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

August 13, 2019

Mr. G. T. Powell
President and CEO
STP Nuclear Operating Company
P.O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - RE: REQUEST FOR RELIEF REQUEST RR-ENG-3-23 TO EXTEND VOLUMETRIC EXAMINATION INTERVAL FOR REACTOR VESSEL CLOSURE HEAD NOZZLES (EPID L-2019-LLR-0021)

Dear Mr. Powell:

By letter dated February 28, 2019, STP Nuclear Operating Company (STPNOC, the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, to extend an inservice inspection interval for South Texas Project, Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), the licensee proposed an alternative, RR-ENG-3-23, regarding extending the frequency of volumetric and/or surface examination for the reactor vessel closure head penetration nozzles and their associated attachment partial penetration (J-groove) welds on the basis that the alternative provides an acceptable level of quality and safety.

The U.S. Nuclear Regulatory Commission (NRC) staff has reviewed the subject request and concludes, as set forth in the enclosed safety evaluation, that STPNOC has adequately addressed all of the regulatory requirements set forth in 10 CFR 50.55a(z)(1) and demonstrated that the proposed alternative provides an acceptable level of quality and safety. Therefore, the NRC staff authorizes RR-ENG-3-23 at South Texas Project, Units 1 and 2, until the end of the fourth inservice inspection interval, which is scheduled to end on August 20, 2027, for Unit 1, and December 15, 2028, for Unit 2.

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

If you have any questions, please contact the Project Manager, Ed Miller at 301-415-2481 or via e-mail at Ed.Miller@nrc.gov.

Sincerely,

/RA/

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

Docket Nos. 50-498 and 50-499

Enclosure:
Safety Evaluation

cc: Listserv

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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF RR-ENG-3-23 TO EXTEND VOLUMETRIC EXAMINATION

INTERVAL FOR REACTOR VESSEL CLOSURE HEAD NOZZLES

STP NUCLEAR OPERATING COMPANY

SOUTH TEXAS PROJECT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated February 28, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19059A469), South Texas Project Nuclear Operating Company (the licensee) requested relief from the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel (ASME Code) Code Case N-729-4, “Alternative Examination Requirements for PWR [Pressurized-Water Reactor] Reactor Vessel Upper Heads With Nozzles Having Pressure-Retaining Partial-Penetration Welds Section XI, Division 1,” at South Texas Project (STP), Units 1 and 2.

Specifically, pursuant to Title 10 of the *Code of Federal Regulations* (10 CFR) paragraph 50.55a(z)(1), “Acceptable level of quality and safety,” the licensee proposed an alternative, RR-ENG-3-23, regarding extending the frequency of volumetric and/or surface examination for the reactor vessel closure head (RVCH) penetration nozzles and their associated attachment partial penetration (J-groove) welds on the basis that the alternative provides an acceptable level of quality and safety.

2.0 REGULATORY EVALUATION

Pursuant to 10 CFR 50.55a(g)(4), “Inservice inspection standards requirement for operating plants,” throughout the service life of a boiling or pressurized water-cooled nuclear power facility, components (including supports) that are classified as ASME Code Class 1, 2, and 3 must meet the requirements, except design and access provisions and preservice examination requirements, set forth in Section XI of editions and addenda of the ASME Code that become effective subsequent to editions specified in paragraphs (g)(2) and (3) of 10 CFR 50.55a and that are incorporated by reference in paragraph (a)(1)(ii) of 10 CFR 50.55a, to the extent practical within the limitations of design, geometry, and materials of construction of the components.

Pursuant to 10 CFR 50.55a(g)(6)(ii)(D), "Augmented ISI requirements: Reactor vessel head inspections," (1) "Implementation," requires all licensees of PWRs must augment their ISI program with ASME Code Case N-729-4, subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of 10 CFR 50.55a.

The regulations in 10 CFR 50.55a(z), "Alternatives to codes and standards requirements," state, in part, that "Alternatives to the requirements of paragraphs (b) through (h) of [10 CFR 50.55a] or portions thereof may be used when authorized by the Director, Office of Nuclear Reactor Regulation…. A proposed alternative must be submitted and authorized prior to implementation."

Section 50.55a(z)(1) of 10 CFR states that alternatives to the requirements of paragraph (g) may be used when authorized by the NRC if the "proposed alternative would provide an acceptable level of quality and safety."

Based on the above, and subject to the following technical evaluation, the NRC staff finds that regulatory authority exists for the licensee to request, and the NRC to authorize, the proposed alternative requested by the licensee.

3.0 TECHNICAL EVALUATION

3.1 Background

At STP, Units 1 and 2, the original RVCHs were replaced using penetration nozzles and associated J-groove welds with less susceptible primary water stress corrosion cracking (PWSCC) materials (i.e., Alloy 690TT nozzles and Alloy 52 weld materials). The STP, Unit 1, replacement RVCH was placed in service on November 18, 2009, and South Texas Project, Unit 2 replacement RVCH was placed in service on May 2, 2010.

3.2 Applicable ASME Code Edition and Components Affected

The current Code of record at STP, Units 1 and 2, for the third 10-year inservice inspection (ISI) interval is the 2004 Edition of the ASME Code. Additionally, starting after August 17, 2017, 10 CFR 50.55a(g)(6)(ii)(D)(1) requires, in part, that "Holders of operating licenses or combined licenses for pressurized-water reactors… shall implement the requirements of [ASME Code Case N-729-4]…."

The affected components are ASME Code Class 1 RVCH penetration nozzles and their associated J-groove welds made of PWSCC-resistant materials. In accordance with ASME Code Case N-729-4, Table 1, these welds are classified as Item No. B4.40 (Table 1). Each STP, Units 1 and 2, replacement RVCH has a total of 64 penetration nozzles. The licensee indicated that the replacement RVCHs at STP have Alloy 690TT penetration nozzles, which are welded to the ferritic RVCH by Alloy 52 filler metal. Alloy 690TT base materials and Alloy 52 welds are known to be more resistant to PWSCC than the previous head nozzle materials made of Alloy 600 and its associated weld materials Alloy 82/182. The licensee stated that based on actual temperature data gathered from several cycles of operation since the replacement RVCHs were placed in service, the average operating temperature is conservatively estimated to be no more than 585 degrees Fahrenheit (°F).

3.3 ASME Code Requirements for Which Relief is Requested

The regulation in 10 CFR 50.55a(g)(6)(ii)(D) requires that licensees for PWRs implement "the requirements of [ASME Code Case N-729-4]… subject to the conditions specified in paragraphs (g)(6)(ii)(D)(2) through (4) of [Section 50.55a(g)(6)(ii)(D)]." Specifically, ASME Code Case N-729-4, Item No. B4.40 (Table 1), requires that the RVCH penetration nozzles and their attachment partial penetration welds of PWSCC-resistant materials be subjected to volumetric or surface examination of all nozzles during every 10-year ISI interval (nominally 10 calendar years), if flaws attributed to PWSCC have not been identified.

3.4 Proposed Alternative

The licensee's proposed alternative is to extend the schedule and perform the volumetric and/or surface examinations for STP's replacement RVCH penetration nozzles and their attachment J-groove welds in spring of 2026 for STP, Unit 1 and spring of 2027 for STP, Unit 2. This would constitute a nominal 17-year period from the time when the replacement RVCHs were placed in service.

3.5 Licensee's Basis for Use of Alternative

The licensee's justification for use of the proposed alternative included the following topics of consideration:

- Electric Power Research Institute (EPRI) report Materials Reliability Program (MRP): "MRP-375, Technical Basis for Reexamination Interval Extension for Alloy 690 PWR Reactor Vessel Top Head Penetration Nozzles," dated February 2014 (ADAMS Accession No. ML14283A046), which provides technical justification to extend the volumetric/surface examination interval for PWSCC resistant RVCH nozzle penetrations from 10 years to 20 years.

- The inspection history at STP, Units 1 and 2, which includes preservice volumetric and bare metal visual (BMV) examinations of STP replacement RVCHs and current BMV examinations of the replacement RVCHs performed in accordance with ASME Code Case N-729-1 and Code Case N-729-4.

- Current available operating experience with Alloy 690/52/152 materials, which demonstrate a proven record of resistance to PWSCC over a span of more than 20 years.

The licensee states that the initial inception of the 10-year inspection interval contained in ASME Code Case N-729 for Alloys 690/52/152 was based on evaluations performed in 2004 to assess the improved PWSCC resistance of Alloys 690/52/152 relative to Alloys 600/82/182. The licensee noted that recent data for PWSCC crack growth rates for Alloy 690/52/152 materials (noted in MRP-375) suggest that those rates are significantly lower than assumed during those initial evaluations. The licensee performed a plant-specific calculation of the required factor of improvement (FOI) to support the proposed inspection interval of 17 calendar years. The purpose of this FOI is to determine the change in susceptibility of the nozzle and weld materials from the original head to the replacement using Alloy 600 materials for reference. As a plant specific FOI increases, the allowed interval between volumetric inspections for a head using Alloy 690 nozzles is equivalent in safety to the currently allowed volumetric inspection interval for a head using Alloy 600 materials. In this plant-specific FOI

calculation, the licensee used the actual temperature of the upper head at STP, Units 1 and 2, and conservatively assumed that calendar years were equal to effective full power years. Based on this calculation, and as documented in its submittal, the licensee determined that an FOI of 5.2 was required to meet the proposed inspection interval of 17 years. The licensee asserted that the required FOI of 5.2 was smaller than an FOI of 10, which supports extending the inspection interval to 20 years, and bounded essentially all of the MRP-375 crack growth rate data for Alloy 690/52/152.

The licensee stated that preservice volumetric examinations were performed on the replacement RVCH penetration nozzles and their attachment partial penetration welds prior to being placed in service at STP, Units 1 and 2. No recordable indications were identified during these examinations. Additionally, BMV examinations were performed in accordance with ASME Code Case N-729-1 and Code Case N-729-4, Table 1, Item B4.30. On the STP, Unit 1 replacement RVCH, these examinations were performed in 2009 (preservice visual examination), 2014, and 2018. Similar visual examinations were performed on STP, Unit 2 replacement RVCH in 2010 (preservice visual examination) and 2015. The licensee stated that these examinations did not reveal any surface or nozzle penetration boric acid indicative of nozzle leakage.

The licensee further stated that the current available operating experience with Alloys 690/52/152 provides approximately 24 years of proven record of resistance to PWSCC, demonstrated through numerous examinations. The licensee also stated that the Alloys 690/52/152 operating experience includes volumetric/surface examinations performed in accordance with ASME Code Case N-729 and includes heads that have operating temperatures that approached 613 °F.

The licensee concluded that, based on its analysis, the proposed alternate inspection frequency results in substantially reduced effect on safety when compared to the RVCHs with Alloy 600 penetration nozzles examined in accordance with the current requirements.

3.6 Duration of Relief Request

The licensee's proposed alternative overlaps the third and fourth 10-year ISI intervals. The third ISI interval is currently scheduled to end on September 24, 2020 and October 18, 2020, at STP, Units 1 and 2, respectively. The fourth ISI interval for STP, Unit 1, is scheduled to start on September 25, 2020, and end on August 20, 2027. The fourth ISI interval for STP, Unit 2, is scheduled to start on October 19, 2020, and end on December 15, 2028.

3.7 NRC Staff Evaluation

In its review of RR-ENG-3-23, the NRC staff considered the licensee's basis for use of the proposed alternative in accordance with 10 CFR 50.55a(z)(1), on the basis that the alternative examination frequency provides an acceptable level of quality and safety.

The NRC staff notes that the inspection frequencies developed in ASME Code Case N-729-4 for RVCH penetration nozzles made of Alloy 690/52/152 were developed, in part, based on conservative assessments of limited crack growth rate data and operating experience for these materials, available when the Code Case was initially issued. The licensee's assertion and technical basis is that the current available crack growth rate data now justifies a longer inspection interval and demonstrates a sufficient FOI for Alloy 690/52/152 material, when compared to Alloy 600/82/182 materials. This FOI then provides the basis for the extension of

the inspection frequency as proposed by the licensee in its alternative. The licensee calculated that it needed an FOI of 5.2, based on its proposed inspection interval and plant-specific operating temperature. The NRC staff independently verified that the licensee's requested alternate inspection interval is reasonably bounded by the licensee's calculated FOI, by using the parameters defined by ASME Code Case N-729-4. The NRC staff also verified that the plant-specific parameters the licensee used for its evaluation were conservative in that the licensee used calendar years as full power years and a conservative estimate for the upper head temperature.

In evaluating the licensee's technical basis for the proposed alternative, the NRC staff notes that the licensee relied, in part, on crack growth rate data from MRP-375. MRP-375 makes use of numerous Alloy 690/52/152 crack growth rate data from various sources to develop FOIs for the crack growth rate equations provided in MRP-55, "Materials Reliability Program (MRP) Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Thick-Wall Alloy 600 Material," and MRP-115, "Materials Reliability Program Crack Growth Rates for Evaluating Primary Water Stress Corrosion Cracking (PWSCC) of Alloy 82, 182, and 132 Welds," for Alloys 600/82/182. The NRC staff notes that although MRP-375 is not an NRC-approved report, the NRC staff finds the licensee's assertions and interpretations to be reasonable. Additionally, as the NRC staff has not validated all of the data reported in MRP-375, it does not consider it appropriate to limit the review of available data to only the crack growth data from MRP-375.

A more detailed review of the data provided in MRP-375 has been performed by an international group of experts as part of an Alloy 690 Expert Panel. The resultant report, "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)," has not been submitted for NRC review. Additionally, the NRC staff has noted some limitations in MRP-386, with respect to plant-specific relief requests. Therefore, the NRC staff found that the licensee's specific FOI cannot be justified by these reports alone. Instead, the staff noted Alloy 690/152/52 crack growth rate data from two NRC contractors, Pacific Northwest National Laboratory (PNNL) and Argonne National Laboratory (ANL). Specifically, the staff compared the licensee's information to data documented by memorandum dated October 30, 2014, transmitting preliminary PWSCC data (ADAMS Accession No. ML14322A587). The majority of the data from PNNL and ANL for Alloy 690 test samples were bounded by the licensee's FOI value of 5.2. The only PNNL and ANL data that was not bound by the licensee's FOI was that associated with weld dilution specimens. This means that certain crack growth rate tests of weld dilution samples would have an FOI of less than 5.2 versus the crack growth rate curves for Alloy 600 weld materials. Therefore, this data would not support the licensee's requested inspection frequency extension. However, the NRC staff did not consider the weld dilution zone data in its assessment of the licensee's proposed alternative. The NRC staff chose to exclude the weld dilution zone data from this analysis due to the variability in results and the limited area of continuous weld dilution for a flaw to grow. For example, in the case of the highest measured crack growth rates, a flaw would have to travel in the heat affected zone of a J-groove weld along the low alloy steel head interface. The NRC staff finds that the probability of a continuously accelerated crack growth in this small area of weld dilution zone is low, based on current limited test results. Therefore, the NRC staff found through risk insights that the probability of this failure path is low, and therefore, would not result in an increased risk of leakage or component failure.

In evaluating the licensee's second basis for use of the proposed alternative, the NRC staff finds that past BMV examinations on the head under consideration is a reasonable means to

demonstrate the absence of leakage through the nozzle or J-groove weld, or both, prior to the time the examination was conducted. The NRC staff also finds that performance of future BMV examinations in accordance with ASME Code Case N-729-4 requirements is adequate to demonstrate the absence of leakage at or prior to the time the examinations are conducted. Additionally, the NRC staff finds that the required frequency of the BMV examinations, in conjunction with the proposed alternative frequency for the volumetric or surface examinations, is sufficient to assure the structural integrity of the RVCH penetration nozzles and their associated J-grove welds at STP, Units 1 and 2. This is based on the current operating experience with periodic volumetric examinations and the BMV examinations. Specifically, the volumetric period examinations have been effective in identifying PWSCC, while the bare metal visual examinations have been effective in identifying minor leakage before compromising the structural integrity of the associated J-groove, the nozzle, or the RVCH.

In evaluating the licensee's other basis for use of the proposed alternative, the NRC staff finds that the current available operating experience supports the licensee's requested inspection schedule. Specifically, the NRC staff is not aware of any service-related PWSCC occurring with Alloy 690/52/152 materials.

In summary, the NRC staff finds the licensee's calculated FOI of 5.2 supports an extension of the Code Case N-729-4 volumetric inspection frequency to 17 calendar years. The NRC staff independently verified that the licensee used conservative plant specific parameters for its evaluation. Additionally, the NRC staff finds that the periodic BMV examinations for the STP, Units 1 and 2 replacement RVCHs provide reasonable assurance of structural integrity. Furthermore, the NRC staff agrees with licensee's assertion that the current operating history with Alloy 690/52/152 materials supports its proposed inspection interval.

Therefore, the NRC staff finds that the licensee has provided adequate technical basis to demonstrate that its proposed alternative examination frequency of 17 calendar years would provide an acceptable level of quality and safety.

4.0 CONCLUSION

As set forth above, the NRC staff determines that the licensee's proposed alternative provides an acceptable 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)(1). Therefore, the NRC staff authorizes RR-ENG-3-23 at STP, Units 1 and 2, until the end of the fourth inservice inspection interval, which is scheduled to end on August 20, 2027, for Unit 1, and December 15, 2028, for Unit 2.

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

Principal Contributor: R. Kalikian, NRR

Date: August 13, 2019

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - RE: REQUEST FOR RELIEF REQUEST RR-ENG-3-23 TO EXTEND VOLUMETRIC EXAMINATION INTERVAL FOR REACTOR VESSEL CLOSURE HEAD NOZZLES (EPID L-2019-LLR-0021) DATED AUGUST 13, 2019

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