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

April 20, 2011

Mr. Edward D. Halpin
President and Chief Executive Officer/
Chief Nuclear Officer
STP Nuclear Operating Company
South Texas Project
P. O. Box 289
Wadsworth, TX 77483

SUBJECT: SOUTH TEXAS PROJECT, UNITS 1 AND 2 - REQUEST FOR RELIEF RR-ENG-3-01 FROM ASME SECTION XI CODE REQUIREMENTS FOR STEAM GENERATOR NOZZLE NON-DESTRUCTIVE EXAMINATION (TAC NOS. ME4766 AND ME4767)

Dear Mr. Halpin:

By letter dated September 20, 2010, as supplemented by letter dated December 2, 2010, STP Nuclear Operating Company (the licensee) submitted a request for relief (RR-ENG-3-01) from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, non-destructive examination requirements applicable to the South Texas Project (STP), Units 1 and 2, steam generator (SG) main steam nozzle inside radius sections. The request is for the third 10-year inservice inspection (ISI) interval for both units. For STP, Unit 1, the ISI interval began on September 25, 2010, and ends on September 24, 2020; for STP, Unit 2, the ISI interval began on October 19, 2010, and ends on October 18, 2020.

The U.S. Nuclear Regulatory Commission (NRC) staff has completed the review of the subject relief request. Based on the enclosed safety evaluation, the NRC staff concludes that inner radius volumetric examination of the SG main steam outlet nozzle is impractical. The results of the NRC staff evaluations indicate that conditions conducive to crack initiation are absent, the nozzle stresses under different loading conditions are acceptable, and there is reasonable assurance that degradation of the subject SG main steam inner nozzle radius is not likely. Granting relief is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest giving due consideration to the burden upon the licensee that would result if the requirements were imposed on the facility. Therefore, relief is granted pursuant to paragraph 50.55a(g)(6)(i) of Title 10 of the *Code of Federal Regulations* for the third 10-year ISI interval at STP, Units 1 and 2.

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

If you have any questions, please contact the project manager, Balwant Singal, at 301-415-3016 or via e-mail at Balwant.singal@nrc.gov.

Sincerely,

James R. Hall for

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

Docket Nos. 50-498 and 50-499

Enclosure:
As stated

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

REQUEST FOR RELIEF FROM ASME SECTION XI CODE REQUIREMENTS FOR

STEAM GENERATOR NOZZLE NON-DESTRUCTIVE EXAMINATION

DURING THIRD 10-YEAR INSERVICE INSPECTION INTERVAL

SOUTH TEXAS PROJECT, UNITS 1 AND 2

STP NUCLEAR OPERATING COMPANY

DOCKET NOS. 50-498 AND 50-499

1.0 INTRODUCTION

By letter dated September 20, 2010 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML102700174), as supplemented by letter dated December 2, 2010 (ADAMS Accession No. ML103470281), STP Nuclear Operating Company (STPNOC, the licensee) submitted a request for relief (RR-ENG-3-01) from the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), Section XI, non-destructive examination requirements applicable to the South Texas Project (STP), Units 1 and 2, steam generator (SG) main steam nozzle inside radius sections. ASME Code, Section XI, Table IWC-2500-1, Code Category C-B, Item Number C2.22, requires that a volumetric examination of the inner-radius section of nozzles at terminal ends of piping runs be performed each inspection interval. The licensee has requested relief from the ASME Code, Section XI requirement for a volumetric examination of the inner-radius section of the STP, Units 1 and 2, SG main steam outlet nozzle because compliance with the requirement is impractical and the configuration of the nozzle precludes obtaining meaningful volumetric examination results. No alternative examination is proposed in lieu of the volumetric examination because the visual examination of the subject area is also precluded by the configuration. The request is for the third 10-year inservice inspection (ISI) interval for both units. For STP, Unit 1, the ISI interval began on September 25, 2010, and ends on September 24, 2020; for STP, Unit 2, the ISI interval began on October 19, 2010, and ends on October 18, 2020.

2.0 REGULATORY EVALUATION

The inservice inspection (ISI) of ASME Code Class 1, 2, and 3 components is to be performed in accordance with Section XI of the ASME Code and applicable edition and addenda as required by Title 10 of the *Code of Federal Regulations* (10 CFR) 50.55a(g), except where specific relief has been granted by the U.S. Nuclear Regulatory Commission (NRC) pursuant to

Enclosure

10 CFR 50.55a(g)(6)(i). Pursuant to 10 CFR 50.55a(a)(3), alternatives to the requirements of paragraph (g) may be used, when authorized by the Commission, if (i) the proposed alternatives would provide an acceptable level of quality and safety or (ii) compliance with the specified requirements would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Pursuant to 10 CFR 50.55a(g)(4), ASME Code Class 1, 2, and 3 components (including supports) must meet the requirements, except design and access provisions and preservice examination requirements, set forth in 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. The regulations require that inservice examination of components and system pressure tests conducted during the first 10-year interval and subsequent intervals comply with the requirements in the latest edition and addenda of Section XI of the ASME Code incorporated by reference in 10 CFR 50.55a(b) 12 months prior to the start of the 120-month interval, subject to the limitations and modifications listed therein. The regulations in 10 CFR 50.55a(g)(4)(iv) states that inservice examination of components and system pressure tests may meet the requirements set forth in subsequent editions and addenda that are incorporated by reference in paragraph 10 CFR 50.55a(b), subject to the limitations and modification listed in 10 CFR 50.55a(b) and subject to Commission approval. Portions of editions or addenda may be used provided that all related requirements of the respective editions or addenda are met. The ASME Code of record for the third 10-year ISI interval at STP, Units 1 and 2 is the 2004 Edition with no Addenda.

3.0 TECHNICAL EVALUATION

3.1 Affected Components

The component affected by this request is the SG main steam outlet nozzle.

3.2 ASME Code Requirements (as stated by the licensee)

ASME [Code,] Section XI, Table IWC-2500-1, [Examination] Category C-B, Item No. C2.22, requires that a volumetric examination of the inside-radius section of nozzles at terminal ends of piping runs be performed in accordance with Figure IWC-2500-4(a), (b), or (d) each inspection interval. The applicable nozzles include those welded to or integrally cast in vessels that connect to piping runs selected for examination under Examination Category C-F, excluding manways and handholes.

3.3 Licensee's Basis for Requesting Relief (as stated by the licensee)

In accordance with the provisions of 10 CFR 50.55a(g)(5)(iii), the STP Nuclear Operating Company (STPNOC) requests relief from Section XI [of the ASME Code] requirement for a volumetric examination of the inside-radius section of the [STP] Unit[s] 1 and 2 steam generator main steam outlet nozzles because compliance with the requirement is impractical. The configuration of nozzles precludes obtaining meaningful volumetric examination results.

Each South Texas Project steam generator main steam nozzle is a one-piece forging with an insert that serves as a flow restrictor comprised of seven nozzles in parallel with the outlet centerline. Due to their flow restrictor-type design, the South Texas Project steam generator main steam outlet nozzles do not contain a high-stress inside-radius section for which the [ASME Code] Section XI volumetric examination is intended. As depicted in the attached drawings [drawings attached to the licensee's letter dated December 2, 2010], the design configuration of the outlet nozzle does not correspond to Figures IWC-2500-4(a), (b), or (d) specified by Item Number C2.22, and does not have a "radius bend". Furthermore, the outlet nozzle configuration precludes a meaningful ultrasonic examination in the area of the inner radius. Because of the design configuration, a volumetric examination would not provide any meaningful results.

Figures 1 and 2 and the attached tables [attached to the licensee's letter dated December 2, 2010] provide stress data for the steam generator outlet nozzle region. The exit ends of the nozzles are not part of this relief request.

3.4 NRC Staff Evaluation

The SG main steam outlet nozzle is an integral part of the head forging. The nozzle has seven 8.5-inch diameter holes drilled for steam exiting into the 29.37-inch inside diameter piping system. The nozzle surface facing inside the SG head is stainless steel cladded. Inserted into the holes are forged stainless steel flow limiters that are welded to the nozzle and SG head surface cladding. The flow limiters extend beyond the pipe side of the holes. Accessibility to the pipe side of the nozzle for examination is restricted by the flow limiters, the narrow gap between holes, and proximity of the attached piping. On the inside SG surface, the configuration of the flow limiters overlapping the hole completely covers the radius. The SG main steam outlet nozzle design is not conducive to any meaningful volumetric examination. For the licensee to examine the inner radius, the outlet nozzle design in the SG head would have to be modified and replaced resulting in associated radiation exposure to maintenance personnel. The configuration of nozzles precludes obtaining meaningful volumetric examination results.

The requirement for examinations of inner nozzle radii is associated with the discovery of cracks located in the inner radius section of feedwater nozzles. The cracks were identified as cycle thermal fatigue from internal water temperature fluctuation. The NRC staff considered the likelihood of this degradation mechanism on the inner radius of the main steam outlet nozzle.

The main steam outlet nozzle is located at the top of the SG head. The steam vapor at this location has traveled through the dryers which removes excess water droplets. During plant operations, this location is subjected to relatively constant, high temperature which prevents the accumulation of liquid moisture at the pipe-to-nozzle location. The only temperature fluctuations are associated with reactor heatup and cooldown, and these are controlled for the thermal effects on components.

The licensee performed stress analysis for different operating loading conditions. These analyses show that nozzle stresses are maintained below ASME Code allowable values. The absence of thermal fluctuations, standing water, and stresses above allowable values during operating conditions minimize the likelihood of cycle thermal-fatigue cracking of the main steam outlet nozzle inner radius.

Based on the absence of conditions conducive to crack initiation and acceptable stress under loading conditions, the NRC staff concludes that there is reasonable assurance that degradation of the subject inner nozzle radius is unlikely to occur. With the inaccessibility of the main steam outlet nozzle inner radius, the replacement of the SG outlet nozzle design with a design that is conducive to volumetric examination is impractical.

4.0 CONCLUSION

Based on the above, the NRC staff concludes that compliance with the ASME Code, Section XI, Table IWC-2500-1, Examination Category C-B, Item No C2.22 inner radius volumetric examination of the SG main steam outlet nozzle is impractical. Also, the configuration of nozzles precludes obtaining meaningful volumetric examination results. The licensee performed stress analysis for different operating loading conditions. These analyses show that nozzle stresses are maintained below ASME Code allowable values. The absence of thermal fluctuations, standing water, and stresses above allowable values during operating conditions minimize the likelihood of cycle thermal-fatigue cracking of the main steam outlet nozzle inner radius.

Therefore, pursuant to 10 CFR 50.55a(g)(6)(i), request for relief RR-ENG-3-01 is granted to STP, Units 1 and 2, for the third 10-year ISI intervals which are scheduled to end September 24, 2020, for Unit 1 and October 18, 2020, for Unit 2. Granting relief pursuant to 10 CFR 50.55a(g)(6)(i) is authorized by law and will not endanger life or property or the common defense and security, and is otherwise in the public interest giving due consideration to the burden upon the licensee that would result if the requirements were imposed on the facility.

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

Principal Contributor: Don Naujock

Date: April 20, 2011

If you have any questions, please contact the project manager, Balwant Singal, at 301-415-3016 or via e-mail at Balwant.singal@nrc.gov.

Sincerely,

/RA by James R. Hall for/

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

Docket Nos. 50-498 and 50-499

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