and did not find indications.

The licensee stated that the ET used for surface inspection of the RPV welds was qualified in accordance with the same qualification procedure and practical trials that were discussed in the Joseph M. Farley Nuclear Plant (Farley) August 1, 2014, letter (ADAMS Accession No. ML14213A484). It was noted in the Farley qualification procedure and practical trials documents that the ET is capable of detecting fatigue, intergranular stress corrosion, and interdendritic stress corrosion cracks as small as 0.04-inch (1 mm) deep by 0.24-inch (6 mm) long. As documented in the NRC staff's safety evaluation dated December 5, 2014 (ADAMS Accession No. ML14262A317), the Farley ET qualification procedure and practical trials were found acceptable.

Operating experience associated with PWSCC of Alloy 82/182 welds has shown that weld repairs performed during original plant construction are a significant contributor in the initiation and propagation of cracking. The licensee stated that a review of the construction records and a weld repair search performed for Unit 1 RPV Alloy 82/182 welds did not identify any weld repairs performed on the subject RPV cold-leg nozzle dissimilar metal welds during original plant construction.

To support a one-time extension to the ASME Code Case N-770-1 examination requirements for the RPV cold-leg dissimilar metal butt welds, the licensee provided Electric Power Research Institute (EPRI) Technical Report Materials Reliability Program (MRP)-349, "PWR Reactor Coolant System Cold-Loop Dissimilar Metal Butt Weld Reexamination Interval Extension," August 2012,[1] as a technical basis. The EPRI developed a generic technical basis document

[1] Publicly available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001025852.

(i.e., MRP-349) by compiling all existing flaw analyses performed to date on Alloy 82/182 welds to support extension of the reexamination interval for the RPV cold-leg dissimilar metal welds. The results of these generic flaw analyses for an assumed circumferential flaw in the RPV inlet nozzle dissimilar metal welds are shown in Figure 5-4 of MRP-349. Figure 5-4 showed that an assumed initial ID surface connected 10 percent through-wall circumferential flaw in the RPV cold-leg nozzle dissimilar metal weld would not grow to the ASME Code maximum allowable 75 percent through wall flaw in less than 10 years of continued operation. It is noted in the MRP-349 technical basis document that the results provided in Figure 5-4 are not representative of any single plant in the Westinghouse PWR fleet, rather they are based on the limiting thickness in the Westinghouse PWR fleet combined with limiting piping loads. Therefore, the results are conservative. The analysis underlying assumptions in MRP-349 were a 25 percent ID weld repair, a postulated initial ID surface connected circumferential flaw of 10 percent through-wall thickness, a short and a long stainless steel safe end, and the cold-leg operating temperature as high as 565 °F and as low as 535 °F. Therefore, the underlying assumptions are limiting.

In addition, the licensee stated that it performed a site-specific flaw analysis to determine the maximum allowable initial axial and circumferential flaw sizes in the subject RPV cold-leg dissimilar metal welds that would be left in service to support continued operation until the spring 2017 (1RE20) refueling outage (i.e., one operating cycle extension, which is approximately 18 months extension from the fall 2015 (1RE19) refueling outage).

In its analysis, the licensee followed the guidance in EPRI's MRP-287, "Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance," December 2010.[2] From the finite element analysis based on the STP, Unit 1, RPV inlet nozzle dissimilar metal welds specific configuration, the licensee computed the welding residual stresses used in the PWSCC flaw growth analysis. The licensee's site-specific flaw analysis is documented in Attachment 1 of its April 24, 2015, relief request.

From its analysis, the licensee determined the maximum allowable initial flaw size for a given plant operation duration by subtracting the PWSCC crack growth for that plant operation duration from the maximum allowable end-of-evaluation period flaw size (i.e., ASME Code required 75 percent through-wall thickness). For the time interval between the fall 2009 and spring 2017 refueling outage, the licensee estimated the plant to operate at full power for 6.7 EFPY, as shown in Table 6-1 in Attachment 1 of the licensee's letter dated April 24, 2015. In its flaw analysis, the licensee calculated the PWSCC crack growth based on the normal operating temperature, the crack tip stress intensity factors resulting from the normal operating steady state piping loads, and the welding residual stresses for the RPV cold-leg nozzle dissimilar metal welds. For calculating welding residual stresses distributions in the axial and hoop directions, the licensee conservatively assumed the ID weld repairs of 50 percent through the dissimilar metal weld thickness.

The results of the licensee's plant-specific flaw analysis are shown in Figures 7-1 and 7-2 in Attachment 1 of the licensee's letter dated April 24, 2015. The licensee showed in Figure 7-1 that the maximum allowable initial ID surface connected 6.4 percent (0.178 inch) through-wall axial flaw in the subject RPV cold-leg nozzle dissimilar metal weld would not grow to the ASME

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[2] Publicly available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001021023.

Code allowable 75 percent through-wall flaw in 6.7 EFPY. The licensee showed in Figure 7-2 that the maximum allowable initial ID surface connected 37.6 percent (1.05 inches) through-wall circumferential flaw in the subject RPV cold-leg nozzle dissimilar metal weld would not grow to the ASME Code allowable 75 percent through-wall flaw in 6.7 EFPY.

The licensee stated that its plant-specific welding residual stresses calculation and flaw analysis for the axial and circumferential flaws conservatively assumed a 50 percent weld repair even though the review of the construction records and fabrication weld repair search indicated that no weld repairs were performed from the ID surface of any of the four RPV cold-leg nozzle dissimilar metal. In its analysis, the licensee stated that it followed the MRP-287 guidance, which requires plant-specific inputs (e.g., as-built safe-end length that bounds the Unit 1, RPV inlet nozzle's safe ends, and a plant-specific operating temperature that bounds the Unit 1, RPV inlet nozzle's operating temperature).

The licensee stated that Table 7-1, Attachment 1 of its letter dated April 24, 2015, summaries its flaw analysis results and shows that the largest axial and circumferential flaw sizes (depth and length) that could remain in service and remain acceptable from the fall 2009 to spring 2017 refueling outage. The maximum allowable initial circumferential flaw depth of 37.6 percent (1.049 inches) through-wall shown in Table 7-1 is greater than the minimum required detectable and sizable flaw depth (i.e., Supplement 10 of Appendix VIII, the ASME Code, Section XI), and thus would have been reasonably detected at the previous inspection of the subject dissimilar metal welds. Furthermore, the maximum allowable initial axial flaw dimensions (i.e., 0.178-inch deep and 0.356-inch long) shown in Table 7-1 are greater than the smallest flaw dimensions detectable by the STP qualified ET procedure, and thus would have been reasonably detected at the previous inspection of the subject dissimilar metal welds.

3.9 NRC Staff Evaluation

The NRC staff evaluated this relief request pursuant to 10 CFR 50.55a(z)(2). The NRC staff considered whether compliance with the specified requirements of 10 CFR 50.55a(g), or portions thereof, would result in hardship or unusual difficulty, and if there is a compensating increase in the level of quality and safety despite the hardship.

Licensee Hardship

The NRC staff concludes that requiring the licensee to comply with frequency of ISI specified in ASME Code Case N-770-1, as required by 10 CFR 50.55a(g)(6)(ii)(F) with conditions, would result in hardship. The basis for the hardship is that the licensee has to remove the RPV core barrel assembly to get access to the ID surface of the RPV cold-leg nozzle to safe-end dissimilar metal butt welds to conduct the required volumetric examination of these welds. The core barrel assembly is heavy, highly radioactive, and movement involves tight clearances between components. Additionally, removal of core barrel assembly from the reactor vessel could cause damage to the assembly, as well as posing safety and additional radiation exposure hazards to personnel involved. Therefore, concerns from damaging the core barrel assembly during removal activities, safety hazards created by the heavy lift operation, and unnecessary personnel exposure resulting from multiple core barrel lift operations (violating as low as reasonably achievable (ALARA) principles) constitutes hardship.

Weld Residual Stress Calculation and Flaw Tolerance Analysis

In evaluating the licensee's proposed alternative (i.e., deferral of the ISI of the RPV cold-leg nozzle dissimilar metal welds from the fall 2015 (1RE19) to spring 2017 (1RE20) refueling outage), the NRC staff assessed whether it appeared that the licensee used appropriate industry guidance, assumptions, and inputs as applied to the PWSCC type flaws when performed plant-specific welding residual stresses calculations and flaw analysis.

In reviewing the license's stress analysis, the NRC staff concludes that the licensee followed the recommendations specified in MRP-287 to determine the welding residual stress distributions through the thickness of the dissimilar metal weld for a PWSCC flaw evaluation. The NRC staff concludes:

- The licensee assumed a 50 percent deep ID surface weld repair with a repair length of 360 degrees around the circumference in the Alloy 82/182 weld to compute welding residual stress distributions in the axial and hoop directions. The welding residual stress computations involved finite element modeling to simulate the steps of the fabrication sequences of the welding process as in the plant-specific drawings. The NRC staff notes that the licensee's search of the construction and fabrication weld repair records showed no weld repairs performed from the ID of any of the subject four RPV cold-leg nozzle dissimilar metal welds. Therefore, the NRC staff concludes that a 50 percent deep ID repair is an adequate assumption and consistent with the recommendations in MRP-287.

- The licensee used the limiting residual stresses from the stresses obtained at three different paths within the dissimilar metal weld and, therefore, is adequate as well as consistent with the recommendations in MRP-287.

- The welding residual stress profiles provided in the relief request were consistent with the profiles in similar weld geometries, fabrication methods, and fabrication history. The NRC staff notes that construction repairs have generally been present when PWSCC has been observed and, therefore, an adequate repair assumption is recommended by the MRP-287 in the evaluation process.

In reviewing the licensee's plant-specific PWSCC flaw evaluation, the NRC staff determined that the licensee followed the recommendations specified in MRP-287 to perform the flaw evaluation. The NRC staff concludes:

- The licensee used the evaluation guidelines and procedures described in ASME Code, IWB-3640, "Evaluation of Flaws in Austenitic Steel Piping," and Appendix C of Section XI to calculate the maximum allowable end of evaluation period axial and circumferential flaw sizes (as shown in Table 5-1, Attachment 1 to relief request). The NRC staff concludes that for the PWSCC flaw evaluation of Alloy 182/82 materials, use of the elastic plastic fracture mechanics approach and the maximum allowable flaw depth limit of 75 percent of the wall thickness is an adequate approach.

- The licensee used MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," November 2004,[3] that provides a crack growth rate for the nickel-based alloy weld materials to calculate the growth of the ID surface flaws due to PWSCC. The NRC staff concludes that MRP-115 is the source of PWSCC growth laws for Alloy 82/182 weld metals and generally accepted by the industry, thus is adequate for this analysis.

- The licensee assumed an aspect ratio (flaw's length/depth) of 2 for an axial flaw for the hoop stresses and an aspect ratio of 10 for a circumferential flaw for the axial stresses. The NRC staff notes that the growth of an axial flaw is limited to the width of the dissimilar metal weld and the growth of a circumferential flaw is rapid for the larger aspect ratio. The NRC staff concludes that this assumption is adequate and consistent with the recommendations in MRP-115.

- The licensee represented the calculated through-wall residual stress distributions by a fourth order polynomial curve fitting. The NRC staff concludes that representing the complex through the wall residual stress field by a fourth order polynomial approximation is reasonably adequate for this analysis.

- The licensee postulated a 10 percent deep ID surface connected initial circumferential flaw and a 2.87 percent (0.08-inch) deep ID surface connected initial axial flaw. The NRC staff notes that the 10 percent ID surface connected through-wall flaw was generally assumed in the flaw analysis due to the UT uncertainty that all flaws with depth less than 10 percent through-wall thickness would not be reliably detected and recorded by UT. Given the licensee's ET qualification process (i.e., as described in the Farley August 1, 2014, letter, and accepted by the NRC in the December 5, 2014, NRC safety evaluation), the NRC staff concludes that the licensee's bounding initial flaw size assumption of 2.87 percent through wall is adequate.

The NRC staff reviewed Figures 7-1 and 7-2 of Attachment 1 to the relief request, and the NRC staff noted that an assumed initial surface connected 2.87 percent deep axial or circumferential flaw in the subject RPV cold-leg nozzle dissimilar metal welds would not grow to a maximum allowable 75 percent deep flaw in five refueling outages (period from the fall 2015 (1RE19) refueling outage to the spring 2017 (1RE20) refueling outage) or 6.7 EFPY.

<u>Confirmatory Independent Flaw Evaluation</u>

The NRC staff performed an independent flaw evaluation to verify the licensee's analysis. The NRC staff's independent flaw analyses determined the maximum flaw depth, leak, and rupture characteristics of the subject welds to a postulated initial ID surface connected (circumferential or axial) flaw. The analyses were based on the requirements of the ASME Code, Section XI, IWB-3640, and an assumed postulated initial flaw due to PWSCC. The NRC staff used the STP, Unit 1, welding residual stress distributions provided by the licensee. The welding residual stress profiles for the axial and the hoop direction were curve fit by a fourth order polynomial

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[3] Publicly available at http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001006696.

approximation. For the PWSCC crack growth, the NRC staff used the 75th percentile crack growth rate data for Alloy 182.

The NRC staff based its assessment of the licensee's proposed alternative on the time for an assumed 2.5 percent (0.08-inch) deep initial surface connected (circumferential or axial) crack to grow to the ASME Code allowable crack depth of 75 percent through wall thickness. The NRC staff determined that the licensee's assumption is reasonable since it is realistic to expect that the ID surface connected flaw of 2.5 percent deep in the subject dissimilar metal welds would be detected by the qualified ET performed in conjunction with the UT before the flaw reaches the allowable crack depth of 75 percent through wall. Based on its independent flaw evaluation and the licensee's flaw analysis, the NRC staff determines there is sufficient safety margin between the inspection frequency and the time for the postulated flaw to reach the 75 percent allowable flaw depth to conclude that inspecting these welds once every five refueling outages or 6.7 EFPY would provide reasonable assurance of structural integrity and leak tightness of the dissimilar metal welds.

Therefore, given the flaw evaluation demonstrating sufficient safety margins, the ID surface examinations by a qualified ET demonstrating reasonable assurance of no surface connected flaws, and the volumetric examinations by UT, the NRC staff concludes that the licensee has provided adequate technical basis to demonstrate that its proposed alternative examination frequency would provide reasonable assurance of structural integrity and leak tightness of the RPV cold-leg nozzle dissimilar metal butt welds.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the proposed alternative provides reasonable assurance of structural integrity and leak tightness of the RPV cold-leg nozzle dissimilar metal butt welds. The NRC staff concludes that complying with the specified ASME Code requirement would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that the licensee has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes the one-time use of the proposed alternative for one operating cycle extension, and that is, up to and including the refueling outage in the spring of 2017 at STP, Unit 1, in the third 10-year ISI interval, which commenced on September 25, 2010, and will end on September 24, 2020.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

Principal Contributor: A. Rezai, NRR

Date: August 21, 2015

hazards resulting from two separate core barrel lift operations. Thus, complying with ASME Code would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety. Accordingly, the NRC staff concludes that you have adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(2). Therefore, the NRC staff authorizes relief request RR-ENG-3-17 at STP Unit 1, for the third 10-year ISI interval which began on September 25, 2010, and ends on September 24, 2020.

All other ASME Code, Section XI, requirements for which relief was not specifically requested and approved in the subject request remain applicable, including third-party review by the Authorized Nuclear Inservice Inspector.

The detailed results of the NRC staff review are provided in the enclosed safety evaluation. If you have any questions concerning this matter, please contact Ms. Lisa Regner of my staff at 301-415-1906 or via e-mail at Lisa.Regner@nrc.gov.

Sincerely,

/RA/

Michael T. Markley, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket Nos. 50-498

Enclosure:
Safety Evaluation

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