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

June 30, 2016

Mr. G. T. Powell
Executive Vice President and CNO
STP Nuclear Operating Company
South Texas Project
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNIT 2 - REQUEST FOR RELIEF NO. RR-ENG-3-20 FOR EXTENSION OF THE INSPECTION FREQUENCY OF THE REACTOR VESSEL COLD-LEG NOZZLE TO SAFE-END WELDS WITH FLAW ANALYSIS (CAC NO. MF7428)

Dear Mr. Powell:

By letter dated March 3, 2016, 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 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(2), 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."

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, two-operating-cycle extension to the end of the Unit 2 fall 2019 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-3-20 at STP Unit 2, for the third 10-year ISI interval, which began on October 19, 2010, and ends on October 18, 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,

Balwant K. Singal for

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-499

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELIEF REQUEST RR-ENG-3-20 REGARDING DEFERRAL OF INSERVICE INSPECTION OF

REACTOR PRESSURE VESSEL COLD-LEG NOZZLE DISSIMILAR METAL BUTT WELDS

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT 2

DOCKET NO. 50-499

1.0 INTRODUCTION

By letter dated March 3, 2016 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML16076A319), STP Nuclear Operating Company (STPNOC, the licensee) requested relief from requirements of the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) specifically related to the frequency of inspection of the reactor pressure vessel (RPV) cold-leg nozzle dissimilar metal (DM) butt welds at the South Texas Project (STP), Unit 2. Portions of the STPNOC letter 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 Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(z)(2), the licensee proposed an alternative frequency of volumetric examination for the RPV cold-leg nozzle DM 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.

2.0 REGULATORY EVALUATION

The licensee requested relief from the frequency of inservice inspections (ISIs) for the RPV inlet cold-leg nozzle DM butt welds as required by ASME Code Case N-770-1. For augmented ISI requirements for Class 1 piping and nozzle DM butt welds, 10 CFR 50.55a(g)(6)(ii)(F) states that the examination requirements for licensees of existing, operating pressurized-water reactors (PWRs) 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(g)(4)(ii), inservice examination of components during successive 120-month inspection intervals must comply with the requirements of the latest edition and addenda of the Code incorporated by reference in paragraph (a) of 50.55a, 12 months before the start of the 120-month inspection interval (or the optional ASME Code Cases listed in NRC Regulatory Guide 1.147, Revision 17, "Inservice Inspection Code Case Acceptability, ASME Section XI, Division 1," August 2014 (ADAMS Accession No. ML13339A689), when using Section XI, that is incorporated by reference in paragraph (a)(3)(ii) of 50.55a), subject to the conditions listed in paragraph (b) of 50.55a.

Pursuant to 10 CFR 50.55a(z), alternatives to the requirements of paragraph (g) of 10 CFR 50.55a 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 Components Affected

The components affected are ASME Code Class 1 RPV cold-leg nozzle to safe-end DM butt welds. These welds are classified as Inspection Item B in accordance with ASME Code Case N-770-1 (Table 1 of the relief request RR-ENG-3-20).

The four RPV cold-leg nozzle to safe-end DM butt welds at STP, Unit 2 (Table 1 of the relief request RR-ENG-3-20), are made of nickel-based alloys that are known to be susceptible to primary water stress-corrosion cracking (PWSCC). Each of the welds is exposed to the 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 October 19, 2010, and is scheduled to end on October 18, 2020.

3.4 ASME Code Requirement

The ASME Code requirements applicable to this request originate in Section XI, Table IWB-2500-1. In accordance with 10 CFR 50.55a(g)(6)(ii)(F), the NRC has mandated an augmented inspection for the welds in this request, and that is to implement the requirements of ASME Code Case N-770-1 with conditions.

In accordance with N-770-1, Table 1, Inspection Item B, the welds under this request are required to be volumetrically examined every second inspection period not to exceed 7 years. The next inspection is expected to be held in the fall 2016 (2RE18) refueling outage.

3.5 Proposed Alternative

The licensee proposed an alternative frequency of inspection for the welds in this relief request. The proposed alternative is a one-time, two-operating-cycle extension (approximately 36 months) to the volumetric examination required by ASME Code Case N-770-1. Essentially, the licensee proposed to perform the volumetric examination of the welds in this relief request in the fall 2019 (2RE20) refueling outage instead of the fall 2016 (2RE18) refueling outage as required by N-770-1.

3.6 Basis for Hardship

The licensee stated that in order to perform the N-770-1 required ultrasonic testing (UT) from the ID surface of the subject welds, removal of the RPV lower internal core barrel assembly is necessary. Removing the core barrel assembly is considered a critical lift due to weight of the component, tight clearances, and the radiation emitted by the assembly. Critical lifts 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 radiation dose for the actual work activities (e.g., removing, transferring, and installing the core barrel assembly) performed during the 2RE14 refueling outage in the spring 2010 was estimated to be 123 milli-roentgen-equivalent-man (mrem). The increase in dose rates in the general area walkway was estimated as 1.3 mrem/hour. Dose rates were taken with the core barrel present on the stand as compared to dose rates without the core barrel. During the 13 days that the core barrel assembly was stored in the stand, the licensee stated that workers could receive an additional dose of 487.5 mrem. In total, the radiation dose associated with core barrel assembly removal activities was estimated as 610.5 mrem. Therefore, STPNOC considered the personnel and plant safety concerns and radiological dose considerations represent a hardship and unusual difficulty.

The licensee stated that it plans to mitigate the RPV cold- and hot-leg nozzle to safe-end welds by a non-welded stress improvement process in the refueling outage (2RE20) in the fall 2019. The proposed alternative permits both the ISI of the welds in this relief request and the mitigation activities of the RPV cold- and hot-leg nozzle to safe-end welds to take place during the same refueling outage (2RE20) in the fall 2019. The alignment of two activities in a single refueling outage limits removal of the RPV core barrel assembly into a one-time event thereby reducing unnecessary radiation exposure and safety hazards to personnel, and minimizes potential damage to the assembly.

The licensee also stated that the N-770-1 required volumetric examination from the OD surface of the subject welds is not possible due to limited access. The access limitations also represent a hardship and unusual difficulty.

3.7 Basis for Use

The licensee stated that in the spring 2010 (2RE14) refueling outage, it performed the UT of the subject RPV cold-leg nozzle DM welds, and did not find unacceptable indications. The licensee performed the UT from the ID surface using the mechanized and encoded technique. The UT was qualified in accordance with Supplement 10 to Appendix VIII, Section XI of the ASME Code. These volumetric examinations were credited toward the ASME Code Case N-770-1 baseline examination required by 10 CFR 50.55a(g)(6)(ii)(F)(1). In the same refueling outage (spring 2010 (2RE14)), the licensee also performed a supplemental surface examination of the welds in this relief request using the eddy current testing (ET), and did not identify unacceptable ID surface-connected indications.

The licensee stated that in May 2015, it performed the UT and ET of the STP, Unit 2, RPV hot-leg DM welds, and did not find unacceptable indications. The absence of indications in the hot-leg DM welds in 2015 provides added assurance that the proposed one-time extension of the inspection frequency by approximately 36 months for the RPV cold-leg DM welds is acceptable and provides reasonable assurance of structural integrity.

The licensee stated that the ET used for the surface inspection of the STP, Unit 2, RPV DM welds was qualified in accordance with the same qualification procedure and practical trials that were discussed in the Joseph M. Farley Nuclear Plant August 1, 2014, letter (ADAMS Accession No. ML14213A484). As documented in the December 5, 2014, NRC safety evaluation (ADAMS Accession No. ML14262A317), the NRC staff found the Farley ET qualification procedure and practical trials acceptable. It was noted in the Farley letter and the NRC staff's safety evaluation that the ET is capable of detecting fatigue and intergranular stress-corrosion cracking (IGSCC) / interdendritic stress-corrosion cracking (IDSCC) cracks as small as 0.04 inches (1 mm) deep by 0.24 inches (6 mm) long. In this relief request, the licensee stated that the STP ET inspection procedure implemented in the spring 2010 (2RE14) refueling outage required that an indication as small as 0.08 inches (2 mm) deep by 0.28 inches (7.1 mm) long be recorded.

Operating experience regarding the PWSCC of Alloy 82/182 welds has shown that weld repairs performed during the original plant construction are a significant contributor in the initiation and propagation of PWSCC. The licensee stated that a review of the construction records including the weld repair documents did not identify significant weld repairs performed on the subject RPV cold-leg nozzle DM welds during the original plant construction.

The licensee provided 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 to support a one-time extension to the ASME Code Case N-770-1 examination requirements for the subject RPV cold-leg DM butt welds.

The Electric Power Research Institute (EPRI) developed MRP-349, a generic technical basis document, 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 DM welds. The results provided in Figure 5-4 of MRP-349 show that an assumed initial ID surface connected 10 percent through wall circumferential flaw in the RPV cold leg nozzle DM weld would not grow to the ASME Code allowable 75 percent through wall flaw in less than 10 years of continued operation. It is noted in the MRP-349 that the results provided in Figure 5-4 are not representative of any single plant in the Westinghouse pressurized water reactor (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. It is also noted that the analysis underlying assumptions in MRP-349 were a 25 percent ID weld repair, a 10 percent deep initial ID surface connected circumferential flaw, 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.

The licensee performed a plant-specific flaw analysis to demonstrate that the allowable axial and circumferential flaws that could be left in service would not grow to the ASME Code allowable flaw size between the planned examinations (i.e., between the spring 2010 and fall 2019 refueling outages). Between the spring 2010 and the fall 2019 refueling outages, the plant is estimated to operate at full power for 8.3 effective full power years (EFPY). In its analysis, the licensee followed the guidance of MRP-287 "Primary Water Stress Corrosion Cracking (PWSCC) Flaw Evaluation Guidance" which requires plant-specific inputs (e.g., as-built safe end length that bounds the STP, Unit 2, RPV inlet nozzles' safe ends, and a plant specific operating temperature that bounds the STP, Unit 2, RPV inlet nozzles' operating temperature).

The licensee stated that the welding residual stresses (WRS) were computed using the finite element analysis (FEA) of the South Texas, Unit 2, RPV inlet nozzle DM welds. The licensee conservatively assumed the ID weld repairs of 50 percent through the DM weld thickness in calculating the WRS distributions in the axial and hoop directions even though the review of the construction records and fabrication weld repair search indicated that no significant weld repairs were performed from the ID surface of any of the four RPV cold leg nozzle DM welds. The PWSCC crack growth was determined using the normal operating temperature, the crack tip stress intensity factors resulting from the normal operating steady state piping loads, and the plant-specific WRS profiles. Details of the licensee's plant-specific flaw analysis are documented in Attachment 1 of this relief request, titled "LTR-PAFM-16-11-NP, Revision 0, Technical Justification to Support Extended Volumetric Examination Interval for South Texas Unit 2 Reactor Vessel Inlet Nozzle to Safe End Dissimilar Metal Welds" (non-proprietary version at Accession No. ML16076A319).

In Figures 7-1 and 7-2 of this relief request, the licensee showed that the allowable axial flaw (3.5 percent depth or 0.098 inch) and allowable circumferential flaw (32.5 percent depth or

[1] Electric Power Research Institute, "Materials Reliability Program: PWR Reactor Coolant System Cold-Loop Dissimilar Metal Butt Weld Reexamination Interval Extension [MRP-349]," August 2012, publicly available at: http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001025852.

0.907 inch) in the subject RPV cold leg nozzle DM welds would not grow to the ASME Code allowable 75 percent flaw in less than 8.3 EFPY. Given the capabilities of qualified UT and ET, the above allowable flaws would have been detected in the last inspection in the spring 2010.

3.8 NRC Staff Evaluation

The NRC staff has evaluated this relief request pursuant to 10 CFR 50.55a(z)(2). The NRC staff focused on 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 was a compensating increase in the level of quality and safety despite the hardship.

Hardship

The NRC staff found that requiring the licensee to comply with the frequency of ISI specified in ASME Code Case N-770-1, as mandated by 10 CFR 50.55a(g)(6)(ii)(F) with conditions, would result in hardship without an increase in the level of quality and safety. The basis for the hardship is as follows. The licensee would need to remove the RPV core barrel assembly to access the ID surface of the welds to conduct the required volumetric examination of the RPV cold-leg nozzle to safe-end DM butt welds. The core barrel assembly is heavy and highly radioactive and there are tight clearances between components; therefore, removal of the core barrel assembly from the reactor vessel would cause potential damage to the assembly and pose safety hazards and additional radiation exposure to personnel involved. Thus, concerns associated with (1) damaging the core barrel assembly during removal activities, (2) the safety hazards created by a heavy lift, and (3) as low as reasonably possible (ALARA), constitute hardship for the licensee.

In addition, the UT from the OD surface of the RPV cold-leg nozzle to safe-end DM butt welds is not possible because the geometry of the welds limits access to the OD. Therefore, access limitations also create a hardship and unusual difficulty.

Proposed Alternative and Assurance of Structural Integrity and Leak Tightness

As a technical basis to demonstrate assurance of structural integrity and leak tightness of the subject RPV cold-leg nozzle DM welds while operating for additional two operating cycles, the licensee provided the results of a plant-specific flaw analysis performed in accordance with IWB-3600 of Section XI. The NRC staff's evaluation of technical sufficiency of the licensee's plant-specific flaw analysis are discussed below.

Plant-Specific Flaw Analysis

The licensee's proposed alternative involves a one-time, two-operating cycle extension of the volumetric examination frequency contained in ASME Code Case N-770-1 from every second inspection period not to exceed 7 years to 9.5 calendar years which is approximately 8.3 EFPY. During its review of the alternative, 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 STPNOC performed the plant-specific WRS calculations and flaw tolerance analysis.

Weld Residual Stress Calculation

In reviewing the licensee's stress analysis, the NRC staff found that the licensee followed the recommendations specified in MRP-287 to determine the WRS distributions through the thickness of the DM weld for a PWSCC flaw evaluation. Of note,

- 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 WRS distributions in the axial and hoop directions. The WRS computations involved finite element modeling to simulate steps of the fabrication sequences of the welding process as were in the plant-specific drawings. The NRC staff noted that the licensee's search of the construction and fabrication weld repair records showed no significant weld repairs performed from the ID of any of the subject four RPV cold-leg nozzle DM welds. Therefore, the NRC staff finds that a 50 percent deep ID repair is an adequate assumption, and consistent with the recommendations in MRP-287;

- The NRC staff finds that the licensee used the limiting residual stresses from the stresses obtained at three different paths within the DM weld, therefore, this assumption is adequate as well as consistent with the recommendations in MRP-287.

- The NRC staff finds that the WRS profiles provided in proprietary Figure 4-2, Attachment 1 of the STPNOC relief request, were consistent with profiles in similar weld geometries, fabrication methods, and fabrication history. The NRC staff noted that, based on operating experience, construction repairs have generally been performed for those welds that have experienced PWSCC. Therefore, the licensee has included an adequate repair assumption in its weld residual stress analysis as recommended by MRP-287.

Flaw Tolerance Analysis

In reviewing the licensee's plant-specific PWSCC flaw evaluation, the NRC staff found that the licensee followed the recommendations specified in MRP-287 to perform flaw evaluation. Of note,

- The licensee used the evaluation guidelines and procedures described in IWB-3640 and Appendix C of Section XI to calculate the maximum allowable axial and circumferential flaw sizes at the end of the evaluation period (as shown in Table 5-1, Attachment 1 to the STPNOC relief request). The NRC staff finds 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 the adequate approach and is, therefore, acceptable.

- The licensee used the crack growth rates for the nickel-based alloy weld materials in MRP-115, "Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," November

2004,[2] to calculate growth of the ID surface-connected flaws due to PWSCC. The NRC staff finds that MRP-115 as the source of PWSCC growth laws for the Alloy 82/182 weld metals is acceptable for use, thus, the crack growth analysis is adequate.

- The licensee assumed an aspect ratio (flaw's length/depth) of 2 for axial flaw for the hoop stresses and an aspect ratio of 10 for circumferential flaw for the axial stresses. The NRC staff notes that the growth of axial flaws is limited to the width of the DM weld, and the growth of circumferential flaw is rapid for the larger aspect ratio. The NRC staff finds that the aspect ratios used in the licensee's crack growth analysis are consistent with the recommendations in MRP-115, and are, therefore, adequate.

- The licensee represented the through wall residual stress distributions by a 4th order polynomial. The NRC staff finds that representing the complex residual stress field by a 4th order polynomial approximation is adequate for this analysis.

- The NRC staff finds that the allowable axial flaw would be 3.5 percent of wall thickness (0.098 inch) and the allowable circumferential flaw would be 32.5 percent of the wall thickness (0.907 inch) for the time period between 2010 and 2019. This means that any flaws less than or equal to the allowable flaw size that were left in service in the spring 2010 examination would not grow to the ASME Code allowable flaw size of 75 percent through wall before the 2019 refueling outage. Given that the South Texas ET is qualified to detect flaw size greater than 2.86 percent depth and that no flaws were identified in the spring 2010 examination, the NRC staff finds that the licensee's flaw analysis demonstrates adequate margin, and is, therefore, acceptable.

Confirmatory Independent Flaw Tolerance Evaluation

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 2, WRS distributions provided by the licensee. The WRS profiles for the axial and the hoop direction were curve fit by a fourth order polynomial 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 0.08 inches (2.86 percent) 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 found that the licensee's assumption is reasonable since it is reasonable to expect that the ID surface-connected flaw of 0.08 inches (2.86 percent) deep in the subject DM

[2] Electric Power Research Institute, "Materials Reliability Program: Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds (MRP-115)," November 2004, publicly available at: http://www.epri.com/abstracts/Pages/ProductAbstract.aspx?ProductId=000000000001006696.

welds would be detected by the qualified ET performed in conjunction with the UT before the flaw reached the allowable crack depth of 75 percent through wall. The NRC staff finds sufficient safety margins exist 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 in the fall 2019 (2RE20) refueling outage (9.5 calendar years, which is approximately 8.3 EFPY) will provide reasonable assurance of structural integrity and leak tightness of the DM welds.

Therefore, based on both the licensee's and the NRC staff's crack growth analyses, the NRC staff finds 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 DM 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 DM butt welds. The NRC staff finds 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 in RR-ENG-3-20 for up to and including the fall 2019 refueling outage at STP, Unit 2, in the third 10-year ISI interval, which commenced on October 19, 2010, and is scheduled to end on October 18, 2020.

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

Principal Contributor: A. Rezai, NRR/DE/EPNB

Date: June 30, 2016.

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 BSingal for/

Robert J. Pascarelli, Chief
Plant Licensing Branch IV-1
Division of Operating Reactor Licensing
Office of Nuclear Reactor Regulation

Docket No. 50-499

Enclosure:
Safety Evaluation

cc w/encl: Distribution via Listserv

DISTRIBUTION:
PUBLIC
LPL4-1 Reading
RidsACRS_MailCTR Resource
RidsNrrDeEpnb Resource
RidsNrrDorlLpl4-1 Resource
RidsNrrPMSouthTexas Resource
RidsNrrLAJBurkhardt Resource
RidsRgn4MailCenter Resource
ARezai, NRR/DE/EPNB

ADAMS Accession No. ML16174A091 *via email **per SE dated June 24, 2016 ML16180A006

OFFICE	NRR/DORL/LPL4-1/PM	NRR/DORL/LPL4-1/LA*	NRR/DE/EPNB/BC	NRR/DORL/LPL4-1/BC
NAME	LRegner	JBurkhardt	DAlley**	RPascarelli (BSingal for)
DATE	6/30/16	6/23/16	6/24/16	6/30/16

OFFICIAL RECORD COPY